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Astronuclear Laboratory

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WANL-TME-573

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INFORMATION CATEGORY CRD	
W. H. Arnold Jr.	10/21/63
Authorized Classifier	Date

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SPECIAL REREVIEW FINAL DETERMINATION	Reviewer	Class.	Date
	KAW	U	3-8-82

GROUP - 1
EXCLUDED FROM AUTOMATIC DOWNGRADING AND DECLASSIFICATION

SUBMITTED BY:

Westinghouse Electric Corporation
Astronuclear Laboratory
Pittsburgh 36, Pennsylvania

PREPARED BY:

NERVA Project Sponsors:

Classification cancelled (or changed to) _____
by authority of _____
by W.F.C. TIC, date SEP 11 1973

APPROVED BY:

W. H. Arnold Jr.

W.H. Arnold, Jr.

CONTRACT YEAR 1963
TECHNICAL
SUMMARY PRESENTATION
October 22-23, 1963

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WANL-TME-573

AGENDA

C.Y. '63 CONTRACT SUMMARY MEETING

TECHNICAL PRESENTATIONS

October 22-23, 1963

SPONSORS

S. Krasik
J. Kenney
F. Retallick
D. Drawbaugh
E. De Zubay
D. Wiebe
F. Spurrier
F. Lorenz
C. Meuche
A. Boltax
F. Montgomery
N. Pollack
F. Henning
W. Anthony

PROJECTS

Introduction
1310, 1311, 1315, 1390
1320, 1331, 1332, 1333, 2500
1334
1343, 1352
1313
1314, 1316
1312
1350, 1351
1353
1361
1363
2600
2610

Tour: WANL, WANEF and WANHEF Facilities

H. Kirschbaum	1340, 1341, 1342, 1345, 1346, 2200, 3300, 3600
R. Ramp	1344, 1347
W. H. Arnold, Jr.	1391, 2710, 2731, 2732, 2733, 2734
G. R. Thomas	1380
W. B. Henderson	1381, 1382, 2800, 3000
A. Poindexter	2100, 2300, 3200, 3400, 3500, 3600, 3800
S. S. Stein	1980, 1981
J. Koetting	1360, 4200
F. Greehan	4100

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ABSTRACT

The present document is a transcript of presentations by the sponsors of the NERVA Nuclear Subsystem projects described in WANL Work Program WANL-WP(J)-017 (May 17, 1963). These talks were given at a meeting held at WANL on October 22 and 23, 1963 to review program progress in Contract Year 1963. The agenda following gives the order in which the projects were discussed, which is also the order in which they were printed. Projects 2900, 4101, 4300, and 5000 were not discussed. A bibliography of external reports issued during the contract year follows the talks.

INTRODUCTION

I should like to welcome you to the Astronuclear Laboratory. As you are all aware the purpose of this meeting is to present to you, at your request, a summary review of our accomplishments on the NERVA Project for Contract Year 1963. The purpose of the review is to assist you in your evaluation of our performance for Contract Year 1963.

We have given considerable thought as to how this might best be done. None of the ideas for organizing the presentation appeared to meet all of our objectives. The one we have chosen, however, does meet many of them and we believe represents a reasonable compromise. The meeting will be divided into two principal sessions: a technical review of the sub-tasks, and an over-all administrative review. The technical review will take one and one-half days of the two days allocated for the meeting; the administrative review one-half day.

The time available for the various speakers is quite limited and for the technical review no speaker is allowed more than thirty minutes and many have substantially less. It is essential that we maintain our schedule if we are to complete the formal presentation in two days.

We have been asked whether questions may be asked of the speakers and we have agreed that up to five minutes may be used for questions at the close of each speaker's presentation. We do ask that you refrain from interrupting the speaker during his presentation and reserve your questions until his talk is completed.

We intend to furnish copies of the transcript of each speaker's formal presentation, including figures, in a bound volume. We hope to have these available for distribution at the close of the second day. Because of reproduction problems we cannot plan to have these books completed sooner. Since each speaker will actually read his presentation, there should be no requirement for note taking.

We will attempt during the short question period to record the question and the speaker's response. We would hope that this will permit us to issue an addendum some time in the near future which will cover the questions and answers. Where the five minute time period may not permit all questions to be answered, we would attempt to record all of the questions and prepare answers which would also be given in the addendum. Thus if you have questions, please ask them or turn them into the chairman so that replies may be prepared.

During the technical portion of the presentation the speakers will confine themselves to a discussion of the technical work program and accomplishments. Most of the speakers will indicate specifically those milestones which are not complete. They will not, however, tick off all of the milestones which have been completed. We hope to have a complete summary of all milestones and their status incorporated in the administrative section of our report.

While we recognize the interest in costs and carryover associated with each sub-task, it is our plan to consolidate this portion of the review and present it during the administrative session. The reason for this is that the cost data are complex and interrelated. We believe it would be completely hopeless to attempt to cover this review in two days in any effective manner and introduce cost and carryover information into each of the technical presentations. We ask, therefore, that you withhold questions relating to sub-task cost, carry-over effort levels, material costs, etc., until that part of the presentation is reached.

We are attempting by this procedure and agenda to cover as much of the pertinent material as we can in the two day formal session. Clearly you are free, as always, to discuss with the individual project sponsors matters which you wish to pursue in greater detail or on a more informal basis.

It is of course impossible to cover all of the points which should be made in the available time. We have tried to cover those which we believe are of greatest importance.

Others which may be of equal or greater importance will probably be raised by your questions. Others may be overlooked during this meeting and may arise in your evaluation. In such event, if the situation warrants, we would intend to respond by submittal of further information following our receipt of the evaluation document. The point which I am making is simply that we do not intend to waive any contractual rights to which we are entitled by this formal presentation.

One further point. I have mentioned previously that certain milestones in our work program are incomplete. I want to make it completely clear, however, that in a contractual sense it is our position that all tasks required by the contractual work statement are complete, without exception. In discussing the milestones we are discussing these solely from the point of view that these represent a measure of the planned work program. From an evaluation point of view, therefore, the milestones are useful targets with which to compare our target objectives and actual accomplishments.

One word of caution when you read the text of the speaker's transcripts. You will find that these are indeed transcripts of talks and are not polished and edited reports. In many cases you will note certain localisms peculiar to WANL. We ask that you overlook these flaws and recognize the material we are furnishing is simply a transcript.

A final word on arrangements. Dr. Arnold will chair the technical portion of the meeting and will be responsible for maintaining the orderly progress of the meeting. We have tentatively planned to stop the meeting late in the afternoon to provide for a short laboratory tour at the Large Site. We would then plan to proceed by bus to inspect our facilities at the Waltz Mill Site and from there proceed to dinner. In order to make up for the time taken up by the tour, we plan to have an evening session at this site after dinner.

Dr. Arnold will now take the chair and proceed with the planned agenda.

NE-1310

NRX-A REACTOR MECHANICAL DESIGN

PURPOSE: "To establish the layout and detail design of flight-type reactors based on the KIWI B-4 for application in NRX-A tests. To establish, on a continuing basis, redesigns required for the development to a qualified flight status. To establish assembly procedures and the design of supporting equipment, such as assembly stands and shipping containers. "

NRX-A BLOCK I REACTOR:

Figure I shows some of the major events associated with the development of Block I reactors.

Much of the preliminary design work was completed in Contract Year 1962 so that by October 15 all layouts defining an uninstrumented top-loaded design were completed. The uninstrumented detail drawings were released just one month later on November 15 for hardware procurement. Following this release, we began the task of completing the assembly drawings and the parts list for the uninstrumented reactor. In parallel with this effort, we started the drawing modifications required for instrumenting NRX-A1, which at that time was a hot test reactor.

As you will recall, the reactor released November 15 had a single exhaust end seal and a significant flow of coolant fluid along the core periphery. Thermal analysis of the design conducted in parallel with the release indicated that there would be unacceptably severe temperature gradients along the core periphery. About the same time, tests conducted by Engineering Mechanics indicated a much lower coefficient of friction than had been assumed for designing the lateral support system. As a result, we were in the process of revising the core periphery by making a dual seal at the exhaust end, adding a seal at the inlet end and increasing the spring bundling force when KIWI B-4A was tested.

After a number of reviews of the KIWI test and the NRX-A design, it was decided that the core periphery should be redesigned to minimize the thermal gradient problem and

to optimize core bundling.

While this re-evaluation of the NRX-A design was in progress, approximately 100 drawings dealing with the NRX-A instrumentation were released for procurement. Work on the instrumented version of the November 15 design was stopped when it became apparent that the core periphery would be redesigned.

In early February, the NERVA program was redirected. The Block I portion of the program was expanded to include both the reactor vibration and cold flow tests which were to be conducted in advance of the first hot test. At the same time, the three hot tests were retained in the program so that the redirection provided for four Block I reactor tests in addition to an expanded component test program. Although much of the preceding work was salvagable, the fact that NRX-A1 became a cold flow test reactor required extensive rework of almost all drawings, particularly those associated with instrumentation.

On March 1, the uninstrumented detail drawings of NRX-A reactors having a re-designed core periphery were released for procurement. Basically, these drawings defined the NRX-A1 and Vibration Phase I reactors.

The instrumentation for NRX-A1 was defined in May, and detail drawings of instrumented components were released in July.

The core periphery redesign was the major design effort conducted during Contract Year 1963. The major objectives of this redesign were as follows:

- (1) Limit the access of cool gas to the core periphery to minimize leakage into the core, thereby reducing potential thermal stress problems.
- (2) Insure that the pressure between fueled elements will be no higher than the external pressure on the bundled core, thereby eliminating element separation and fluttering.
- (3) Bundle the core so that it behaves as a unit without unacceptable restraint of axial and radial thermal expansion.
- (4) Develop a design sufficiently flexible to accommodate changes in bundling load, sealing, and fluid flow patterns simply by altering of parts rather than by procuring new hardware.

These objectives were accomplished by adding a series of distributed seals along the

core periphery. These seals substantially reduced the flow of coolant flow and raised the bulk temperature of the fluid, thereby easing the thermal gradient problems at the core periphery. Another significant change was the addition of a band which bundled the inlet end of the core independently of the lateral support system.

The redesign of the core periphery is reported in WANL-TME-411.

Once the thermal and fluid flow effects of the periphery redesign were known, it was then necessary to run the corresponding stress and vibration analysis. An appreciation of how the scope of this work increased can be had by examining the documentation records which show that while only one formal analysis was released by Mechanical Design during the first half of Contract Year 1963, fourteen reports were released during the last half of the year. This same high-level of activity continues throughout the first quarter of Contract Year 1964, with approximately 15 reports scheduled for completion. The reports covered such major areas as fuel elements, the lateral support system and core eccentricity.

At the close of the calendar year, we had completed all the tasks originally associated with designing the NRX-A1 and had made substantial progress in defining the design of all Block I reactors.

ASSEMBLY STANDS AND FIXTURES:

This brings us to a consideration of the second major area of work conducted on Project 1310, assembly stands and fixtures design. There is, logically, a close inter-relationship between the reactor to be assembled, the tools required to assembly it and the directions written to assist those responsible for the assembly. Ideally, a piece of equipment will be defined and then the tools and methods required to handle and assemble it will be developed, i.e., there is normally a product design-equipment design-assembly manual design sequence in developing products and equipment. Because of schedule commitments, however, the inter-relationship in this instance was somewhat unique in that, to a great extent, the design of the assembly stands and equipment was defined in parallel with the design of the reactor. As a consequence, these items were far more sensitive to change than normally would have been the case. For example, the redesign of the core periphery and the re-definition of instrumentation placement all had direct impact on, and necessitated substantial changes in, the

assembly stands and fixtures and assembly procedures.

Figure 2 shows some of the major events related to development of these tools.

The original work on the assembly stands was directed toward assembly of the design released November 15. The designs of these stands were completed and the drawings released for procurement in November 1962. Approximately 50 stand and fixture drawings were released at that time.

Following release of the drawings, work was started on the reactor assembly procedures manual. This work stopped in mid-January when the core periphery redesign began. When the core periphery redesign was finally released, all of the major stand and fixture drawings (approximately 50 in number) were either reworked or replaced by new drawings. The assembly equipment redesign was completed in March and all drawings were released by April 1.

Shortly after work was resumed in April, the first assembly procedure manual was completed. In all, four manuals were released: Preliminary NRX-A Assembly Procedures, Phase I Vibration Assembly Procedures, and NRX-A1 and NRX-A2 Assembly Procedures.

Five major stands were designed and built during the year: the outer reflector stand, inner reflector stand, core and reactor stand, turning stand, and instrumentation, shimming and re-orificing stand. In addition, numerous auxiliary stands, fixtures and tools were designed and released for procurement. This equipment was fabricated, tested, installed and successfully used to assembly the NRX-A Assembly mockup and the Phase I Vibration reactor. The cold flow test reactor (NRX-A1) is now being assembled.

Following the redesign of the assembly stands and fixtures, a design report (Assembly Stand Design Report, WANL-TNR-124) was issued in September describing the design philosophy, equipment and stress analysis of each piece of equipment.

SHIPPING CONTAINER DESIGN AND DEMONSTRATION SHIPMENT:

This brings us to the last major objective of Project 1310, development of a shipping container and demonstration of its effectiveness. Figure 3 shows the six major steps taken to fulfill this objective.

The design specification for the NRX-A shipping container was prepared in October, 1962 and released for procurement bidding in November. The design criteria and related special requirements developed for the shipping container itself were based on design parameters defined in Westinghouse Equipment Specification 675351. Mechanical Design participated in the technical evaluation of vendor bids and continued to supply technical support during the design phase by reviewing and approving all vendor design drawings and calculations, witnessing the proof tests at the vendor's plant, and evaluating the results of the impact and pressure tests. The container qualification tests are documented in WANL-TNR-116, NRX-A Shipping Container Design Report.

The completed container was received at WANL in June 1963. Subsequently, it was used to ship an instrumented, simulated reactor from WANL to NRDS and back. This movement combined transportation by both rail and truck.

It was concluded from these demonstrations that the transportation environmental loads are of low magnitude and are well within the range indicated in the literature. It was further concluded that the maximum accelerations loads imposed on the reactor were well within both the shipping container and reactor design criteria. A detailed discussion of the shipping demonstration is given in WANL-TME-458, published in August.

SUMMARY:

With one exception, NRX-A Design Report, all milestones for Project 1310 were completed in Contract Year 1963. This report, which was scheduled to be issued in September, was written in September. However, the incorporation of comments received on the final draft have been delayed due to effort associated with NRX-A1 assembly.

NE-1311

REACTOR SHIELD MECHANICAL DESIGN

PURPOSE: "Initially, Project 1311 had as its major objective the definition of the mechanical design of a reactor shield which would meet the environment and radiation attenuation requirements imposed during flight. This shield was to be based on the KIWI B-5 shield designed in Contract Year 1962. The project also provided for the continuation of KIWI B-5 shield effort associated with design reports and stress analysis."

SUMMARY:

Early in 1963 the shield program was redirected such that the KIWI B-5 type shield was deleted from the Block I program and replaced by an aluminum unit.

Figure 4 shows some of the major events related to shield development.

KIWI B-5 shield reports were issued in December of 1962 and March of 1963. The layouts defining the NRX-A shield based on KIWI B-5 were released in October 1962, followed by uninstrumented details in November and instrumented details in February 1963. The layout of the aluminum replacement shield unit was completed in March. In May, the detail drawings were released and by July all instrumentation drawings had been issued. The design report, which included stress and vibration analysis of the shield, was also released in July.

All shield milestones were completed by July of 1963.

NE-1315

E-ENGINE REACTOR MECHANICAL DESIGN

No work was performed under this project.

NE-1390

REACTOR MODELS AND MOCKUPS

PURPOSE: "To provide partial models and mockups of reactor structural and core parts and complete models and mockups of reactor designs to assist in engineering analysis and for appraising mechanical arrangements."

PARTIAL MODELS AND MOCKUPS:

Numerous partial models and mockups were completed during the year to assist design engineers in evaluating component parts and developing solutions to problems. Typical models included instrumented components, locking devices, lateral support assembly, filler strips and support blocks.

COMPLETE MODELS AND MOCKUPS:

Two major mockups were constructed: the assembly checkout mockup and the instrumentation mockup. In addition, half-scale models were made of KIWI B-4D and NRX-A1 for delivery to SNPO.

The major purpose of the assembly checkout mockup was to isolate and resolve problems relating to the interface dimensions of NRX-A1 reactor components, to check out assembly stands and fixtures, to provide training for assembly personnel, and to evaluate design changes. The assembly mockup was used to evaluate the NRX-A1 assembly procedure prior to starting assembly of the reactor and several changes in procedure were adopted as a result of this activity.

Figure 5 shows two of the major components of the assembly checkout mockup; the outer reflector and the graphite reflector.

The instrumentation mockup was used primarily to resolve instrumentation routing problems, especially the bundling and harnessing of leads. A mockup is the only practical way to develop instrumentation routing since it is essentially a trial and error procedure which does not lend itself to solution by layout. Furthermore, it is very important that routing be clearly defined prior to instrumenting a reactor to minimize bending and handling of the actual instrumentation leads. Figure 6 shows the core support plate mockup after leads

were properly routed and fastened in place. Figure 7 shows the major mockup during the initial phases of routing NRX-A1 instrumentation.

Figure 8 shows the half-scale models of KIWI B-4D and NRX-A1 recently delivered to SNPO. These models, like the others, are multi-purposed tools which can be used for orientation, design evaluation, problem solving and many other related functions.

There were five milestones in Project 1390, all of which were completed in Contract Year 1963.

MECHANICAL DESIGN SUMMARY

So far I have touched on the major events that occurred during Contract Year 1963. We traced our progression from the conceptual stage through the definition of the NRX-A1 reactor design and discussed the related work that had to be accomplished to achieve our goals. Except for the printing of one report, the thirty-four milestones that were established were met. In order to complete these 34 major events, literally thousands of other things had to happen at a relatively fixed point in time along the way. In addition, all of this work was documented, and this documentation required a significant portion of the total Mechanical Design effort. A full listing of the major reports issued during Contract Year 1963 is attached to this document.

I should like to spend the last few moments this morning discussing the quantitative type things that were done and analyzing their significance from a sense of achievement, learning and problem solving viewpoint.

Figure 9 presents one of those "sheer weight of numbers" pictures which serves two purposes: first, it gives us a general picture of the kind and volume of information that must be generated to make a program such as this "go", and second, it serves as a good spring board from which we can evaluate a few more significant, but not so readily apparent, facts.

A first glance at this figure tells us that during this past Contract Year, Mechanical Design released 2400 drawings, processed 2000 drawing revisions, and reviewed some 3200 VR's and AR's to carry out the necessary business of building a reactor. Now what does this tell us? First it tells us that a lot of work was done, and secondly, it infers that much rework was done - there was a good deal of forward two steps, backwards one. A substantial part of this rework came about because we were doing things in parallel that traditionally would be done in sequence and because we had to define all aspects of the effort from creation of the design to selection and orientation of vendors who were capable of translating the design into acceptable hardware.

We feel that we have taken several major steps to compensate for these unusual conditions. One of these steps, and a major one because it is related to one of our major continuing problems - drawing releases - would be worth examining at this point. The flow pattern by which we evolve a design drawing is shown in somewhat oversimplified form on Figure 10.

Basically, this procedure calls for creation of conceptual layout drawings to study various approaches which may fulfill the requirements of the design task. These layouts are coordinated informally with other design activities to establish acceptable approaches to the design. This layout activity then continues until one or more designs are created which merit more intensive study.

Next, detail layouts of component assemblies are prepared to investigate the merits of the design concept. Supporting calculations of a preliminary nature are made to aid in the selection of materials, the sizing of flow paths, and the selection of physical dimensions. The general overall configuration and interfaces are studied on these layouts.

Various study layouts, which supplement the component design layouts, are made during this time to investigate the effects that tolerances, fabrication problems, and assembly techniques may have on the design.

A master design layout is initiated as soon as the design, as defined by component layouts, begins to firm up. This drawing is based on nominal dimensions taken from the various component and assembly layouts. The master layout is used to disclose any discrepancies between dimensions on the various component layouts.

After a design has been studied in sufficient depth to satisfy preliminary technical requirements, the assembly and component layouts are released to other activities, such as Reactor Analysis, Materials Development, Manufacturing Engineering and Reliability and Quality Control, for review and comment. Following this review, which had much greater depth than earlier design reviews, the component parts are released on detail drawings for hardware procurement.

To check on the accuracy of detail drawings, a tolerance stack-up layout is prepared, using the detail drawings as the source of the dimensions. This layout establishes the fits and clearances between parts and verifies the correctness of the dimensions which define the detail parts. The detail drawings are not presumed to be correct until after any errors or discrepancies which are discovered on the tolerance stack-up drawing are resolved and incorporated on the respective detail drawings. The original component and assembly layouts are used for investigating drawing changes; the stack-up layout is not changed until design change has been approved and the detail drawing has been revised.

Although considerable analysis is performed in parallel with the layout design and detail drawing phases of the engineering effort, concentrated engineering analysis normally follows the release of detail drawings. As a result, considerable changes in detail design can be expected after "final" detail drawings have been released.

So far we have presented a sequential picture of how the mechanical design effort flowed during the year, and a brief glimpse of the quantity of work produced. Logically, we like to know where we are going and how we can best get there. In addition, we like to keep an account of what it costs us to get our results.

Figure II shows how Mechanical Design manpower was expended during several

arbitrarily chosen periods over the past several months. This kind of sampling, which is taken periodically, allows us to judge whether our efforts are properly balanced and whether the accent at a given moment is being placed where it ought to be. As you can see, mechanical design and analysis accounted for approximately 25% of our time; the remaining time was spent in the necessary peripheral activities such as material review, coverage of assembly operations and, of course, maintaining the tremendous volume of paper work generated in projects such as this.

As a closing thought, I think it would be well to repeat a recurrent theme which has sounded throughout these remarks. Essentially, the magnitude and impact of a change can often be lost on a person if he isolates that change to the one thing being modified. If we say "let's redesign the core periphery" and we isolate our view to thinking that this will take X-number of changes in X-number of components, the job is dangerously over-simplified. These direct changes set off a chain reaction body of changes in everything from assembly equipment to instrumentation. I am confident that the experience gained in evaluating the related changes that come about as the result of direct changes will enable us to do a better job of defining "the whole job" during Contract Year '64.

MAJOR EVENTS: BLOCK 1 REACTOR DEVELOPMENT

EVENT	1962			1963								
	O	N	D	J	F	M	A	M	J	J	A	S
INITIAL DESIGN												
LAYOUTS COMPLETED	△ 15											
UNINSTRUMENTED DETAILS COMPLETED		△ 15										
INSTRUMENTED DETAILS RELEASED				△								
KIWI B-4A TEST		△										
CORE PERIPHERY REDESIGN												
DESIGN STUDIES CONDUCTED		△		△								
LAYOUTS COMPLETED								△ 1				
UNINSTRUMENTED DETAILS COMPLETED								△ 1				
INSTRUMENTED DETAILS COMPLETED										△		

595400A

MAJOR EVENTS: ASSEMBLY STANDS AND FIXTURES DEVELOPMENT

EVENT	1962			1963								
	O	N	D	J	F	M	A	M	J	J	A	S
DRAWINGS RELEASED		△ 15										
DRAWINGS REVISED							△ 1					
ASSEMBLY PROCEDURES MANUALS RELEASED							△ 30	△ 30	△ 30	△ 30		
DESIGN REPORT RELEASED												△ 30

595401A

MAJOR EVENTS: SHIPPING CONTAINER DESIGN AND DEMONSTRATION SHIPMENT

EVENT	1962			1963								
	O	N	D	J	F	M	A	M	J	J	A	S
DESIGN SPECIFICATIONS COMPLETE	△ 20											
MONITOR VENDOR ACTIVITY		△ 1	△	△	△	△	△	△	△ 15			
CONTAINER RECEIVED AT WANL									△			
DUMMY REACTOR SHIPMENT								OUT	△ 28	△ 6	BACK	
SHIPPING CONTAINER DESIGN REPORT COMP.											△ 30	
SHIPPING DEMONSTRATION REPORT COMP.											△ 30	

595403A

MAJOR EVENTS: SHIELD DEVELOPMENT

EVENT	1962			1963								
	O	N	D	J	F	M	A	M	J	J	A	S
KIWI B-5 SHIELD												
DESIGN REPORT ISSUED			30									
STRESS AND VIBRATION REPORT ISSUED						30						
NRX-A SHIELD												
LAYOUTS COMPLETE	15											
DESIGN RELEASE COMPLETE		15										
INSTRUMENTATION RELEASED					25							
SHIELD REPLACEMENT UNIT												
LAYOUT COMPLETE						19						
DESIGN RELEASE COMPLETE								24				
INSTRUMENTATION RELEASED										26		
DESIGN REPORT ISSUED										30

DOCUMENTATION SUMMARY

<u>ITEM</u>	<u>QUANTITY</u>
● REPORTS -----	35
● LAYOUTS -----	145
● AR'S -----	1000
● DEVIATION-TYPE VR'S -----	850
● WAIVER-TYPE VR'S -----	850
● NMDR'S -----	1200
● DRAWINGS ORIGINALLY RELEASED -----	2400
● DRAWINGS REVISED -----	2000

DESIGN CYCLE

595414A

MANPOWER EXPENDITURE

NE-1320

THERMAL AND FLUID FLOW DESIGN

PURPOSE: "To perform the thermal and fluid flow design required for the NERVA flight nuclear subsystem, to develop analytical techniques required to carry out the thermal and fluid flow design of the reactor and to specify experiments as required to verify the design of components prior to the reactor tests. To specify, monitor and analyze the thermal and fluid flow testing of NERVA reactors."

SUMMARY OF ACCOMPLISHMENTS

The thermal and fluid flow analysis required to permit preparation of all drawings and specifications for NRX-A Block I was completed. The development of analytical techniques and codes required to carry out the thermal and fluid flow design of the reactor and analytically predict the performance was completed. The specification of component tests required to verify the reactor component design and the test requests required for preparation of test specifications were prepared.

DESIGN ANALYSIS

The rated conditions for the NRX-A Block I reactors are shown in Figure 1. The conditions which determine this rating are specifically the reactor exit gas temperature which is established at 4090°R, and the hydrogen flow rate through the reactor which is established at 71.3 lb/sec. These conditions are those expected for the hot bleed engine.

The initial NRX-A reactor design was completed in November, 1962 and incorporated a single nozzle end core seal with coil springs and tile blocks as a core lateral support. In addition the shield design released was a two pass lithium hydride stainless steel unit for use with the hot bleed engine cycle. This shield was a modification of the KIWI B-5 shield which had been designed for the heated bleed engine cycle. Sufficient thermal design and fluid analyses were made to technically support this reactor design release.

As a result of the KIWI B-4A reactor test and further detailed analysis of the released design, the NERVA reactor program was re-oriented early in the contract year. The redirection of the design included redesign of the lateral seal system and replacing the lithium hydride-stainless steel shield with an aluminum block. The redesigned reactor resulting from this re-orientation was released in March of 1963. A preliminary thermal and fluid flow analysis of the redesigned multi-seal lateral support was completed in May 1963 and reported.

Analysis of this reactor design has been carried out for steady state design operating conditions, for part load operating conditions on both gaseous hydrogen feed and liquid hydrogen feed, and for startup, shutdown, and cooldown conditions. These are reported in detail in a Block I design report which was completed at the end of the contract year. The analysis indicate that the reactor will operate as designed within the degree of confidence one places on analytical techniques and assumptions and approximations used in the solution of the analytical problems. Further work is continuing on detailed analysis, and a component test program was initiated at WANL to verify the analysis and the analytical techniques used.

In the core area, analysis has been done to define the hot channel factors and the temperature gradients which will exist in fuel elements, tie rod units and support blocks. This information will be combined for each individual reactor core to determine the thermal capability of the unit. An indication of the results on hot channel factors is shown on Figure 2. This analysis has included the effect of the elimination of pins between the fuel elements and support blocks and has considered radial conduction within the fuel element itself. It has been learned from this study that the biggest unknown is the power distribution in the periphery of the reactor. Future effort will be directed at minimizing this unknown in order to improve the thermal capability of the core.

The methods and procedures for orificing the cores and the design and characteristics of the individual orifices has been determined. This method and information is presently being used in orificing the cold flow core.

Considerable effort was expended in the lateral support and seal area in determining the desired characteristics of the multi-seal design. A thorough analysis has been conducted at design and at test conditions planned for the reactor. This analysis is presently undergoing experimental verification through a series of planned component tests and through the use of the cold flow test unit.

The design of the lithium hydride-stainless steel shield was completely analyzed and the B-5 configuration was air-flow tested and reported. The aluminum replacement block for the shield has likewise been analyzed to determine that no detrimental conditions will occur during the test.

In the area of performance analysis, two transient computer programs have been further developed and used in predicting conditions. These programs are used to determine the conditions throughout the reactor during startup, shutdown, and cooldown phases of operation. These calculations help determine the operating limits of the reactor system. Figures 3 and 4 show the presently predicted operating limits curves for the NRX-A Block I reactors when utilizing liquid hydrogen feed and gaseous hydrogen feed respectively.

REACTOR AND COMPONENT TESTING

The requirements of component testing for the Block I reactors was specified and incorporated in the WANL component testing program. Analysis of the completed experiments were performed and reported on a continuing basis. The requirements for reactor testing in the form of operational considerations and in the form of data requirements were specified and incorporated in the NRX-A test plans and specifications.

SUMMARY:

A total of 15 milestones were required by the project, all of which were completed. Carry-over into Contract Year 1964 is considered to include the following tasks:

1. Test Predictions of NRX-A1 cold flow test unit
2. NRX-A1 variation request support

3. Component test analysis on the following tests:

- (a) impedance ring flow tests
- (b) reflector flow tests on H₂
- (c) statistical testing of fuel elements
- (d) support block flow tests

NE-I33I

REACTOR NUCLEAR DESIGN

PURPOSE: "To provide nuclear design information required for the reactor mechanical, thermal and control systems design; to provide required nuclear analysis to ensure safe handling, and ground support operations; to evaluate analytical techniques of computing reactor nuclear characteristics which provide means for readily assessing proposed design changes; to plan for and to analyze reactor and engine tests for nuclear design information."

SUMMARY ACCOMPLISHMENTS

The nuclear design information required to permit preparation of all drawings and specifications of NRX-A Block I was completed. The fuel element specifications have been developed for seven zones required to meet thermal design criteria on fission power distribution.

NUCLEAR DESIGN AND ANALYSIS

The nuclear characteristics of the NRX-A have been predicted using analytical techniques which have been carefully compared to experiment. Predictions have been supported by utilization of Los Alamos measurements on the geometrically similar KIWI B-4 ZEPO, KIVA, KIWI B-4A and B-4B experiments.

The fuel element specifications were determined to meet nominal operating reactivity requirements plus a margin to cover calculational uncertainty and effects of manufacturing tolerances on reactivity as well as to approach within $\pm 20\%$ a flat radial power distribution. The core consists of seven different fuel loading zones with the maximum being 417 milligrams/cc of fully enriched uranium. The distribution of this loading is shown on Figure 1. The accompanying power distribution for this fuel loading is shown on Figure 2. Calculations have been performed to determine the characteristics of the core with this fuel loading and a method was established for predicting the power distributions in each core channel as a result of the critical tests and the analysis in order to allow proper orificing. The reactivity characteristics of the NRX-A2 reactor are indicated on Figure 3.

A poison wire design consisting of B_4C powder in a stainless steel tube was developed for use in insuring subcriticality of the reactor during assembly and shipping. This system ensures the reactor will be shutdown even if it were accidentally flooded during shipment.

Studies were performed at the request of the government for the possibility of utilizing the KIWI B-6 oxide core in the NRX-A2 and for the case of possibly using a new oxide core for the NRX-A2.

NUCLEAR MEASUREMENTS AND EVALUATION OF ANALYTICAL TECHNIQUES

Nuclear test requests covering operation both at WANEF and NRDS of NRX-A were completed. A series of experiments were conducted in conjunction with Los Alamos at the KIWI B-4 ZEPO assembly. These experiments provided information on the reactivity difference between KIWI and NRX fuel elements, the reactivity worth of the tantalum carbide graphite shim pencils, and the reactivity worth of the boron carbide wires used to provide core shutdown.

A standard design method is being utilized in analyzing the NRX-A cores. This standard design method is the result of careful comparison with existing KIWI data.

SUMMARY:

A total of 9 milestones were listed for this project, all of which were completed. There is no carry-over from Contract Year 1963 on this project.

NE-I332

SHIELDING AND RADIATION ANALYSIS

PURPOSE: "To develop detailed radiation heating distribution throughout all components within the pressure vessel; to perform detailed nuclear analysis of the shield replacement unit; to determine the operational radiation environment of the engine and the heating rates and activation of engine components; to support studies of facilities, shielding and handling of components within facilities; to evaluate analytical methods of computing radiation fields and heating rates."

DESIGN AND ANALYSIS ACCOMPLISHMENTS

A lithium hydride-stainless steel shield which had been designed for the B-5 reactor was incorporated in the NRX-A units at the beginning of this contract year. In March, 1963 it was decided to remove the lithium hydride-stainless steel shield and replace it with an aluminum block.

The shielding and radiation analysis required to permit preparation of all drawings and specifications for NRX-A Block I was completed. Analyses included detailed radiation heating distribution in all NRX-A components within the pressure vessel. A detailed analysis of the aluminum shield replacement unit for use in the NRX-A Block I was completed. The operational radiation environment around the engine with the flight type shield was determined. Activation of all NRX-A components within the pressure vessel was calculated. Analytical methods used in radiation analysis were improved and evaluated by comparison with available experimental data. Part of the work covered

during this period was an evaluation of the shield design criteria.

As an example of one type of information which is generated on this project, Figures 1 and 2 are presented showing the gamma and neutron radiation environment around the NRX-A when on the test car at Nevada. This type of information is used for designing the reactor components and for designing the necessary test equipment.

KIWI B-5 SHIELD TESTING

A detailed description of the KIWI B-5 shield and its radiation design and analysis was presented.

Measurements of the gamma ray and neutron doses and integrated neutron fluxes were made with and without the Westinghouse B-5 shield on a critical assembly of the KIWI B-4 core at Los Alamos. Generally good agreement was obtained between calculated and experimental data.

SUMMARY:

A total of 4 milestones were listed for this project, all of which were completed. There is no carry-over from Contract Year 1963 on this project.

NE-1333

REACTOR CRITICAL TESTS

PURPOSE: "To perform nuclear testing of NRX-A series reactors to determine reactivity and power distribution. If measured reactivity does not meet design objectives, test the shimmed reactor until required reactivity is obtained."

ACCOMPLISHMENTS

A facility was completed and the necessary experimental equipment purchased and installed in order to carry-out NRX critical tests. In preparation for testing extensive

effort was applied to producing the necessary reports and documentation to operate such a critical facility. These documents consist of the following:

1. Safety Report
2. Health Physics Manual
3. Operating Limits
4. Operations Manual
5. Experimental Plan

The design of the experiment to be performed on the NRX-A reactors was carried out under this project. This developed the so-called PAX experiments to be performed.

In addition effort was expended on designing flexible critical experiments in support of the NRX program in order to allow the procurement of long lead items. This work was accomplished under an amendment to the basic contract.

Figures 1, 2, and 3 show the critical facility, the test stand and the control console. Since a detailed inspection of the facility is part of this review, I will not dwell on the details of the facility.

SUMMARY:

A total of 5 milestones were listed for this project. The one milestone calling for completion of the NRX-A2 PAX experiment was not accomplished. Carry-over from Contract Year 1963 is considered to consist of the effort required to perform the PAX experiment on NRX-A2.

NE-2500

RIFT CONTRACT SUPPORT

PURPOSE: "To provide, in cooperation with Aerojet-General, technical consultation to the RIFT vehicle contractor to ensure compatibility of the engine and vehicle; to provide information on the nuclear environment

and operational requirements of the nuclear subsystem."

ACCOMPLISHMENTS

The RIFT contract support provided, in cooperation with Aerojet-General, technical consultation to the RIFT vehicle contractor to ensure compatibility of the engine and vehicle, and information on the nuclear environment of the nuclear subsystem.

Factors affecting the radiation shield design criteria for a NERVA vehicle were reviewed and published. Figure 1 illustrates the WANL concept of the interrelationship of the shield design criteria factors with high reliability as the main criteria. Results of this review showed that the two quantities which must be determined in order to set the shield requirements are: 1) the maximum allowable liquid hydrogen propellant temperature rise, and 2) the engine and payload radiation environment restrictions.

Meetings to discuss tank interface problems and to provide information concerning the radiation environment around the nuclear subsystem were attended and participated in by WANL representatives as follows:

Tank Interface Meeting at AGC-REON, November 19-20, 1962

RIFT Support Coordination Meeting at MSFC, Huntsville, Ala., Dec 3, 1962

RIFT Support Coordination Meeting at WANL, Pittsburgh, Pa., July 1-2, 1963

Meeting to discuss Attitude Control System of RIFT Vehicle at SNPO-C,
July 9, 1963

RIFT Nuclear Engineering Report at Lockheed, Marietta, Ga., Sept. 10, 1963

SUMMARY:

There were no milestones listed for this project and there is considered to be no carry-over on this project.

STATION	DESCRIPTION	W - $\frac{\text{lb}}{\text{sec}}$	P - psia	T - $^{\circ}\text{R}$	ENERGY INPUT $\frac{\text{BTU}}{\text{SEC}}$
1	Dewar Conditions	71.3	60	44	
2	Pump Outlet	71.3	959	53	
3	Nozzle Inlet	71.3	949	53	
4a	Nozzle Tubes Outlet	69.3	750	165	
4b	Flange Coolant Outlet	2.0	750	55	
5	Reflector Inlet	71.3	750	162	
6	Reflector Outlet	60.1	706	241	1.74×10^4
7	First Pass Shield Outlet	60.1	699	239	4.8×10^2
8	P.V. Annulus Outlet	10.7	749	220	2.67×10^3
9	Second Pass Shield Inlet	70.8	699	238	
10	Second Pass Shield Outlet	70.8	689	242	8.5×10^2
11	Support Plate Outlet	70.8	689	248	2.0×10^3
12	Fuel Outlet	66.53	560	4277	1.02×10^6
13	Tie Rod Outlet	4.27	560	800	8.6×10^3
14	Reflector Seal Leakage	0.5	560	1200	
15	Nozzle Chamber	71.3	560	4090	

NRX-A CYCLE DESIGN POINT CONDITIONS

MOST PROBABLE NUMBER OF CHANNELS ABOVE DESIGN TEMPERATURES

		<u>NUMBER OF CHANNELS ABOVE</u>		
		4600 °R	4960 °R	5500 °R
25% Power				
	With Radial Conduction	0	0	0
50% Power				
	With Radial Conduction	220	7	0
75% Power				
	With Radial Conduction	4000	680	6
100% Power				
	With Radial Conduction	3000	785	20

CURVE NO.
595287

REACTOR STEADY STATE OPERATING MAP FOR COLD (90 °R)
HYDROGEN FEED

CURVE NO.
595286

REACTOR STEADY STATE OPERATING MAP FOR
LIQUID HYDROGEN FEED

CURVE NO.
562438

NRX-A FUEL LOADING DISTRIBUTION

CURVE NO.
595373

NRX-A FISSION DENSITY vs. RADIUS FROM R-θ CURVE, T=1700°K
DRUM ANGLE = 130° NORMALIZED TO 1.00 CORE AVERAGE

REACTIVITY TABLE

OPERATIONAL REQUIREMENTS

	<u>5 MIN</u>	<u>20 MIN</u>
NET COLD TO OPERATING REACTIVITY CHANGE	-\$0.73	
REACTIVITY LOSS DURING OPERATION	-\$0.30	-\$1.5
MINIMUM REQUIRED REACTIVITY	\$1.03	\$2.23
MINIMUM REQUIRED REACTIVITY WITH STATISTICAL UNCERTAINTY	\$1.43	\$2.86
ESTIMATED CONTROL WORTH	\$7.79	

NRX-A2
REACTIVITY

{	ESTIMATED REACTIVITY	-	ANALYTICAL	$\$4.8 \pm 1.63$
	ESTIMATED REACTIVITY	-	EXPERIMENTAL	$\$4.0 \pm 1.2$

SHIM CAPABILITY

POSITIVE	\$1.64
NEGATIVE	\$11.4

GAMMA DOSE RATE IN $\frac{\text{RADS (CARBON)/HR.}}{\text{FISSION/SECOND}}$

AT 1120 MW (THERMAL)
THE
FISSION RATE = 3.6×10^{19} FISSIONS/SECOND
AT
20 MINUTES

CURVE NO.
595263

GAMMA ISO-DOSE RATE CONTOURS OUTSIDE THE NRX-A PRESSURE
VESSEL SHOWING PERTURBATION BY THE PRIVY ROOF SHIELD

AT 1120 MW (THERMAL)
THE
FISSION RATE = 3.6×10^{19} FISSIONS/SECOND
AT
20 MINUTES

(INTEGRAL FAST NEUTRON FLUX, $\frac{\text{NEUTRONS/CM}^2 - \text{SEC}}{\text{FISSION/SECOND}}$, FOR
NEUTRON ENERGIES GREATER THAN 1 MEV)

CURVE NO.
595264

FAST NEUTRON ISO-FLUX CONTOURS OUTSIDE THE NRX-A PRESSURE
VESSEL SHOWING PERTURBATION BY THE PRIVY ROOF SHIELD

View of WANEF Critical Facility (Included is Westinghouse Atomic Energy Dept. Facility)

Control Room

CURVE NO.
564431

INTERRELATIONSHIP OF FACTORS AFFECTING SHIELD
DESIGN CRITERIA - FIGURE 1

NE-1334

REACTOR PHYSICS AND MATHEMATICS

PURPOSE: "To support the design of the NERVA Nuclear Subsystem by development of analytical techniques required for calculation of nuclear properties in both steady state and transient conditions; to provide the basic nuclear data required for reactor analysis; to develop new digital computer programs and to adapt and modify existing programs for calculation of reactor properties."

GUIDANCE AND SUPPORT FOR NUCLEAR DESIGN ACTIVITIES:

The basis for the nuclear design standard design method was formulated just prior to the start of CY 1963. This report was followed by a series of reports directed to Reactor Analysis elaborating the above scheme. At the same time people in the project worked closely with the Nuclear Design, Shield Design and Nuclear Analysis groups evaluating and comparing calculational methods.

NEUTRON SPECTRA:

In this area, we provided methods and techniques for calculation of the neutron spectrum distribution in the thermal and epi-thermal range to improve activity calculations as well as power distribution calculations at interfaces and discontinuities between fuel zones.

There has been a continuing effort this CY toward the development of improved techniques for calculating spectrum averaged cross sections both in the thermal and epi-thermal neutron energy range. The technique for the thermal range is embodied in the computer programs TNS 1 and TNS 2, described in TME-263. At the same time, a study was made of the theory of slow neutron scattering in crystals. The SUMMIT computer program for calculating neutron spectra in crystals was made operational.

WANL-TME-573

This program was used to calculate neutron spectra using various scattering to absorption ratios. Part of this work was reported in TME-282. During the past year, we have attempted to make the GAM computer program for calculating neutron spectra in the epi-thermal range operational. A number of alterations have been made in this program to eliminate certain defects found by making comparisons between GAM and MUFT results. Two further alterations to the GAM program, the treatment of resonance scattering, were begun during this year but are not yet completed.

CROSS SECTIONS:

Our work program calls for us to collect and evaluate nuclear cross section data for nuclear design; apply nuclear models to provide information in regions in which experimental data is not available; and prepare appropriately averaged cross section data for use in nuclear design calculations.

During the past CY we started the compilation of a new cross section library. This library is based on primary sources, that is, in the experimental measurements as reported in the archive journals or upon authoritative and documented compilations such as the Brookhaven National Laboratory reports, BNL 325 and BNL 400. A computer program has been written to compute elastic scattering matrices in the laboratory system from differential cross sections in angle and energy in the center of mass system. Internal reports written during the year specifically on cross sections are: U^{235} cross sections, preparation of inelastic cross sections, effective Boron cross sections, Beryllium cross sections, calculations of resonance integrals of U^{235} and U^{238} including Doppler Broadening. The negative doppler coefficient we found for U^{235} has recently been partially confirmed. A report on the effect of temperature dependence of Beryllium scattering cross sections at low energies, TME-152, was issued.

NEUTRON DISTRIBUTIONS:

Here we provide improved multi-dimensional transport methods for calculating reactivity and power densities in the reactor, reflector, control vanes and shield. We also improve and modify existing transport methods and compare results.

Since we were aware that the Los Alamos Scientific Laboratory was well along toward the development of two-dimensional S_n transport program at the beginning of CY '63, we did not attempt to duplicate their work. Instead we continued with the development of a transport program based on the Conditional Monte Carlo technique. During CY '63 this program was completed and a report, TNR-106, describing the program was issued. A revision of the program as originally planned was also described in the latter report. The revision was completed in CY '63. Results using this program for a reactor without control vanes have been compared with results using the Los Alamos TDC transport program. The agreement is good. We were not able to make comparisons with the S_n code for a reactor with control vanes. However, we have compared results from the Conditional Monte Carlo program with results using the CURE diffusion program for a reactor with 12 control vanes, shown on figures 1 and 2. A report showing the latter results is in preparation.

Since these last slides were made about a week ago, we have received an apparently debugged version of the LASL-DDK code which will do r, θ problems in the S_n approximation. The first comparisons we have made with the results shown in the slides clearly show that the differences between the Monte Carlo transport theory and diffusion theory are in qualitative agreement with the differences between DDK and diffusion theory. However, one must still represent a three-dimensional problem with an artificial leakage term in a DDK - r, θ problem and for multigroup transport there is no single correct and unambiguous way to do this. There is not time to go into this in detail here, but this work is continuing.

KINETICS:

We develop space-time dependent techniques for determination of the spatial power distribution in the reactor as a function of time during transient conditions such as start-up.

An original method based on the variational principle has been developed for finding the spatial power distribution in the reactor during transient conditions. A computer program using this technique to solve time-dependent neutron diffusion equations with two space dimension has been completed. This work was reported in TNR-133. During the next CY heat transfer and fluid flow feedback effects will be included in the program. Figures 3 and 4 illustrate some results of this effort.

COMPUTER PROGRAMMING:

Our TAD calls for us to: "Provide computer programs in support of new calculational methods. Assist and support the development of major computer programs. Guide and assist in application of existing programs and make required modifications and corrections."

A computer program for correlation and spectrum analysis of test data has been completed. This program has been used to analyze measurements on the B4-A cold flow test motion picture film. A theoretical flow induced vibration model has been formulated and a program written for the digital computer. Figure 5 illustrates this effort.

Several computer programs have been written at the request of other project activities. Part of this work has been reported by the group making the request. Examples of this work are: Parts Lists program for Program Control, Astrofuel Information Retrieval, a program for calculation of the transfer function of the control system requested by Control Analysis, several programs for calculation of the pressure and velocity distribution of flow in a network requested by Reactor Analysis and Advanced Systems. We have also made operational several computer programs obtained from

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WANL-TME-573

outside. Examples are: the Los Alamos DTK transport program, the LASL MESHER program, the GAM slowing down program, the HAFEVER program for calculation of excitation cross sections, ZUT and TUZ programs for calculation of resonance integrals, the DYANA program for vibration analysis of complicated systems, a two-dimensional thermal stress program obtained from Batelle Memorial Institute, the "Random Vibration Program" obtained from the Marshall Space Flight Center, Huntsville, Alabama.

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FIG 1 COMPARISON OF THE THERMAL FLUX CALCULATED BY CONDITIONAL
MONTE CARLO AND BY THE LASL TDC TRANSPORT PROGRAM
(TDC IN S_4 , r , z MODE)

COMPARISON OF THE THERMAL FLUX CALCULATED BY CONDITIONAL
MONTE CARLO (r , θ , z) AND BY THE CURE (r , θ) DIFFUSION PROGRAM

- TITLE: THERMAL FLUX AS FUNCTION OF TIME

595838B

595712B
CHANGE IN POWER RISE DUE TO CHANGE IN LOCATION
OF REACTIVITY INSERTION

AMPLITUDE OF THE 22.2 CYCLE PER SECOND COMPONENT VS RADIAL POSITION

NE-1352

REACTOR COMPONENT FLOW AND CORROSION TESTS

PURPOSE: "To assess the flow and corrosion capabilities of reactor components under various environmental conditions; to measure the corrosion in fuel elements at operating pressures and temperature for fuel element evaluation and development; to develop methods for rapid and accurate measurements of fluid flow; and measure fluid flow behavior in reactor fuel assemblies and components under environmental conditions.

Originally, the scope of the work envisioned under Project 1352 was limited to corrosion studies on fuel elements and minor flow tests with either ambient or low temperature hydrogen. After the failure of the KIWI B-4 reactor, this portion of the program was greatly expanded and encompassed a much wider range of test endeavors.

INTRODUCTION:

The scope of the tests under the Component Test Program now embraced not only the previously planned work on the corrosion behavior on hydrogen in coated fuel elements, but also included experiments on flow induced vibrations of the type suspected of being the damaging mechanism for the KIWI B-4, a greatly increased number of flow pressure drop behavior tests of many more reactor components, and a greatly expanded subassembly type of flow testing such as reactor reflector segments and complete control drums. Many difficulties were encountered in redirecting the program from a relatively small, clearly defined one to a program which at its inception coincided with a marked change in the design of the NRX-A1 reactor. The program as well as the design of the various tests and test rigs was done under considerable pressure both in time and in scope. In many instances the reactor design changes required that preliminary planning of tests be accomplished before the configuration of the reactor component had been finalized. Such a parallel effort of design and experimental approaches in many cases required that the initial plan had to be modified repeatedly in order that a good logic could be maintained between test program and the final design configuration. The time scale was such that the first several

months of the contract year were devoted to a relatively slow build-up in the initial plan for only corrosion testing and small scale flow tests. The overall plan was greatly accelerated in approximately January, and a great effort was centered in devising, planning, procuring, assembling, and checking out the many rigs and experimental devices now used in the Component Test Program. During the last several months the data started to become available from the variety of the tests being performed as part of the Component Test Program.

For discussion purposes, the component flow tests grouped themselves into approximately four (4) categories: corrosion work, major and minor flow tests, and experiments involving flow induced vibrations.

CORROSION TESTS

Corrosion of any core material, if excessive, will seriously jeopardize reactor performance from the viewpoint of fuel material losses. These in turn lead to neutronic deficiencies as well as attendant disruptions of channel heat transfer and flow balances. To investigate the material loss of the fuel elements, a hydrogen corrosion facility was built. The primary purpose of this facility was to simulate environmental conditions close to that encountered in the actual reactor operation. The facility consisted of a flow loop in which a fuel element was mounted as a continual part of the hydrogen flow passage. The element was physically located in a cylindrical water cooled horizontal pressure vessel, containing a helium atmosphere on the outside of the test vessel. The vessel had nine viewports which permitted observation and measurement of the element surface temperature. Gaseous hydrogen flowed through the holes in the element, then through a heat exchanger, a pressure reducing valve, a filter, and was finally vented to the atmosphere. This equipment is shown on Figure 1. One hundred corrosion tests were performed during the contract year 1963. These included system checkout, fuel element contact development work and fuel element tests in support of the development program and the investigation of heat transfer and pressure drop relations. Figure 2 shows the relation between weight loss and temperature obtained from a selected number of five-minute tests with coated, enriched fuel elements.

The testing program to date has resulted in a continual improvement of the test equipment and test techniques so that consistent and reproducible information can be obtained. The facility is now capable of performing quality control corrosion tests as well as heat transfer and fluid flow behavior tests in addition to fuel element coating development tests. Milestones completed for 1963 were the completion of the test rig procurement, the start of corrosion tests and the issuance of a pre-production testing report. Since production elements were not available as planned, statistical testing of these did not start, and will be carried over the CY 1964.

The fuel elements are retained in the core by the support blocks. The purpose of these support blocks is to hold the fuel elements in the core as part of the complex structural system. In order to determine the interrelation of the tie rod support system with the support blocks, the structural integrity of these blocks must be assured under operating conditions. In the first tests, a plexiglas model, which essentially duplicated the geometry in the immediate vicinity of the tie rod support cone was used to obtain preliminary information on the flow pattern at the exit of the tie rod channel. Hot and cold water was used as the fluid medium. The tests were designed to determine whether the tie rod support cone is adequately cooled by the center channel flow and to determine the optimum longitudinal position of the tie rod support cone. The test results indicated that the cone tip would be more adequately cooled if recessed approximately a quarter inch below the face of the support block rather than the original design condition of one hundredth inch recession. These results of course, must still be checked in high temperature tests to be conducted in a corrosion furnace. Since the availability of test specimens was extremely limited, these tests were not completed and will be carried over into Contract Year 1964.

To determine the total integrity of the support system block and fuel elements, a single cluster high temperature test under environmental conditions, very closely simulating the reactor, was designed. A fuel cluster will be suspended by standard reactor

hardware in the center of a vertically cooled high pressure furnace. Inside the furnace, the electrodes are connected to the cluster at the top and bottom by means of suitable graphite chucks. The cluster simulates the NRX-A operating conditions by electric resistance heating and pressurized internal hydrogen flow. Hence the entire assembly can be evaluated at hot reactor conditions. Figure 3 shows the general arrangement of the test. The heat exchanger and filter which follow the furnace are supplementary to the test and permit the hydrogen to be filtered prior to burning in the atmosphere. Work on this test was suspended for a time as a result of a program review with the customer in order that complete investigations of the various design methods necessary to electrically insulate certain component parts could be verified. Once these design difficulties were suitably solved, the furnace, heat exchanger, and elbow transition pieces would be designed and are now in the process of fabrication. The furnace and heat exchanger are to be delivered the first week of December 1963 and the elbow a week or so later. While a test specimen could have been procured during the contract year, that is, one cluster might have been procurable, the delay resulting from the extremely careful design studies of the insulation precluded the fabrication and delivery of the furnace, heat exchanger, and transition piece during the Contract Year 1964.

In summary then, the high temperature work accomplished one hundred fuel elements tested for process development and flow information during this contract year. Support block models were studied to determine the optimum method of cooling the support block and a single cluster furnace was designed and procurement was initiated.

MAJOR & MINOR FLOW TESTS

The minor flow tests embraced many various small parts of the reactor. The purpose of these tests was to determine the flow rate-pressure drop characteristics of various components in order that the required flow matching could be obtained. Models of the components were designed, fabricated and tested. The components of the tie rod

hardware included the tie rod hanger, centering bushing, stainless steel liner and cone support. A number of these components are shown in Figure 4. Hydrogen gas at both ambient and low temperatures (to -320°F) was used at pressure levels of 200 to 600 lbs/sq. in. The various geometries were reported and the data was presented in the form of a pressure loss coefficient as the dependent variable and flow Reynolds number usually based on the minimum area restriction as the independent variable. Another component of this type was the fuel element orifice which controls the flow through an individual flow channel in the reactor. Fuel element orifices of all twelve sizes to be used in the NRX-A reactor were examined, flow tested and calibrated. These tests led to the determination of master orifices, each selected from a given size range, to be used as the secondary standard for the production testing of the orifices to be used in the reactor assembly. In addition to these components the impedance ring limiting the flow between the inner and outer reflector was also tested. This component required the use of a section of a total circumference, and these tests were made on air. All of the milestones were completed on the minor component tests and the information was available for NRX-A analysis where required.

The lateral support and seal system consists of a complex flow system combining radial flow through filler strips between the core and seal annulus, axial flow pass seal segments and leakage past the spring pins. This test, as well as the design, evolve progressively as the program was redirected from the initial design release of November to the April configuration. The effectiveness of this system is being established by the use of a series of suitable models which will determine the individual flow characteristics of each seal as well as the composite effects. The following series of models have been designed, procured, have been tested, or are about to be tested.

1. Low pressure single seal test model. This was the first and simplest of the models using the design concept but not actual dimensions. This was tested on air.

2. High pressure single seal model. This model simulates a 14" section of the actual lateral support segment. It is being tested isothermally at high pressure levels with hydrogen.
3. The multiple seal isothermal model is essentially a duplicate of the single seal repeated eighteen times thereby simulating one-ninth of the inner reflector. Testing will be made isothermally at high pressures.
4. The heated graphite single seal model. This model, while similar in geometry to the single seal model will permit the use of high temperature hydrogen to determine seal efficiency and corrosion results. Internal heating to simulate conduction from the core to the filler strips is also incorporated.
5. Nozzle interface seal models. These models will determine the adequacy of the nozzle interface seal. One is a short flat segment and the other a 10" diameter scale model of the full size.
6. Full scale circular model. This model will be a heated full scale circular model using six seals in series.

The milestone for the initiation of testing on these models was met one month before the due date. The evolution and change in the plans of this test made the milestone of completing the procurement of a model meaningless.

A full size partial section of the beryllium reflector was made to simulate the flow passages through this component. The hydrogen is introduced through a series of holes in the lower support ring, flows through the reflector and is exhausted through a similar set of holes in the upper support ring. The model was constructed to determine the flow rate pressure drop characteristics of this component. This was necessary in order that the proper flow balance of several parallel paths supplying the core with hydrogen could be established. The test fixture is shown in Figure 5. Initial tests were conducted using air at ambient temperature and pressure of 120 lbs/sq. in. gauge. Final tests were made with hydrogen. In order to accomplish these tests, two flow loops had to be fabricated

and installed. The first was a flow loop for air flow testing this component. The second was a flow loop which permitted cold hydrogen at high pressures to be supplied to the model. In the hydrogen loop temperatures as low as -320°F and pressures as high as 700 lb/sq. in. absolute were obtained. Good agreement between air and hydrogen has been obtained for well rounded restrictions and long flow passages, but the agreement is not so good for sharp-edged short tubes representing the geometry of the support ring. Tests of the flow drum were divided into two parts. The first consisted of evaluating the design and off design flow coefficients for the orifices formed by the drive shaft mechanism to the control drum. The second part consisted of flow distribution studies and flow pressure drop measurements of an entire drum. For the drive shaft restriction studies, the initial configuration was tested over limited flow conditions concurrent with the major redesign previously mentioned. Consequently, a new set of models were designed, fabricated and instrumented. The testing of this new set of models, six in number, is now being completed. Final testing with hydrogen gas at cryogenic temperatures is contemplated in order to verify the design. A complete control drum has been obtained and the assembly of its components in the housing is presently occurring. All of the milestones have been met on the reflector segment but the redesign of the control drum required additional air testing with the redesign component. Hence, hydrogen testing of some components is yet to come.

FLOW STABILITY TESTS

After the failure of the KIWI B-4, a program was initiated to study flow induced vibrations, which might also occur from hydrogen coolant flow phenomena in the NRX-A reactor. Specifically, the program was designed to evaluate the effectiveness of the NRX-A lateral support and seal system, to determine the flow stability of the reactor core as a function of the flow induced vibration, and to verify flow and pressure analyses. The test plan evolved to carry out this program in two forms. In the first part, a seven cluster reactor prototype core is evaluated under dynamic and static flow conditions. In the second part, the investigation is applied to a full scale plugged core reactor. The seven cluster

prototype reactor core uses standard reactor clusters complete with filler strips and other components necessary to simulate the reactor assembly, and include a modified lateral support and seal system, inner reflector support plate and outer support. An assembly of this test device is shown on Figure 6. Different core geometries and designs will be studied and evaluated during the coming contract year. Immediate plans include additional NRX-A geometry tests, a soft support system simulating the KIWI B-4 as well as plugged core and solid core tests. Initial tests on nitrogen and hydrogen were made during September at flow rates of hydrogen up to 0.6 lb/sec or 35% of the design flow (approximately that expected in the cold flow test of NRX-A1). No indications of flow instability were found. Plugged core tests are for the purpose of obtaining information on a full size core. By plugging the major flow passages through each element, it is possible to study the flow vibration stability in a full size reactor with small flows. These small flows representing the leakage flow through the core and through the lateral support system. This full scale, plugged core, test reactor will be equipped with prototype reactor hardware complete with seal pressurization system which will allow the pressure in the lateral support and seal system to be varied externally without disassembly. The design of the test unit as well as the modification required in the support loop has been completed and assembly on this reactor has been started. Both the seven cluster rig and the full size plugged core tests presently are ahead of schedule.

In summary then, all of the tests associated with the Component Test Program have proceeded at a rapid rate, met the required dates unless limited by available hardware and have produced usable design information.

NE-1343

TRANSIENT FLOW ANALYSIS AND TEST

PURPOSE: "To obtain basic data required for the solution of reactor design and control problems under conditions of mixed (two-phase) liquid propellant flow; to obtain basic data in support of analytical models which describe the pulsating flow during transient operations of the reactor. "

INTRODUCTION:

The aim of Project 1343 was to obtain basic information on two-phase flow phenomenon and associated heat transfer. This project consisted of setting up a cryogenic test loop to be used with liquid hydrogen, obtained cooldown behavior of various simple geometries, from the test results obtained on these simple geometries deduce heat transfer and pressure drop coefficients, and finally develop mathematical models which could be used to check the phenomenon involved in these cooldown experiments. A picture of the vacuum insulated cold flow loop is shown on Figure 7. The summary of results obtained from these experiments consist of

1. Tests made with gaseous hydrogen closely check the data produced by previous workers. The temperatures of these experiments were considerably lower than any previous work. They ranged from approximately 145°R to approximately 250°R .
2. The experimental results on heat transfer correlate well with some work done previously as shown on Figure 8.
3. This data, when used with the computer TRAF code for two-phase behavior, is not good, but for single phase gas, the agreement between experimental and calculated results are excellent.
4. A statistical approach of separating liquid and gaseous heat transfer has promise, but the prediction of liquid probability as a function of time is yet not clearly defined.

A series of tests were made using single and multiple passage test sections made of copper, aluminum, stainless steel and in the future, beryllium. These test sections simulate a single flow passage in the reflector during cooldown conditions. Successful heated test runs have been performed which essentially duplicated the conditions in the reflector resulting from nuclear heating. All of these tests are based on a short time interval occurring during reactor start-up. The liquid hydrogen fuel loop used to carry out these experiments consisted of a pressurized dewar transfer system and the necessary valves required to route liquid hydrogen to the test section. Temperatures and flow rates, respectively, were measured with thermocouples and calibrated orifices. Transient two-phase flow and heat transfer data has been analyzed on the basis of postulated two-phase flow models and heat transfer mechanisms established from experimental data. The initial two-phase heat transfer correlation, presented on Figure 8, in which the overall heat transfer coefficient is considered a unique function of the wall to wall temperature difference is shown. More sophisticated approaches in which attempts are made to statistically predict the liquid and gas probability of occurrence at any one time are now being investigated. This type of approach shows greater promise to adequately explain the two-phase phenomena. In order to calibrate equipment and check the measurements, cooldown of test sections was first obtained from gaseous hydrogen. The correlation with the modified Dittus Boelter equation was excellent. Transient response of the test section, when predicted by the computer calculated cooldown (TRAFT code) was excellent. The agreement, however, was very poor and showed large discrepancies when the same approach was attempted with two-phase treatment of the data. Experimentally, little or no difficulties were encountered in measuring the cooldown with copper, aluminum or stainless steel test sections and no difficulty is expected when the test section is made of beryllium. However, when attempts were made to introduce internal resistance simulating nuclear heating in the reflector with the use of an aluminum test section, the results were very unsuccessful. The

major difficulty was due to the extremely low resistance of the aluminum test section. When the material was changed to stainless steel, the first successful internally generated cooldown data was obtained. On the five milestones listed for this project, four were met, the exception being the delivery of the beryllium test section which was delayed by the supplier.

High Temperature Test Cell

562377A

Weight Loss Spectrum for Coated, Enriched Elements

Figure 86 - General Arrangement

FLOW INDUCED VIBRATION TEST RIG

TRANSIENT TWO PHASE STUDIES

COMPARISON OF TWO-PHASE HEAT TRANSFER COEFFICIENTS FROM VARIOUS SOURCES FOR HYDROGEN

NE 1313

MECHANICAL ANALYSIS AND TEST

PURPOSE: "To determine through design analysis and non-nuclear testing the capability of the reactor to withstand required environmental and operating conditions. The non-nuclear testing consists of a systematic plan supplementing the design analysis of static, vibration, and shock tests of components, assemblies and subassemblies. To evaluate core designs and predict safety margins; to assist in finalizing dimensions, tolerances, and quality control limitations in areas where complex configurations do not allow reliable analysis."

INTRODUCTION:

The past Contract Year, 1963, has been a year of rapid change meeting the needs of a growing program. The concepts which were our guideposts have themselves been overtaken and our sights raised to encompass far broader responsibilities than were envisaged even as short a period as one year ago. This has entailed continuous planning which has been, and still is, as important a part of the operation as the actual physical accomplishments.

Our planning finally took definite shape in the form of the program published as WANL-TNR-095. This document met an urgent need and provided a basis from which we could define and mold a requisite organization. As with any fixed position in a growing area, there is still a constant need for revision and change as the work outdistances the written planning. As a result, revisions have been made and more are planned. These will be reflected in future reissues of this and similar documents.

At the start of the 1963 Contract Year, Engineering Mechanics was composed of approximately twelve engineers and twenty technicians. As of October 1, starting Contract Year 1964, the department included forty-three engineers and fifty-eight technicians, representing a growth of approximately 300%. Despite the problem of integrating such a large group of people while starting a new comprehensive work program during such a short

period, the department has provided the necessary preliminary test results reasonably on schedule.

Corresponding to the growth in manpower, there has been a similar growth in departmental equipment and facilities. The laboratory is now equipped to perform almost any mechanical test on a component scale.

As in any such effort, there have been certain frustrations and unavoidable delays which upset the planned schedules to some extent. Not the least of these has been the difficulty of obtaining test specimens due largely to new "state of the art" manufacturing problems. As a result, in some instances it has not been possible to meet the designated schedules.

The basic program was designed to review the need for and draw up a more comprehensive plan for appropriate and necessary tests, as well as inaugurate a modest laboratory test and associated analysis effort. The laboratory effort included specific test requests from the design groups which were handled on an individual basis. There was little possibility of integrating these test requirements into a comprehensive program. Also, the program proved to be more inclined to the analytical approach toward assisting the design effort in the evaluation of developmental hardware. As the tempo of the overall program increased, and designs became more well defined, the need for a broader base testing program encompassing both qualification testing and experimental component development became apparent. A program much broader in scope was prepared for the latter half of the Contract Year.

It was now evident that it was necessary to develop a broad perspective regarding the mechanical behavior of all graphite components such as fuel elements, support blocks, and lateral support details. It was also evident that the areas of concern suggested by the KIWI full-scale tests, as well as the preliminary component test series and analysis, required broad qualification test coverage under all environmental extremes.

At the end of Contract Year 1963, most of the tests were well underway, some nearing completion. Preliminary test series on core components: fuel elements, support

blocks and tie rods have been completed as scheduled. The seven-cluster assembly, partial core assembly, lateral support components, and outer reflector component tests are either complete, or proceeding approximately as scheduled. The first series of reactor assembly vibration and shock tests was started in September. The pyro-tile lateral support assembly, control drum assembly, graphite barrel and the support system assembly mechanical tests are in the process of being assembled, or checked out.

Following are some of the highlights of the year's activities.

(A-1) (A-3) - FUEL ELEMENT TESTS:

These tests were designed to determine the physical characteristics of unfueled, uncoated fuel elements. The preliminary results obtained indicate that the strengths of the elements may substantially exceed manufacturer's reported data. For example, the dynamic modulus of elasticity of tested fuel elements was 2.5×10^6 psi compared to 1.9×10^6 psi reported by the manufacturer, while ultimate compression strength for these elements measured 20 to 50% greater than reported by the manufacturer. Fatigue strength tests were conducted on 60 specimens (figure 1) and internal damping characteristic tests were started. Over 60 tests were conducted on elements made to vibrate laterally within a given side clearance to determine empirical relationships between input acceleration, element side clearance, method of clamping and cycles to failure of single fuel element specimens (figure 2).

(A-5) - SUPPORT BLOCK AND PYROLYTIC PARTS MECHANICAL TESTS:

Axial static load tests were performed on regular interlocking support blocks and an average strength of 3900 lb was determined. This compares to a 500 lb design load requirement. Preliminary shear test strengths of 1500 lb were measured for support block interlocks.

(A-6) - TIE ROD TESTS:

Tensile tests of tie rods made of Inconel X-750 and Inconel X-718 materials at temperatures of -320°F , ambient, 900, 1150, and 1250°F were conducted. The data

obtained from the ambient and low temperature tests indicated that the strength of either material exceeds the 50,000 psi design requirement for the NRX-A1 Cold Flow Tests (figures 3 and 4).

Sufficient data was obtained from ambient and elevated temperature fatigue tests to construct modified Goodman diagrams and endurance curves (figures 5 and 6).

(A-9) - FUEL CLUSTER ASSEMBLY MECHANICAL TESTS:

Axial and lateral vibration tests were performed on a seven fuel element cluster assembly of production hardware. Inspection of a seven-cluster assembly after lateral vibration testing through frequency ranges of 10 cps to 500 cps at peak input acceleration levels of 1 g through 5 g revealed only slight fretting and chipping on some of the mating surfaces of the fuel elements and support blocks.

A test was also conducted on an assembly containing a center cluster of coated elements. A 3 g input peak acceleration at 30 cps for thirty minutes was followed by 58 cps for an additional thirty minutes. Visual examination of the individual clusters revealed no visible damage or fretting.

(B-1) - LATERAL SUPPORT SPRING MECHANICAL TESTS:

Initial development work proved graphite cloth to be unusable as a spring. Further tests showed that single and double helical steel springs retained their characteristics and dimensions over extended vibration periods.

(B-2) - LATERAL SUPPORT AND PYRO TILE SUBASSEMBLY TESTS:

Test pieces and fixtures have been received and tests are about to start. Some preliminary activities including investigations of various bonding cements for bonding the pyro tile to graphite have been partially completed.

(B-4) - GRAPHITE BARREL TEST:

Preliminary activities including test piece procurement and test fixture design and procurement were completed. Installation of the experimental facilities for hydrogen permeability studies and buckling studies has been started.

(B-5) - PARTIAL LENGTH CORE TESTS:

Vibration, static and shock loading tests were completed on models of a 9-inch reactor core section. The models tested were:

1. Solid graphite core with 3" x 3" graphite support blocks and NRX-A1 type helical springs with a 98.25 lb/in. spring rate.
2. Core of 9-inch lengths of KIWI-B4 type elements suspended in the NRX-A1 type lateral support system.
3. Core of KIWI-B4 type elements with KIWI-B4A lateral support with a 1.6 lb/in. spring rate.

The results of the vibration tests showed that the core made up of fuel elements bundled with NRX-A1 type lateral support springs, behaved similarly to the solid core. With a KIWI-B4A type lateral support system, no motion was transmitted to the core due to the low spring rate. It was found that the lateral support system was highly damped due to the sliding friction at the lateral support block-core interface. The equivalent system lateral support spring constant with the NRX-A1 type lateral support was determined to be 5280 lb/in. The force required to "break" static friction with the NRX-A1 lateral support was 615 lb/in. The coefficient of friction was 0.29.

Analog computer response curves derived from a simple mathematical model and the experimental response curves obtained from the tests showed good correlation (figure 7).

The results of the shock test series of experiments showed that:

1. The system is highly damped by coulomb friction.
2. A bundling pressure of 3.3 psi prevents relative motions between fuel elements.
3. The core will return to a dead zone 0.11 in. wide centered about its normal static deflection determined by $W/k = 0.104$ in.
4. The dynamic friction force necessary to overcome coulomb friction is 275 lb.

(C-2) - SUPPORT SYSTEM MECHANICAL TESTS

A preliminary test plan describing test objectives, loads, temperature environments, instrumentation and methods of data acquisition and analysis for the project has been completed. The test rigs have been received and assembled and the support system reactor hardware is now available for static load and thermal gradient tests. An IBM computer program has been written to provide automatic reduction of test data.

(D-1) - REFLECTOR SEGMENT TESTS

Compatibility tests on Ti-Ti, Ti-Al, Ti-Be, Al-Be metal combinations showed that galling occurred at the contact interface for all combinations when cycled between -320°F and $+500^{\circ}\text{F}$. A coating of 40 to 70 micron graphite particles suspended in sodium silicate applied to the Al-Ti and Be-Ti combinations prevented galling between these materials.

A series of compatibility tests to evaluate the galling characteristics of various nut and bolt material combinations were completed. The material combinations tested were 304 SS-Be, Ti-Be, Ti-Ti, Ti-(Ni-Span-C), (Ni-Span-C)-Ti. The bolt-nut combinations were preloaded to a bolt tensile stress of 60,000 psi and test cycled between -320°F and $+500^{\circ}\text{F}$ until galling occurred. The following observations were noted from these tests:

1. All uncoated combinations galled during assembly.
2. 0.0002 in. thick silver plate permitted seven test cycles before galling.
3. Tuftriding was effective in Ti bolts for seven cycles.
4. Coating threads proved more effective than coating nuts.

Test apparatus for determining deformation and relaxation of tie bolts during cycling from -320°F to $+500^{\circ}\text{F}$ was procured and assembled.

The resistance of the beryllium reflectors and control drums to brittle fracture under steady state and thermal gradient transient conditions is being tested. No specimen fractures have occurred to date with thermal gradients up to four times greater than predicted in planned reactor tests.

(D-3) - CONTROL DRUM ASSEMBLY EVALUATION:

All hardware except for the control drum itself, has been received and assembled. Preliminary test runs with a simulated drum have been made to check out the rig and instrumentation.

(G-1) - REACTOR ASSEMBLY MECHANICAL TESTS (Figure 8):

A series of core assembly vibration tests have been started. No relative motion between the core and inner reflector has been observed during low level vibration tests run to date. This substantiates the experimental results obtained from the B-5 test series which indicated relative motion would only occur at input levels greater than 1.6 g.

These first low level vibration tests have yielded the following information:

1. The stresses in the core filler strips and fuel elements were relatively low. Stresses induced by shipping environments will be insignificant.
2. No relative displacement between the inner core reflector has occurred with input levels below 1 g.
3. No separation of clusters has been observed.
4. Rocking modes of vibration of the core were observed during the low level tests; however, these were anticipated. This mode of vibration will not occur in the actual reactor assembly unless the coulomb friction force between the nozzle support ring and inner reflector is exceeded.

Analytical studies have been made in order to predict and understand the vibration characteristics of the G-1 reactor. The response to a given input will be non-linear due to heavy coulomb damping in the system. Lumped mass models have been developed and are being analyzed for each test configuration. These analyses determine input levels which can be safely applied to the reactor, the response to be expected, and the frequencies at which major responses will occur. It was found that an input of approximately 0.12 in. single peak displacement of 5 g magnitude is necessary to impact the core at 13 cps. A resonance appears on the mobility curve at 27 cps indicating a force sensitivity at that frequency. A rocking mode of the core can be expected due to tie rod restraint at

the dome end. The results from these studies when combined with the core assembly vibration test results should provide sufficient information to develop an accurate theoretical representation of the G-1 reactor assembly vibration system.

A shock analysis has also been performed on the NRX-A1 leaf spring design.

Digital techniques which could be used in solving the coulomb friction equation are being investigated and a solution to simple problems, where the friction force can be represented in equation form, has been found.

Good correlation between experimental tests and theoretical analysis have been observed thus far and accurate predictions of the dynamic behavior of the reactor may be possible.

NRX-B PLATE PLASTIC MODEL TEST:

High strain values measured in the base plate of a full-scale plastic model of the original tube and plate design indicated that stresses would be excessive. A new forged steel design was developed and a full-scale plastic model was tested to determine deflection and strength characteristics. Strain measurements indicate that the stress at highest operational load will occur at the center of the plate and will be about 50% of the A-286 metal strength.

FRICTION TESTS:

Specimens from seven types of graphite, two types each of aluminum and stainless steel, Inconel X, beryllium, titanium, and Ni-Span-C were tested in various combinations and environmental conditions to determine static and dynamic coefficients of friction.

STUDY OF MODES OF FRACTURE AND DEFORMATION:

A total of 45 specimens of ATJ, ZTA, and HLM graphite were tested at atmospheric, 200, 500, 750, and 1,000 psig pressures. The average tensile strength was found to be considerably greater than indicated in manufacturer's published data; however, it is uncertain whether hydrostatic pressure does in fact effect tensile strength.

NUCLEAR COMPONENTS TEST:

Post-irradiation examinations of teflon and poly-imide specimens irradiated in the GETR shuttle facility revealed extensive damage occurred to teflon, but poly-imide remained stable. These materials were candidates for use as retainer rings in the control drum ball bearings. Additional potential retainer ring material specimens including bronze, poly-imide, tungsten diselenide and molybdenum diselenide were prepared for irradiation tests in the Plumbrook Reactor.

GENERAL SERVICE TESTS:

In addition to the original requirements of the 1963 work program, many other tests have been performed in answer to specific work requests from other WANL departments. Some of these tests were conducted to assist the design and analysis groups in determining design criteria. A number of the tests were conducted for developmental and exploratory purposes to assist in the selection of candidate designs and verify the design analysis on various reactor parts. Because of the large number and random nature of these tests, no attempt has been made to report the results of these special tests in a categorized manner. Internal reports can be cited for most of the results and only brief comments are presented here.

THREADED HOLES IN BERYLLIUM

A test series was conducted to evaluate the load carrying capacity of threaded holes in beryllium. Similar holes are required in the NRX-A control drums and outer reflector, and due to the brittleness of this material, the tendency for crack propagation at these points was unknown. The results were as follows:

1. With a conventional tapped hole, a bolt tensile stress of 140,000 psi was required to fail the thread in the beryllium blocks.
2. If a bolt engagement of two diameters was used, and a counter bore was machined one diameter deep at the entrance of the

WANL-TME-573

threaded hole, tensile stresses greater than 220,000 psi could be consistently developed in the bolt.

These results showed that the probability of failure in beryllium components which originate from normal stress in threaded holes is small.

NRX-A CORE BUNDLING

This project was conducted to determine the lateral pressure required to reduce the gaps between fuel elements to practically zero. Generally, production fuel elements in a core bundle will have gaps due to tolerance build-ups, bows and twists.

The results of the test were used by Reactor Mechanical Design to derive the present criteria for lateral support in the core.

STATIC TENSILE TEST OF LASL DESIGN TIE RODS

Five non-instrumented LASL design 17-7 ph stainless steel tie rods were tested in straight and offset tension. Two were tested at room temperature and the remaining three at -320°F. At room temperature, the fracture load of 1800 lb was substantially the same for the offset test and the straight pull test.

For all cases, failure occurred at a thread root.

NRX-A CLUSTER LOCKING DEVICE

This test was performed to evaluate structurally several different designs for locking the tie rod ends to the core support plate. A limiting criteria for the test was that the torque requirements shall be small enough for easy remote disassembly after a reactor run.

The results indicated that either a two or three crimp design would be the optimum for meeting all requirements. This feature was incorporated into the present NRX-A1 configuration.

PULL-OUT FORCE REQUIRED TO REMOVE POISON WIRES SIMULTANEOUSLY
FROM THE CORE

This project was conducted to determine the total force required to remove the poison wires from the core and to observe any noticeable damage to the elements after pull. The results indicated an increase in force required as the distance from the reactor center increased. Also, the peripheral wires caused gouging of the element during the removal process. Presently, the wires are coated with teflon to reduce gouging. Design of the pull-out mechanism is based on the loads determined from the test.

PERMEABILITY OF GRAPHITE TO HYDROGEN FLOW

In the NRX-A reactor, there is a graphite seal between the core and graphite barrel. During operation, hydrogen can leak around and through this seal. This test was conducted to determine the best available graphite for minimizing such leakage. The results indicate that ZTA graphite is most resistant to hydrogen permeation. These findings resulted in the impregnation of the NRX-A graphite barrel.

NOZZLE END SEAL SPRING DEVELOPMENT

This project was conducted to determine a design for a nozzle end seal spring utilizing carbon cloth. At the time, carbon cloth was considered as the material to be used for spring construction. The static tests proved that grade WCB graphite cloth met the desired load deflection characteristics required. Later dynamic tests showed that it was not capable of returning to original shape as fast as the loads were applied.

NRX-A1 INNER REFLECTOR AXIAL SPRINGS

A series of springs are required to seat the inner reflector against the nozzle plate. A test was conducted to evaluate a spring from the Raymond Spring Company (Part No. 963 B 450). The technical data and results indicate

that this particular spring is not adequate to meet the design requirements.

PRESSURE TEST OF ALUMINUM BARREL

The reactor core has a graphite inner reflector barrel which fits into an aluminum shell. The graphite barrel has diamond shaped cutouts around its outer surface to seat the lateral support system. This test was conducted to determine if the aluminum sheath would depress into these cutouts. A 0.032 in. sheet of 2219-T31 aluminum was put under a 150 psig pressure to simulate operating conditions. The technical data and results verified that the test specimen would be adequate to meet operating requirements.

MISCELLANEOUS

The friction between adjacent reactor components has been an important design consideration and a number of tests have been run which provided for the more comprehensive current program as described in WANL-TNR-095. One of these preliminary activities involved a report on the properties of commercial graphite. Another test, entitled "Friction Binding of Sliding Parts" was performed as a part of a backup design effort in the lateral support seal area.

A number of tests were run to support the design effort in evolving the present reactor configuration. A study entitled "Core Bundling Periphery Dimensions" led to a statistical dimensional study of core bundling. Tests on threaded fuel elements, instrumentation lead brackets, interlocking of support blocks, support cone cryogenic strength, and FCX element load test have been run and the results furnished to the design activity involved. All results contributed in a material manner to the reactor design.

Other tests have been performed in direct support of the reactor assembly effort and for quality control determinations. Instrumented support blocks were tested to determine the loss of strength due to material removal. The NRX-A1 flow orifice removal following assembly, and the bonding strength of orifice jets were reported to design personnel. Deflection versus strain for leaf springs and

WANL-TME-573

a strain gaged spring assembly were run to obtain immediate information on leaf spring variations. A large number of graphite pins was broken to correlate results with x-ray inspections.

Several tests have been run on NRX-A1 components to check out LASL designs in support of the present design effort. The NRX-A1 lateral support, the NRX-A1 leaf spring, and reflector upper support ring have been tested.

Basic capabilities of beryllium shapes have been determined and reported. Properties of threaded forms were determined.

CONCLUSION:

The foregoing are only highlights as was originally indicated. A more comprehensive summary is available in the Year End Program Report. Detailed reports on the various tests are available as shown on the various distribution lists you will receive. The department has been able to fulfill its originally intended role of supporting NERVA reactor subsystem mechanical design and with the Program Redirection, it is also performing its role in component qualification testing required to establish the necessary confidence for conducting the forthcoming full-scale reactor tests.

S-N CURVE FOR NINETEEN HOLE CHESWICK ELEMENTS
UNFUELED AND UNCOATED

595908A

A 1-8 PRELIMINARY DATA

595909A

YIELD STRENGTH (.2%)
INCONEL X 750 & INCONEL X 718 NRX-A DESIGN TIE RODS
595910A

ULTIMATE TENSILE STRENGTH
INCONEL X 750 & INCONEL X 718 NRX-A DESIGN TIE RODS
595911A

TENSILE LOAD VS RESONANT FREQUENCY
REGULAR TIE ROD, INCONEL X 750

595905A

MODIFIED GOODMAN DIAGRAM
AMBIENT TEMPERATURE

595906A

Experimental

Analog

Typical Acceleration of Inner Reflector
Below Resonance

Typical Acceleration of Inner Reflector
Above Resonance

Typical Core Acceleration

Comparison Of Analog And Experimental Results For The Partial Core Test

NE 1314

NRX-B REACTOR DESIGN

PURPOSE:

Original Program - WANL-WP(J)-015 dated October 29, 1962

"To establish the layout and detail design of flight type reactors based on the KIWI B-6 for application in NRX-B reactor tests. This work is to serve as a basis for development to a qualified flight status."

Redirected Program - WANL-WP(J)-017 dated May 17, 1963

"To develop the design of one or more flight type reactors, designated NRX-B, intended to serve as backup designs for the NRX-A program. The objective is to develop designs for entire reactors and for portions of reactors which solve the problems encountered in the NRX-A design in new and different ways. This could lead to reactors of totally different design than NRX-A or to concepts of portions of the reactor which can be incorporated in modified NRX-A designs."

EARLIER WORK BACKGROUND:

During Contract Year 1962 a preliminary reactor design had been defined which employed a cooled metal outlet end support plate. The reactor was designed to fit into the existing NRX-A pressure vessel. Contract Summary Report (WANL-TNR-076) mentions this activity.

Several support plate designs were studied. A design assembled mechanically from machined forgings was preferred, and a plastic model of it was ordered for structural testing.

A stress analysis of the graphite transition blocks used between the fuel elements and the support plate indicated that they had a high probability of survival.

ORIGINAL PROGRAM PROGRESS: - C.Y. 1963 (WANL-WP(J)-015)

The original Work Program specified that the NRX-B reactor would have a beryllium reflector and be like NRX-A except for the cooled metal support plate at the core outlet

end.

Therefore, during the earlier portion of C.Y. 1963, effort was concentrated on NRX-B components which were not common to the NRX-A reactors.

Cooled Support Plate

Stress and thermal analysis of the preferred design continued and indicated adequate strength and cooling.

Several support plate and transition block combinations were studied including transition blocks supporting seven, twelve and nineteen fuel element clusters. The support plate used with the nineteen element clusters had coolant passages which were less restricted than the other designs.

Figure 1 shows this design. Tubes and baffles pressed into holes in the machined forging provide coolant passages.

Figure 2 is a photograph of a 1/6 sector plastic model of the plate. The black parts are pyrographite insulation and structural graphite retaining pieces. Drawings were released of a metal model of the complete plate for use in component testing.

Figure 3 shows an instrumented plastic model which was ordered during C.Y. 1962 and structurally tested this year.

Transition Blocks

Figure 4 shows one of forty (40) graphite transition blocks which were procured and structurally tested. The test results (WANL-TME-440) indicated that the analysis (WANL-TME-441) was conservative and that the component reliability required in NERVA should be achievable.

Hot Flow Test Rig

Figure 5 shows the hot flow test rig which was released for procurement. In it a single axial flow tube of the cooled support plate can be exposed to simulated reactor operating conditions. Hot hydrogen is supplied from the fuel element corrosion facility. Cold hydrogen coolant is provided from a separate source. The rig will be used to verify plate design details experimentally.

Tangential Lateral Support

Figure 6 shows the tangential lateral support design devised to align the core radially and tangentially, with the cooled support plate. Low rate springs bundle the core and allow it to expand freely. Lateral g forces are carried out tangentially through separate high rate springs from segments which interlock with the outer fuel elements. Drawings of this design were released to procure shock and vibration test hardware.

Figure 7 shows the tangential support system combined with the peripheral seal at the core outlet end proposed for NRX-B. A shock and vibration analysis (WANL-TME-443) was performed on this design.

Thus, at the middle of the Contract Year, the NRX-B project was directed towards the refinement of a reactor with a cooled metal support plate and a tangentially supported bundled core with an outlet end seal.

REDIRECTED PROGRAM PROGRESS: - C.Y. 1963 (WANL-WP(J)-017)

The redirected program (WANL-WP(J)-017 dated May 17, 1963) emphasized the development of new designs and concepts which could solve potential NRX-A problems in different ways. A creative phase was specified during which new concepts would be defined, followed by the evaluation and selection of two preferred conceptual designs. This work was completed on schedule and reviewed with NASA and AGC representatives at the NRX-B Review Meeting held at WANL on August 6 and 7, 1963. All of the work scheduled in the Milestones was accomplished.

The concepts which were initially identified as candidates for study and evaluation were:

1. Counterflow tie tube structures.
2. Cooled metal hot end support plates.
3. Graphite hot end support plates
4. Modular cores.
5. Multipass cores.

Counterflow Tie Tubes

Figure 8 is a photograph of a model of the concept. The tie tubes are supported from a plate at the core inlet end as in the NRX-A design. Design studies of several arrangements were made. The designs were analyzed and the bundled core with individual coaxial tube assemblies appeared to be the most practical version of this concept, particularly with respect to the sealing of core leakage. The design has a greater thermal margin than the NRX-A and can be operated to as low as 35% of the reactor design flow without reduction in outlet gas temperature. It was recommended for application to later NRX-A versions.

Cooled Metal Support Plates

Figure 9 shows a fabricated plate designed by LASL. When the KIWI B-6 project was cancelled, Westinghouse continued the plate procurement work as part of the NRX-B component program.

The design and analytical work on the NRX-B cooled metal plate which had progressed during the first half of the Contract Year was continued in the redirected program.

Analysis during this period (WANL-TME-468) confirmed earlier indications of substantial thermal and mechanical margins of safety. The latest designs appear to be capable of operating from full power down to 5% power and flow with complete loss of the pyrographite flow tube insulation. By varying the tube orificing, uniform flow distribution could be achieved with a total pressure drop of 28 psi through the plate at the design condition. Drawings were released for hardware to adapt the 1/6 sector plastic model of the plate shown earlier (in Figure 2) for use as a flow distribution test component to check the analytical results.

Graphite Plates

Figure 10 shows a graphite plate supporting a modular core. This plate had flow holes which changed direction in the interior with abrupt section changes, thermal gradients, and manufacturing problems.

Figure 11 shows a more practical design which was analyzed in more detail. Maximum tensile stresses were less than 1000 psi which appears to be quite acceptable in RVA graphite. This material is available in blocks of suitable size for the manufacture of a segmented plate of this design. Judging from the analysis performed on this design, the segmented design is theoretically adequate. However, practical problems such as coating and poison wire insertion have no obvious solution. It was, therefore, decided and proposed in the NRX-B Review Meeting, to discontinue the design work on graphite support plate concepts. To better understand some of the more practical problems, including manufacture, it was proposed to procure and test a graphite plate under simulated reactor loads. Drawings have been released and material ordered.

Modular Cores

Figure 12 illustrates a modular core supported on a cooled metal plate at the outlet end. The modules are supported in pin-ended fashion between plates at the inlet and outlet ends. This design was selected as the NRX-B Conceptual Design No. 1. Preliminary layouts were made of this design and an initial analysis was completed. A continuing effort is being made to analyze and optimize the details of this design as the main backup for NRX-A.

A number of alternate module detail designs have been defined for application to this concept.

Figure 13 shows a module constructed from clusters of fuel elements bonded together.

Figure 14 shows a cluster of elements loosely supported in a tubular graphite casing.

Figure 15 shows circular section fuel elements supported in internally webbed B1

type module structures.

The analytical work performed on the modules to date indicates the concept to be feasible structurally and thermally provided that the components to be bonded together are compatible with respect to thermal expansion. A comprehensive graphite bonding program has been started to develop joining techniques.

Multipass Cores

Multipass cores were identified as worthy of study in the conceptual design phase of the program, but the particular geometries studied did not appear to have any application as immediate backups to the NRX-A program. Multipass cores generally have problems involving the separation of the passes, sealing insulation, and excessive pressure drop.

Core Cage Structure

Figure 16 shows a core cage design which was developed during the conceptual design phase. Some of the supporting analysis is contained in WANL-TME-457, 449 and 454. Groups of fuel elements are supported in a metal honeycomb structure extending across the core. Initially, the design had axial control rods and the analyzed control worth was sufficient to shut down the reactor when immersed in water. Poison wires were, therefore, unnecessary in this design.

Figure 17 shows a core cage reactor which has peripheral drums only.

Figure 18 shows a similar reactor which also has a central beryllium island accommodating three additional control drums.

At the time of the design review meeting, a very limited amount of design and analysis had been completed, and this is now being continued on the latter two versions to enable a proper evaluation of the concept to be made.

The core cage designs have a low graphite volume fraction and require fuel loadings considerably higher than NRX-A and approaching 600 mg/cc. The thermal margin of the metal cage structure appears to be comparable with that of the NRX-A tie rods.

DESIGN REVIEW:

The most pertinent analytical work performed in the NRX-B program during C.Y. 1963 is contained in the brochure (WANL-TME-528) which was distributed to the attendees. This document contains twenty-seven TME's and reduced prints of the designs produced.

YEAR END STATUS:

Design and analysis work is continuing of the cooled support plate reactors. Major design effort is being concentrated on the graphite modular core as the main contender for selection as NRX-B. Some design effort is continuing to refine the tangential lateral support for the bundled core which is the second contender. Design work on the graphite plate reactors has stopped. Hardware is being procured for a segmented graphite plate to investigate manufacturing problems and for use in structural tests. The graphite cementing program is continuing and graphite modules of various designs will be released for procurement and testing in the near future. The procurement of cooled plate test models is also continuing. An intensive design and analysis effort is being conducted on the metal core cage design to determine whether it can be seriously considered as an NRX-B contender.

SUMMARY:

All milestones on NE 1314 project were completed on schedule.

NE 1316

KIWI B-6 SUPPORT

PURPOSE:

Original Program - WANL-WP(J)-015 dated October 29, 1963

"To provide for the mechanical design support and fabrication of cooled bottom support plates for the KIWI B-6A and B-6B reactor tests and mechanical and cryogenic testing of the support plates."

Redirected Program - WANL-WP(J)-017 dated May 17, 1963

"To provide for fabrication of test sectors of full size plates for mechanical and cryogenic testing, procurement of long lead tooling for bottom support plates, structural testing of plastic "forged" plate, and braze development."

PROGRAM BREAKDOWN:

The NE 1316 project was divided into four parts. The design liaison between Los Alamos and Westinghouse together with release of drawings under NE 1316A. The funding for the plate hardware was provided in the Program Control activity under 1316B. The Manufacturing Engineering Services were in NE 1316C. 1316D provided for the Quality Control work.

ORIGINAL WORK PROGRAM PROGRESS:

Figure 9 shows the LASL B-6 support plate design conceptually. The brazed Inco 718 structure is enclosed in a Nickel shroud assembly. The spaces inside the structure and shroud are designed to distribute coolant to the metal surfaces.

Initially, LASL planned to procure four (4) full size metal support plates. In October, 1962, nineteen preliminary sketches were received from Los Alamos for use in ordering raw material. Preliminary specifications for the Inco 718 material were prepared, and preliminary detail drawings were released for procurement of the Inco 718 parts.

During November, 1962, a meeting was held between Los Alamos and Westinghouse personnel to discuss manufacturing details, tolerances, specs., etc. Preliminary detail drawings were released for procurement of the Nickel tube castings.

During December preliminary drawings were released for procurement of the Nickel parts.

During this period Los Alamos decided that a small seven (7) hole metal model of the plates should be manufactured for use in preliminary testing to check the design.

The Los Alamos detail drawings for the full size plates and a preliminary assembly drawing of the seven hole plate were received by Westinghouse during January, and these were converted to the Westinghouse format and released on a preliminary basis. Quotes were requested for the two 7 hole plates and from one to four full size plates. Final detail drawings were also released for procurement of the Nickel tube castings.

Calculations were begun at Westinghouse at the request of Los Alamos to determine the pressure drop and flow distribution in the B-6 plate design under off-design conditions.

During February a plastic model of a 60° sector of the full size support plate and artist's perspective drawing (Figure 19) were made by Westinghouse at the request of Los Alamos.

A thermal analysis of the B-6 plate was started. The effects of flow rate and exhaust temperature on the temperature distribution in the plate were determined. The effect of loss of insulation was also studied. Where only one insulation tube was lost, the flow of coolant to the uninsulated tube was reduced 30%.

CANCELLATION OF LASL B-6 REACTOR PROGRAM:

During this period the Los Alamos B-6 reactor development program was cancelled. Westinghouse continued the plate procurement as part of the NRX-B component program. Only one full size plate was required, however, instead of the four planned by LASL.

REDIRECTED WORK PROGRAM PROGRESS:

Preliminary detail drawings for the seven hole plate were received from Los Alamos in March. These were completed, checked and revised where necessary by Westinghouse during April and May. During May the detail drawings of the seven hole and full size plates were released for procurement and instrumentation drawings for the seven hole plate were transmitted to LASL for their comments.

The final assembly drawings for the seven hole plate, specifying the instrumentation details were completed in June.

A contract was let to Portland Copper and Tank Works in June for the procurement of two 7 hole and one full size plate.

During this period, the brazing and Nickel tube casting portions of the program were completed.

In August a final release was made of the seven hole plates which clarified the procurement requirements for the instrumented and uninstrumented versions.

The thermal analysis of the B-6 plate under off-design conditions was completed, and the results presented during the NRX-B Design Review Meeting (WANL-TME-453). At 5% flow and power, the maximum plate temperature was 1150°R compared with 650°R at design conditions. Under accident conditions (loss of insulation in one tube) the maximum metal temperature was 2435°R at 100% flow and 2022°R at 5% flow and power.

YEAR END STATUS:

Tooling is 90% complete for the 7 hole plate and 20% for the full size plate.

Manufacturing is 50% complete on the 7 hole plate and 10% on the full size plate.

Delivery of the 7 hole plate is now expected in January, 1964, with delivery of the full size plate in February, 1964.

Delays in the B-6 plate schedule have resulted from problems experienced in the vendor's shop and from a delayed Los Alamos design schedule. Portland Copper and Tank Works are also working on the NRX-A core barrel and have tended to give priority to this work.

COOLED BOTTOM SUPPORT PLATE

PLASTIC SUPPORT PLATE INSTRUMENTATION

HOT FLOW TEST

TANGENTIAL SUPPORT SCHEME

LATERAL SUPPORT SYSTEM WITH TANGENTIALLY RESTRAINED CORE

Counter-flow Tie Tube

BRAZED SUPPORT PLATE

MODULAR CORE WITH GRAPHITE SUPPORT PLATE

GRAPHITE SUPPORT PLATE DETAIL

**PIN-ENDED MODULE FOR CORE WITH
COOLED BOTTOM SUPPORT PLATE**

19 ELEMENT CEMENTED TYPE MODULE

**PIN-ENDED MODULE FOR CORE WITH
COOLED BOTTOM SUPPORT PLATE**

CORE CROSS SECTION

**MODULAR GRAPHITE CORE WITH
COOLED BOTTOM SUPPORT PLATE**

MODULAR METAL CORE STRUCTURE

**37 ELEMENT CORE CROSS - SECTION
MODULAR METAL CORE STRUCTURE**

**61 ELEMENT CORE CROSS - SECTION
MODULAR METAL CORE STRUCTURE**

NE-1312

EVALUATION AND DEVELOPMENT OF REACTOR NON-FUEL MATERIALS

PURPOSE: " To provide reactor materials data on fabrication and irradiation behavior needed in support of design activities, hardware procurement and analysis of test hardware. To define operational limitations of reactor non-fuel materials. To develop new materials as required. "

VENDOR LIAISON, CONSULTATION, AND EVALUATION

As the NERVA project moved increasingly into the manufacture of components, supplier evaluation became more important. Evaluation entailed vendor facilities, personnel and procedures. Liaison was maintained with these vendors to provide technical help and to monitor critical operations. Arrangements were made to get better data on existing grades of graphite and to work towards the development of improved materials for NERVA requirements.

Consultation on materials problems with design, manufacturing, quality control, and purchasing personnel was a major function. Typically, requests for consulting help were urgent, were of short notice, were largely unforeseen, and were not amenable to advance planning.

MATERIALS EVALUATION, APPLICATION AND BEHAVIOR

Designers needed a materials manual as a source of data for design purposes. The review of the important materials requirements of NERVA components and the priorities subsequently determined guided the acquisition of data. The Material Data Sheets are intended to provide the special types of materials data needed by WANL designers, particularly the mechanical and physical properties as a function of temperature. Where available this information was obtained from the literature and from vendors. Critically needed data were and are being obtained by experiment. To date, over 300 Materials Data Sheets have been issued, covering 42 materials.

Two reports have been issued on tie rod materials; one for justification of the selection of Inconel X-750 as the prime material and Inconel 718 as the alternate; the second to describe the metallurgical aspects of the development of methods for fabricating tie rods by hot upsetting each end and cold roll forming of threads on one end.

WANL-TME-240 described the requirements of dome end support, core support, and bottom end support rings and recommended the use of the titanium-base alloy containing 5% aluminum and 2-1/2% tin, with low interstitial content (ELI grade). To obtain hardware, numerous discussions of processing and evaluation specimens were held with vendors to assure satisfactory properties. An addendum to TME-240 provided additional information regarding properties obtained in the production forged rings and discussed the effect of hydrogen and irradiation on the mechanical properties of the alloy.

The material for cluster plate application in the NERVA engine was found, during the year, to require a high thermal neutron capture capability to suppress the thermal neutron flux peak which would otherwise cause heating problems at the dome end of the core. Among the materials considered for this application were hafnium, a silver-indium-cadmium alloy, B¹⁰ bearing stainless steel, and 304 stainless steel containing gadolinium. Hafnium has definitely been eliminated because it has no advantage over Ag-In-Cd, it hydrides, and is considerably more costly. The Ag-In-Cd alloy does not have sufficient strength at temperatures over 600°F; however, it has been evaluated and selected as an alternate for the NRX-A test series because the environments in these early tests are not severe. B¹⁰ stainless steel was evaluated and possesses adequate properties under NERVA conditions; however, the boron 10 isotope is extremely costly and in short supply. During this program, the development of powder metallurgy techniques using a blend of 1.2% gadolinium oxide (Gd₂O₃) and type 304 stainless steel powders was begun. Sample material has been fabricated and preliminary results of evaluation are encouraging.

To date, most of the effort on beryllium has involved day-to-day problems involving design requirements, material characteristics, interpretation of nondestructive test results and vendor liaison. Experimental programs to evaluate the sensitivity of beryllium to surface damage and the properties of extruded high purity materials compared with "standard" purity materials have been initiated. The problem of reclamation of AEC-owned beryllium scrap for NERVA use was considered, and it was concluded that conversion of this scrap into powder is feasible. Work was also done on the reuse of beryllium reflectors considering the effects of irradiation on mechanical properties and density (swelling) as well as chemical purity considerations. From a metallurgical viewpoint, reuse of reflectors was shown to be feasible.

Methods were developed to compact lithium hydride powder by cold pressing and heat treating (sintering). Suitable heat treatment minimizes outgassing and samples were encapsulated in stainless steel containers for irradiation testing.

Although graphite problems have been both numerous and difficult, most of the graphite effort has involved day-to-day problems, such as: consulting on design problems, preparation of purchasing and processing specifications, liaison with vendors to aid in procurement and to improve quality of both structural and pyrolytic grades, evaluation of vendor qualifications, and determination of the disposition of sub-specification materials. During the latter half of the year, several experimental programs were begun, including:

- a). Evaluation of designs for graphite tensile specimens, including the glueing of specimens to simplify gripping problems and to permit evaluation of small specimens, the investigation of graphite specimen design to avoid fracture beyond the gage length and to obtain a true perspective of the real properties.
- b). Evaluation of properties of graphite grades of interest for NERVA application (tensile strength, flexural strength, density, thermal expansion, thermal conductivity and anisotropy). High temperature test capability, up to 3000°C, was established.

- c). Evaluation of advanced materials, particularly graphite for the NRX-B design of reactor.
- d). Evaluation of corrosion of graphite grades for which arrangements have been made to perform corrosion screening studies of graphite in a flowing hydrogen atmosphere.

FABRICATION DEVELOPMENT

Inasmuch as joining problems for graphite loom increasingly important and numerous, a comprehensive program has been planned to obtain fundamental information and thus acquire the required technology. A comprehensive graphite joining program has been formulated which is intended:

- 1) to survey graphite-joining literature and determine the state-of-the art,
- 2) to evaluate promising joining materials and processes selected from the literature,
- 3) to apply the best available joining materials to graphite component fabrication and repair operations in NRX-A and NRX-B configurations,
- 4) to develop better joining methods as necessary, and
- 5) to determine the properties and reliability of the joints.

To date, 1 has been accomplished and necessary materials have been procured.

An experimental impregnating apparatus has been built for evaluation of methods to control permeability to hydrogen.

The LASL process for making aluminum with natural boron and boron 10 additions by powder metallurgy has been established and several improvements made, particularly in the method of sintering of compacts by resistance heating. Production of control plates is nearly routine and the determination of physical, chemical and mechanical properties has started.

In order to adjust the reactivity of the nuclear core, control materials (nuclear shim rods) were required.

These are of two types; a poison shim rod, containing 25 volume percent tantalum carbide and a graphite shim rod. A process was developed for manufacturing these shim rods in the form of hollow cylinders (0.088" OD, 0.030" ID and 10" - 12" long).

Welding and heat treating procedures have been established for manufacturing core support barrels from two aluminum alloys, 2219 and 6061. The former possesses better mechanical properties after welding, but requires a carefully controlled solution heat treatment and liquid quench after welding. During manufacture with both materials, several shop difficulties have required considerable trouble shooting. For example, excessive dimensional distortion during quenching of the 2219 alloy and failure of the 6061 alloy to respond to aging were explained and remedial actions taken.

Techniques have been developed for leak-tight brazing of instrumentation leads from the reactor. To accomplish this, the stainless steel sheaths which surround the instrumentation leads are locally gold-plated to enhance brazeability. Also, OFHC copper inlays are brazed into stainless steel plugs using a hydrogen atmosphere. Finally, at assembly the instrumentation sheaths are induction brazed to the copper inlay. Each of these steps required development and evaluation.

Electron beam welding proved to be a satisfactory method of joining the end cap and cylinder components of control drums. These butt joints in aluminum alloy 6061 T-6 were unusually susceptible to weld metal cracking, but careful control of the developed procedure has resulted in satisfactory joints. As an aid in fit-up of the joint, a magnetic force field method of forming was developed.

RADIATION EFFECTS PROGRAM

An irradiation effects program prepared to acquire the data needed for the non-fuel materials comprising the NERVA reactor was proposed for AGC and SNPO approval. Irradiation was proposed at temperatures approximating operating temperatures and with dosages equal to calculated values. Dose rate is considered extremely important and thus the Plumbrook test reactor is considered highly desirable because of its high flux. For other materials the GTR facility was proposed.

All pre-irradiation tests were completed for the GTR 37/W001, 150-hour irradiation which began September 17, and terminated September 27, 1963. The pre-irradiation control tensile tests on plain and notched samples of Inconel 718 Inconel X-750, R-235, and A-286 were performed in the NASA remote tensile equipment at the GTR site. A preliminary capsule design concept has been completed for a proposed PBRF 37/W301 irradiation. In this concept, the capsule has a capability for holding tensile specimens and electrical resistivity specimens. Shim heaters will be employed to maintain the temperature at 1200°F. For ambient irradiations, the shim heaters are replaced with water coolant channels. In addition, materials have been procured or are on order for CY '64 irradiation programs.

Samples of lithium hydride have been irradiated with the following results: The flux wire analysis has been completed and a correlation between percent diametral expansion, irradiation temperature, and integrated flux has been made. Extensive swelling did not occur until an integrated flux of approximately 10^{16} n/cm² was obtained at temperatures of approximately 250°C. For integrated flux 10^{16} n/cm² and at all irradiation temperatures, the percent diametral expansion never exceeded 0.32%. It is significant that there was no marked difference in the behavior of shield capsules containing sintered LiH pellets and those containing cold pressed pellets.

SPECIFICATIONS

Specifications for materials and processes are essential to the NERVA program. Accordingly, a system has been devised and procedures established provide adequate specifications. These procedures include the gathering of the technical content, preparation in standard format, review by appropriate departments, revision as required, formal approval, reproduction and issuance. Approval signifies both WANL and vendor approval. Only recently has time permitted a systematic submission of specifications to vendors for formal comment and approval. During the year, 38 new materials specifications and 48 revised ones were issued. Also, 63 new process specifications and 30 revised ones were issued.

STATUS

The milestone objectives listed for the Project NE-1312 in WANL-WP(J)-017 have been completed during the year and there is no carry-over.

NE-1350

FUEL ELEMENT FABRICATION

PURPOSE: "To provide fuel elements and other graphite core parts for reactors required for the nuclear subsystem development program."

Table I summarizes the accomplishments in fulfillment of this objective.

There were 2884 enriched elements extruded for hot core usage. Of these, 1522 have been processed through graphitization and 1319 through machining. Seventy-six elements were shipped to LASL for the ZEPO experiments and an additional 86 have been sent to WANL for the evaluation of physical properties and hydrogen corrosion performance.

In support of the enriched element activity, 458 kg of coated particles were received (from two separate vendors) and processed through extensive receiving inspection tests prior to use.

Prior to the start of enriched fuel usage, 375 elements were produced with natural fuel as a final process checkout. Two hundred ninety-two of these were used for the development of coating parameters and a total of 62 were shipped to WANL for various tests.

Approximately 4,000 unfueled shapes (solid, single hole, and 13-hole) were extruded and of these 2,190 (in the as-graphitized condition) are on hand and will be used for the hot cores. There have also been 2,229 pieces sent to WANL for various tests and mock-ups and as feed material for smaller parts.

In the miscellaneous category, there were 57 support blocks coated and shipped to WANL. These were used for the "cold flow test" and for "poison shim removal tests." Also, keys were glued on approximately 170 peripheral elements for the cold flow core.

TABLE I

SUMMARY OF ITEMS PRODUCED IN
COMPLIANCE WITH OBJECTIVE I - PROJECT 1350

<u>TOTAL ENRICHED FUEL ELEMENTS EXTRUDED</u>		2,884
No. through Graphitization	1,522	
No. through Machining	1,319	
No. through Coating	160	
No. shipped to WANL	86	
No. shipped to LASL	76	
<u>TOTAL NATURAL FUEL ELEMENTS EXTRUDED</u>		375
No. through Graphitization	375	
No. through Machining	292	
No. through Coating	292	
No. shipped to WANL	62	
<u>TOTAL UNFUELED SHAPES EXTRUDED</u>		4,000
No. Graphitized and On Hand	2,190	
No. Shipped to WANL	2,229	
<u>MISCELLANEOUS</u>		
No. regular support blocks - coated	25	
No. round support blocks - coated	32	
No. keyed elements - glued	170	

NE-1351

FUEL ELEMENT PROCESS DEVELOPMENT AND CONTROL

PURPOSE: "To carry out the process development required to support the fuel element fabrication activity. To develop the specifications required for effective process control. To provide consultation and training services to candidate fuel suppliers. To establish a program for qualification of NERVA fuel suppliers."

PROCESS DEVELOPMENT:

MATERIALS:

The major materials effort during the year was directed toward the evaluation of fuel particles, the writing of improved fuel particle specifications, and the procurement of fuel beads for NRX-A2 and A3 production.

The first shipments by NUMEC of material produced to Specification XF-3, Rev. 1 (written in CY 62) were received on December 19, 1963. The information from processing the initial material resulted in the writing of a new specification (PDS 30050) for the second procurement. Contracts for the second procurement were negotiated with 3M and Davison, and 3M deliveries began in July 1963. Approval was given Davison Chemical to proceed with their production in September 1963. A total of 906 kgs of enriched particles were received during the year, and both NUMEC and 3M material have been processed into finished elements.

The continued evidence of fuel loss during coating led to efforts to procure more migration resistant fuel particles. In March 1963, purchase orders were placed with Davison and 3M for experimental fuel lots having variations in the processing and properties of pyro-carbon coated UC_2 particles. The objective of these orders was to examine the effects of particle size, substrate carbon content, coating thickness, coating temperature, and coating density. A summary of the materials is shown in Table II. Neither of these two vendors was completely successful in supplying a high carbon substrate (above

10 w/o) and delivery is not entirely completed to date.

Considerable work was also done to determine the extent and mechanism of uranium migration through the pyro-carbon coating. A Leco Induction Furnace for high temperature testing was requisitioned in March, received in April, installed in May, and made operable in late July. Thermal cycles have been of two general types: 1900°C for nine hours, or 2100 to 2400°C for 1/2 hour and evaluation of thermal damage has been carried out by micro-x-ray examination, alpha counting, nitric acid leaching, and hot HCl gas leaching. Results from this program have been erratic and at this time are of little value as methods for measuring particle damage.

PROCESS:

The basic fuel element manufacturing process was sufficiently refined to permit production of fueled and non-fueled core components for NRX-A2 and A3 and other special engineering tests. NRX-A2 was not completed as anticipated because of unexpected problems in coating process, but development of the coating process has advanced to a point that allowed repetitive production coating to begin late in the contract year.

The batch formulation equation developed during the last contract year was found to be in error when fuel particles from different suppliers were introduced. The discrepancy was found to be caused by the variation in bead composite density and bead total uranium content and the batch equation was modified to take into account these two factors. With this modified equation, it has been possible to calculate batch formulations so that the as-graphitized loading is within $\pm 1\%$ of aim loading.

The mixing stage of the process was examined to determine its effectiveness from the standpoint of uranium uniformity and to determine if particle damage was occurring during mixing. As shown in Table III, the mixing uniformity tests produced good results and showed that uniform uranium distribution was obtained early in the mixing process. The results of the particle density studies were not nearly as conclusive, however, there was positive evidence that some damage is occurring and that additional work is needed in this area.

TABLE II
DEVELOPMENTAL FUEL LOTS

<u>Supplier</u>	<u>Core Diameter μ</u>	<u>Total Carbon in Core, w/o</u>	<u>Type of Coating</u>	<u>Nominal Coating Thickness μ</u>	<u>Coating Density gm/cc</u>	<u>Coating Temperature °C</u>
Davison	50 - 100	9	Lam Pyc	25	-----	< 1500
Davison	100 - 150	9	Lam Pyc	25	-----	< 1500
Davison	50 - 150	9	Lam Pyc	25	-----	< 1500
Davison	50 - 150	> 9	Lam Pyc	25	-----	< 1500
Davison	50 - 150	> 9	Lam Pyc	30-35	-----	< 1500
3M	50 - 150	6-7	Lam Pyc	25	-----	< 1500
3M	50 - 150	8-9	Lam Pyc	25	-----	< 1500
3M	50 - 150	> 9	Lam Pyc	35 ₋₅	-----	< 1500
3M	50 - 150	> 9	Lam Pyc	25	1.4 - 1.5	1400
3M	50 - 150	> 9	Lam Pyc	25	1.8 - 1.9	1400
3M	50 - 150	> 9	Lam Pyc	25	1.4 - 1.5	1800
3M	50 - 150	> 9	Lam Pyc	25	1.8 - 1.9	1800

TABLE III
URANIUM HOMOGENEITY IN PROCESS

Sample Point	Sample No.	w/o Uranium
As Blended	S-2-1	20.35
As Blended	S-2-2	20.32
As Blended	S-2-3	21.58
After 1st Chop	S-3-1	20.46
After 1st Chop	S-3-2	20.38
After 1st Chop	S-3-3	20.40
After 1st Extrusion	S-4-1	20.40
After 1st Extrusion	S-4-2	20.33
After 1st Extrusion	S-4-3	20.45
After 2nd Chop	S-5-1	20.33
After 2nd Chop	S-5-2	20.36
After 2nd Chop	S-5-3	20.50
After 2nd Extrusion	S-6-1	20.50
After 2nd Extrusion	S-6-2	20.55
After 2nd Extrusion	S-6-3	20.53
After 3rd Chop	S-7-1	20.40
After 3rd Chop	S-7-2	20.39
After 3rd Chop	S-7-3	20.40

Examinations for structural integrity on early fuel element extrusions showed the existence of an extrusion pattern in the web sections between the outer 12 and inner 6 holes. This extrusion pattern was attributed to (1) non-uniform material flow through the die, and (2) the formation of a resin rich area and the mating of these areas as material is spread and reformed passing through the die. Die modifications were made which permit the insertion of calibrated peripheral feed holes and which in turn allow a balanced flow of material to be maintained through the die. This has enabled the extrusion pattern to be greatly reduced and only that portion which is related to the formation of binder rich skins remains. Additional work is continuing towards minimizing this condition.

With increased die flexibility and a better understanding of material flow and behavior in the die, work was directed towards producing extrusions that would require no machining to produce the final shape. Approximately 250 non-fueled elements were extruded with the following objectives:

- a. Attempt to only extrude holes to size
- b. Attempt to only extrude OD hex to size
- c. Attempt to extrude both to size simultaneously

No problems were encountered when extruding holes or OD to size individually. However, corner cracking did occur when an attempt was made to simultaneously size both OD and holes. Modification of the entry portion of the die hex plenum eliminated the corner cracking and 114 elements (unfueled) were produced without any cracking tendencies.

The dimensional reproducibility obtainable in the as-graphitized state is shown in figure 1 and, at this point, it appears the limiting factor in making extrusions to size will be the ability to provide graphitizing temperature profiles which will produce uniform shrinkage over the length of the element.

No changes were made to the process thermal cycles with the exception of the graphitizing cycle which was re-evaluated and a compromise reached between that cycle which would produce the minimum damage to the particles and still yield a dimensionally stable element. The final cycle established is five hours at 1900°C. The 1900°C temperature

was selected, based on metallographic studies, as the maximum temperature which could be used without rapid uranium migration. The five-hour time was selected after a time at temperature-stability evaluation as summarized in Table IV.

Both small scale laboratory development and actual production equipment were used to establish optimum leaching conditions for fuel elements prior to coating. Although all work indicated that a cycle of 1600°C for one hour in a mixture of anhydrous HCl and helium gases was adequate, there were indications that blistering during coating could be minimized by increasing the leaching temperature and a final temperature of 1700°C for leaching time of one hour was selected.

A furnace ("C" Furnace), designed for coating fuel element ends only, was re-designed with a bigger coil, and was plumbed with an appropriate gas distribution system to enable the full length vapor deposition of niobium carbide in the element cooling channels. Considerable effort was required to arrive at a final fixture design and equipment selection that would allow coating runs to be made with a reasonable assurance that the equipment would not fail or lines plug during the run. The first actual coating problems were related to the inability to produce full length coatings, and this was found to be caused by fixture leakage. A second major problem appeared when the operation was scaled up and the gas flows required to coat an increased number of elements caused a great shift in furnace temperature profiles, and the resultant deposition of unsatisfactory coating profiles. This problem was circumvented by differential packing of the furnace insulation to provide heat sinks at the furnace hot spots. However, it was soon found that some of the materials used were thermally unstable and efforts were then directed towards varying coating gas composition and flows to achieve satisfactory profiles. It was ultimately found that a two-step flow variation could be used by itself to produce acceptable coatings.

Because of the problems encountered with "C" Furnace, a new furnace with a much larger heat zone and with the additional ability to vary heat input to the coating zone was designed, procured, and installed. This gave much greater operational flexibility in that gas flow and temperature could now be varied independently. Typical coating profiles are

TABLE IV
DIMENSIONAL CHANGES DURING COATING

F.E. No.	Side	As Machined	As Coated	Δ Dimensions	Time at 1900°C Graph.
4469	A-D	.7533	.7532	.0001	5 hr.
	B-E	.7534	.7532	.0002	
	C-F	.7536	.7534	.0002	
4298	A-D	.7537	.7535	.0002	5 hr.
	B-E	.7534	.7532	.0002	
	C-F	.7538	.7537	.0001	
4452	A-D	.7532	.7532	.0000	5 hr.
	B-E	.7536	.7532	.0004	
	C-F	.7535	.7532	.0003	
4238	A-D	.7524	.7520	.0004	9 hr.
	B-E	.7533	.7530	.0003	
	C-F	.7531	.7526	.0005	
4239	A-D	.7535	.7529	.0006	9 hr.
	B-E	.7533	.7528	.0005	
	C-F	.7526	.7522	.0004	
4241	A-D	.7520	.7516	.0004	9 hr.
	B-E	.7523	.7519	.0004	
	C-F	.7533	.7531	.0002	

shown in figure 2 (a) and (b) with (a) depicting the profiles obtainable in "C" Furnace by flow adjustment and (b) showing the profile obtainable in coating "F" Furnace with temperature adjustments.

During the contract year, in excess of 70 coating runs were made in the production equipment and were required to develop both the equipment and process parameters which led to start of production coating late in the contract year.

In addition, two small developmental graphitizing furnaces were modified and converted for support block coating. Fixturing is designed to coat eight blocks at a time, but furnace temperature limitations are presently restricting block coating runs to four blocks per furnace.

All machining equipment and related tooling was activated with no major problems. Results to date indicate a tool life of approximately eight fuel elements per reamer and approximately 64 rods per broach.

NON-DESTRUCTIVE TESTING

The prime non-destructive test method for determining the structural integrity of fueled elements (radiography) required a re-evaluation when the fuel type changed from UO_2 to pyro-coated UC_2 . The new type of fuel, because of its larger particle size, produced a mottled radiographic film on which it was impossible to distinguish defects of a size previously detected on UO_2 -loaded fuel elements.

An eddy current test seemed to offer the possibility of better definition of the small defects in the new type fuel element and this was verified by LASL on their "bread board" eddy current system. Two eddy current systems similar to the LASL "bread board" design were built by WAFF and are being used to determine the structural integrity of all as-machined fuel elements. These two units are capable of testing a sufficient number of fuel elements to meet the production requirements, but to do this must be operated on a continuous basis. In order to provide more testing capability and to provide for any electrical or mechanical breakdown in the equipment, two more eddy current systems will be procured in the next contract year.

The present eddy current test equipment is not capable of testing the NbC lining for defects because of high background interference, and an eddy current test system, based on an Argonne National Laboratory design, is being built by Westinghouse Research with the help of Argonne Laboratory personnel and will be evaluated for examination of lining defects.

The thickness of the NbC liner which is vapor deposited on the ID of the fuel element channels must be determined to enable certain neutronic and thermodynamic calculations to be made, and to make corrections for fuel loading calculations as determined by gamma counting. The thickness of the coating is being determined non-destructively by a beta back scatter technique. During CY 1963, equipment was build at WAFF which gives the capability for measuring the thickness of vapor deposited NbC on either fuel elements or support blocks. Equipment has been procured to perform read-out either by meter reading or by decade scaler.

One disadvantage of the present probe design is the fragile nature of the light pipe, an inherent quality of glass or quartz rod of the diameter and length necessary for these probes. A plastic light pipe would be somewhat more rugged, but plastic light pipe introduces a new drawback in that it also scintillates when subjected to Beta rays and this adversely affects the sensitivity of the test. A possible method of improving the probe design is to replace the scintillation detector with a solid state detector, such as a lithium drifted silicon diode. Since the interaction of the back scattered betas with a diode produces an electrical signal at the detector, a coaxial cable is all that is needed to bring the signal out to an amplifier-scaler-timer combination for read-out. The fragile light pipe and the photomultiplier tube, which converts the light pulse signals to an electrical signal, associated with a scintillation detector are replaced by the coaxial cable when using a solid state detector. A lithium drifted silicon diode has been purchased from Solid State Radiations to investigate the feasibility of using this probe design for measuring NbC thickness. At the present time, calibration of this probe is in progress.

Gamma counting equipment has been designed for determining fuel element loading. One gamma count system and its associated transport system has been activated and the calibration of the system completed. This system has been in use for approximately six months.

A second gamma system for determination of uranium loading is now being delivered by the vendor (80 per cent completion in CY 1963). This system is essentially the same as the one now in use by WAFF, except that in addition to the printed record of counts per unit time, a tape will be punched containing the same information. This punched tape will expedite the calculations and return of information from the data handling system, since the information need not be manually key punched. The original gamma count system now in use is also being modified to have a punched tape output in addition to the printed tape. The data handling system will be discussed in more detail later in this report.

A third system, used for checks on green mix before extrusion was designed and has been activated for three months.

SPECIFICATIONS:

Process Specifications 294510, 294511, 294512, and 294513 were written to define the basic element manufacturing process. In support of these specifications approximately twelve AFF procedures were written to cover the detailed shop operations.

Some 40 QCII's which define the detailed inspection operations which must be performed were also issued and a Facility Quality Control Manual was started and is approximately 15% complete. Also, a system for collecting and reporting both in process and quality control data was developed during this report period. At this time, final computer programming is being done. This system will be used for storing and reporting data and will be capable of comparative listings to determine correlation between processing parameters and final element properties.

Complete element histories will be available and will be used to clear elements through Quality Control for shipment.

Finally, in the period January through September 1963, 393 new drawings were made which define equipment tooling and inspection apparatus. Of these, 211 were made at Astrofuel and the remainder subcontracted.

QUALIFY CANDIDATE FUEL SUPPLIERS

A formal briefing for the five selected candidate fuel suppliers was held in early November 1962, and at this session current process development data was presented by LASL and WANL representatives.

From February 1963 through September 1963, seven two and three-day training sessions were held at Astrofuel. Emphasis was placed on presentation of the most current process and materials development data available. Formal presentations were made to the candidate suppliers by Materials, Reactor Mechanical Design, and Reactor Analysis representatives located at WANL and a tour of related facilities at WANL was made with emphasis on the hydrogen corrosion facility.

In addition, 240 drawings, copies of all AFF's and QCII's, and various other pertinent documents were transmitted to the vendors at their request.

SUMMARY:

In summation, production of elements for NRX-A2 and A3 was started during the contract year. NRX-A2 elements, scheduled for completion, will carry over approximately 2-1/2 months into the next contract year. Specifications, as required, were issued for NRX-A2 and A3 production, but the summary fabrication report for NRX-A2 was not completed and will carry over approximately one month after the completion of A2 fabrication.

FIGURE IV

SIZE DISTRIBUTION FOR ALL COOLANT
PASSAGES IN 145 AS-GRAPHITIZED FUEL
ELEMENTS - 19 MEASUREMENTS PER ELEMENT.

SIZE DISTRIBUTION FOR ACROSS FLATS
DIMENSIONS OF 54 AS-GRAPHITIZED FUEL
ELEMENTS - 12 MEASUREMENTS PER ELEMENT.

NE 1353

EVALUATION AND DEVELOPMENT OF REACTOR FUEL MATERIAL

PURPOSE: "To determine and evaluate the properties and capabilities of the reference fuel."

At the start of contract year 1963, the primary objective of Project NE 1353 was to determine and evaluate the properties and capabilities of the reference fuel. A change order issued in June, 1963, increased the scope of activities to include work leading to the development of improved processing and operational capability of nuclear rocket fuel material. The activities performed on Project NE 1353 will be briefly reviewed under three major topics: fuel evaluation, fuel development, and irradiation activities.

FUEL EVALUATION:

THERMOPHYSICAL PROPERTIES

Equipment was developed to obtain data on the thermal expansion and thermal conductivity of fuel material over the temperature range from -196°C to 2500°C . Measurements of thermal expansion were initiated as a function of temperature, thermal and corrosion history, fuel loading, atmosphere, and manufacturing variables. Thermal conductivity measurements were initiated in the room temperature range. A topical report, "Thermophysical Properties of NERVA Fuel Material," WANL-TNR-131, was prepared which summarizes the details of this program.

Representative high temperature data on fuel shapes with fuel loadings ranging from 0 to 440 mgU/cc is shown in Figure 1. Attempts to obtain a quantitative relationship between uranium loading and thermal expansion have not been successful to date apparently due to local variations in the expansion of the fuel elements under study. This is clearly evident by comparing the data for 264, 310, and 444 mgU/cc material shown in Figure 1.

All samples evidenced shrinkage when they were heated above the graphitization temperature of the fuel (1900°C). This effect can be seen in Figure 2. In most cases the thermal expansion above 1900°C was obtained by use of the cooling curve trace.

Measurements on several samples indicated such a procedure was valid since the second heating curve superimposed over the first within experimental error. In one series of experiments, hydrogen was substituted for the normal helium atmosphere in an effort to determine its effect on fuel element contraction above 1900°C . Although the details of the shrinkage in the two environments varied somewhat for representative samples, no significant difference in overall behavior was observed up to 2200°C . Above this temperature, the corrosive effect of hydrogen on the apparatus prevented further comparisons.

Studies in the lower temperature range have not been emphasized to date, due to the extreme interest in the elevated temperature range. However, sufficient measurements have been made to adequately check out the equipment and to correlate the various pieces of apparatus. Measurements made at lower temperatures (25° to 1000°C) form a reasonable fit with the higher temperature data described previously.

Thermal conductivity equipment to cover the 25° to 2500°C range are being developed. Two measuring methods have been studied; one based on thermal diffusivity and the second based on radial heat flow. In both cases, an effort is being made to use samples sectioned from the existing fuel element geometry rather than manufacturing special samples which might give different results due to size or process variables.

The thermal diffusivity method is based on the use of an optical laser to pulse the sample with a high energy light beam. The time and temperature characteristics of the resultant thermal effect across the sample is studied by the use of a suitable high-speed temperature measuring device such as a thermo-pile, thermocouple, or optical pyrometer. From such measurements and the sample geometry, the thermal diffusivity is calculated. Preliminary room temperature values of thermal conductivity for fueled and unfueled graphite in the radial direction were found to be $90 \text{ BTU/in-ft-}^{\circ}\text{F}$. Thermal conductivity values of $80\text{--}85 \text{ BTU/hr-ft-}^{\circ}\text{F}$ were obtained by the radial heat flow method for unfueled graphite.

MECHANICAL PROPERTIES

The work performed on the mechanical properties of fueled graphite can conveniently be divided into four general areas: flexure, tensile and compressive strength measurements, and general studies. Each of these topics is discussed in some detail below. A more detailed summary of the results of this program is presented in WANL-TNR-130, "Mechanical Properties of NERVA Fuel Material."

Flexure Testing

Flexure testing constituted the most complete single phase of the mechanical testing program due to inherent advantages of sample preparation and test simplicity. The work performed had two distinct objectives; basic information for design purposes and quality control measurements.

The basic information studies were performed at room temperature on pre-production lots of fuel obtained from the Astrofuel Facility. Fuel loadings were varied from 0 to 444 mgU/cc. Other variables studied included axial strength variations, test volume effects, orientation of element during fabrication; and secondary effects such as leaching, coating, machining, and hydrogen corrosion. A modified "thirds" point method was normally used with a specimen length of 6-1/2 inch on the full element cross-section.

Significant results of the study are given in Tables 1, 2 and 3 and Figures 3, 4, and 5, which illustrate the effect of uranium loading and axial position on the flexure strength of the NERVA fuel element geometry. In general, it can be seen that little if any systematic correlation of strength with these variables was obtained. Although only a limited amount of data was obtained on the effect of coating, leaching, and hydrogen corrosion testing, it was observed that only minor variations were obtained on any of the samples tested. There are some indications that coating may strengthen the sample somewhat while machining, leaching, and hydrogen corrosion testing tend to decrease the strength.

TABLE 1

Room Temperature Flexure Strength Data on Pre-Production Fuel Elements

Rod No.	Extrusion Batch No.	Uranium Content mgU/cc	Description of Condition	Number of Test Specimens	---Outer-Fiber Stress (psi)---		
					Avg.	Range	Std. Dev.
4635	427	0	Unmachined	6	6390	5820-6680	330
4636	427	0	Unmachined	6	6550	6210-6800	200
4638	427	0	Unmachined	6	6360	6020-6660	220
3743	346	0	NbC Coated	6	6710	6550-6880	120
2795	255	180	Unmachined	8	6000	5190-6400	340
3121	286	180	Machined	7	5250	4410-6070	550
3136	287	180	Machined	6	5820	5570-6050	190
3148	288	180	Machined	7	5660	5060-6070	340
3166	289	180	Machined	7	5550	5110-6250	370
3178	290	180	Machined	6	5430	4780-5850	410
3187	291	180	Machined	7	5980	5660-6530	450
3196	292	180	Machined	7	6050	5740-6400	260
3204	293	180	Machined	7	4920	4360-5850	480
3222	294	180	Machined	6	5560	4120-6200	730
3247	296	180	Machined	6	5630	5370-5960	260
2796	255	180	Leached	5	5580	5330-5740	160
2807	256	180	Leached	6	5470	5280-5790	170
1326	115	220	Machined	7	4640	2090-5660	800
1742	155	220	Machined	6	5060	4600-5590	400
1749	156	264	Machined	7	5550	5280-5960	220
1757	157	310	Machined	7	4920	4630-5410	260
1758	157	310	Machined	8	4940	4310-5580	320
2824	257	310	Machined	6	5590	5090-5980	280
2818	257	310	Leached	6	5520	5300-5660	150
1768	158	343	Machined	6	4690	4090-5230	500
1763	158	348	Machined	8	4990	3120-5500	410
1765	158	348	Machined	6	4680	5500-5720	170
2836	259	417	Unmachined	9	4940	4180-5710	460
2839	259	417	Unmachined	9	5110	4720-5800	410
2842	259	417	Unmachined	9	5110	4700-5820	290
2844	259	417	Machined	7	5010	4270-5610	400
2831	258	417	Machined	4	4950	4730-5300	250

(continued on next page)

TABLE 1 (Continued)

Rod No.	Extrusion Batch No.	Uranium Content mgU/cc	Description of Condition	Number of Test Specimens	---Outer-Fiber Stress (psi)---		
					Avg.	Range	Std. Dev.
4237	391	444 (E)	Unmachined	7	4990	4630-5240	210
4305	396	444 (E)	Unmachined	7	5200	5010-5490	160
3993	371	444	Machined	6	5420	5340-5610	100
3994	371	444	100% Proof Tested	7	5620	5270-5920	250
2138	194	444	Leached	1	4160	- -	-
2135	194	444	Coated	6	4820	4580-5400	400
3809	-	444	H ₂ Tested	6	5530	5140-5840	240
4050	-	444	H ₂ Tested	5	5490	4870-6000	410

(E) Enriched Uranium

TABLE 2

Flexure Properties of Natural Uranium Beaded Fuel Elements

Uranium Loading (Weight per cent)	Number of Elements per Loading	Number of Specimens	Rupture Stress at Outer Fiber (psi)	
			Mean	Std. Dev.
8.9 (180 mgU/cc)	10	60	5,300	750
10.75	2	13	4,620	640
12.7	1	7	5,540	210
14.5	2	13	5,230	650
16.9	1	6	5,670	150
18.7	2	11	4,980	320
20.2 (440 mgU/cc)	1	6	4,810	360

The quality control flexure testing to date has been limited to unfueled elements designated for vibration and cold flow testing. These elements were manufactured by Oak Ridge (Y-12) from the basic formulation and process used by Astrofuel, but the graphitization temperatures were higher (2350° - 2400° C) than the normal fuel graphitization temperature (1900° C) used at Astrofuel. The average strength of these elements was about 5000 psi, slightly lower than the 5200 psi average strength obtained for all loadings on the Westinghouse material.

Compressive Testing

Compressive strength tests on full cross-section specimens cut from the reference fuel element shape were made at room temperature as a function of axial position and uranium loading. The samples were 1-1/4 to 1-1/2 inches in length with the end faces machined to obtain a high degree of parallelism. As in the case of flexure testing, no significant correlation between uranium loading and strength was found, nor was there a correlation between strength and axial position of the sample. The mean value for compressive strength for over 50 samples was $11,000 \pm 750$ psi.

Tensile Testing

A limited amount of tensile tests on specimens sectioned from the fuel element shape were performed. Samples cut both in the direction parallel and transverse to the extrusion direction were tested at room temperature. The results of these tests show the transverse strength to be 1900 psi and the parallel strength to be 2600 psi (for 180 mgU/cc loading). Consideration has been given to the significant difference between the tensile strength obtained from tensile tests (2600 psi) and tensile strength obtained from flexure measurements (5200 psi). Although the discrepancy between the two types of measurements has not been fully rationalized, several possible explanations are being investigated. The most probable explanations are thought to involve the following: the two test procedures actually are testing material of different properties (surface vs interior) and the fuel element is in a high degree of residual internal stress with the outside in residual compression.

General Studies

The work performed under this category includes proof test experiments, effect of imperfections on strength, residual stress studies, and a high temperature property measurements.

A single batch of unfueled graphite elements was obtained from Astrofuel and divided into a control and working group. The latter elements were subjected to "proof" flexure tests at 50% and 75% of the average flexure strength as determined by normal testing of three of the control group elements. The "proof tested" elements were then sectioned and a group of flexure specimens was tested in the normal way. In addition, fatigue specimens were taken from the group of proof tested elements and compared with samples taken from the original control group. The results of flexure and fatigue tests on the two types of specimens (pre-stressed and unstressed) were compared. No significant effect was observed as a result of the pre-stressing procedure. The results of these studies and other studies of fueled elements indicated that proof stressing might have merit provided that the corrosion and high temperature properties of the elements were not impaired. A detailed report of this program is given in WANL-TME-540, "A Preliminary Investigation of Mechanical Proof Tests on NERVA Fuel Elements."

Studies of residual stresses and the effect of characteristic imperfections (internal and surface) on fuel element mechanical properties are in an early stage. Both areas are being actively studied with particular emphasis being placed on obtaining a reasonable explanation for the apparent anomaly involving the tensile strength obtained from flexure and tensile testing.

Work on high temperature testing in tension and compression was initiated toward the latter part of the work period. The initial effort was directed toward developing adequate fixturing to sustain operation up to 2500°C. No fundamental difficulties up to 2500°C were encountered, but the initial results were complicated by alignment problems.

Preliminary compression data obtained on 7-hole specimens (0.5 inch diameter samples cut from the 19-hole element) at 2500°C indicate a compressive strength of 10,000 psi. Also, preliminary compressive creep experiments at 2500°C indicate the following minimum creep rates: 0.06%/hour at 2420 psi; 0.16-0.22%/hour at 4830 psi; and 3.8%/hour at 7260 psi.

Hydrogen Corrosion Studies

Studies of the behavior of reference fuel material during hydrogen corrosion tests were performed on pre-production and production grade fuel elements. The testing program contributed materially to the Westinghouse Astrofuel Facility coating development activity and has added substantially to knowledge of hydrogen corrosion attack of NbC coated reference fuel material. Many of the accomplishments of this program are summarized in WANL-TNR-129, "Hydrogen Corrosion Testing of NERVA Reference Fuel." The details of the testing procedure and equipment utilized to perform the testing are summarized in WANL-TME-502, "High Temperature Testing of Pre-Production Fuel Elements," and elsewhere in this report.

The goals of the program were to delineate the operating limitations of the reference fuel element in environments closely simulating reactor operation, to provide information leading to improved coatings, and to develop a satisfactory quality control test for the ND 10201 core. Production problems associated with the application of the NbC protective coatings and problems originating with the hydrogen corrosion test combined to disrupt the planned test program and force the prime effort toward the development of a satisfactory quality control test specifically for the ND 10201 core. Test results obtained in the latter part of the test program indicated that a satisfactory quality control test had been achieved and that future work could be directed toward the developmental program.

The prime criterion for element performance during hydrogen corrosion testing was considered to be weight loss assuming the element remained intact with no visible holes in the surface or communication between channels. Neglecting test temperature, weight losses of 12 grams or less were considered to be acceptable based on assuming carbon

TABLE 4

Hydrogen Corrosion Tests of Fueled Elements with Production
or Production-Quality Coatings Which Survived the Test Intact

Element Number	Coating Run No.	-----T _{max} (°C)-----		Time at T _{max} (min)	-- Weight Loss--	
		surface	interior		grams	(%)
4865	C-26	2200	2450	5.0	1.5	0.22
4438	C-26	2440	2750	2.8	26.5 ⁽¹⁾	3.9
7918	C-26	2440	2750	2.0	2.0	0.29
5121	C-27	2230	2490	5.0	2.0	0.29
5143	C-27	2470	2780	4.0	4.5	0.65
4867	C-27	2430	2730	5.0	5.0	0.72
5077	C-27	2200	2450	10.0	30 ⁽²⁾	4.4
7919	C-27	2130	2370	5.0	1.5	0.22
5205	C-27	2210	2460	5.0	3.5	0.51
5171	C-27	2330	2670	5.0	4.5	0.65
5166	C-29	2200	2450	5.0	7.0	0.99
8293	C-29	2200	2450	2.5	1.0	0.17
5243	C-30	2200	2450	5.0	3.5	0.51
5334	C-30	2230	2490	4.5	5.0	0.80
4899	C-30	2210	2460	5.0	2.0	0.29
5196	C-30	2210	2460	5.0	2.0	0.29
4875	C-33	2210	2460	5.0	3.0	0.44

(1) Outer twelve holes orificed with 60 mil orifices.

(2) Corrosion hole near center of element on a face; hydrogen attack of faces at hot end.

losses only. A summary of test results on elements which were taken from coating batches considered to be representative of production quality batches is given in Table 4. The histogram depicted in Figure 6 illustrates the relative survival rate (based on five minute test*) of these elements compared to elements considered to be inferior based on the apparent adherence of the coating. It can be seen that at surface temperatures up to 2300°C, all elements with present production quality coatings would be acceptable under the criteria indicated previously.

A key factor in interpreting the test results presented is the actual temperature of the test. Calculations based on theoretical considerations, experimental temperature measurements, and indirect evidence based on fuel particle migration studies indicate that a substantially higher temperature exists within the interior of a fuel element under test compared to the measured surface temperature. The resultant gradient, which varies along the length of the element according to the overall axial temperature gradient and on the exact test conditions may reach 300°C or higher at a surface temperature of 2450°C. A pictorial representation of this effect is illustrated in Figure 7. The result is that the electrical heating test is more severe than its reactor analogue if based only on the surface temperature. A quantitative weight loss relationship between isothermal radial temperature distributions and the condition exhibited in the present electrical heating test cannot be established with the information presently available. The result is that operating limitations imposed on the reference fuel based on surface temperature measurements are conservative.

Most of the elements submitted for hydrogen corrosion tests have been subjected to a detailed post-mortem analysis. As a result, techniques for evaluating fuel element performance have been developed which have contributed materially to improvements in the coating process. The role of many types of flaws commonly present in vapor deposited coatings were qualitatively studied with respect to their influence on corrosion behavior. Cracks were always found to be present at room temperatures due to the high

* The criterion used here was that the element survived intact with no obvious flaws.

In all cases the weight loss requirements were also met.

temperature at which the NbC is deposited and the much larger coefficient of thermal expansion of NbC relative to the graphite matrix. The nature and distribution of the cracks determine the degree of closure at higher temperatures due to the reverse expansion effect; and thus, how the coating will perform in service. Among the factors determined to be important during the investigation were coating thickness, grain size, adherency, appearance of coating flaws, and coating profiles.

Metallography and mercury porosimetry studies of corrosion tested fuel elements were found to be quite valuable to delineate corrosion effects on a local scale compared to the gross effects obtained by weighing. Prior to a catastrophic failure, it was found that two general types of corrosion attack occur as a result of coating imperfections; there are pore enlargements and local undercutting. The relative severity of the two effects depends on the local temperature and the type of defect present. Both types of corrosion can occur at the same location.

Detailed studies of the NbC coating (by chemical analysis, microprobe analysis, x-ray analysis, isothermal heating, and detailed metallography) were performed to more adequately evaluate the effect of impurities (uranium in particular) and elevated temperatures on coating behavior. Definite evidence of melting, apparently due to uranium migration, was found at approximately 2500°C. The final result of these studies is expected to be the establishment of a corrosion failure mechanism and an accurate determination of maximum fuel element capabilities.

High Temperature Studies

Studies of the effects of parametric time-temperature treatments on the uranium migration behavior of pyrocarbon (PyC) coated UC₂ fuel particles was completed. Particles supplied by NUMEC and 3M which are planned for use in NRX-A2 and A3 tests were studied. Alpha counting of coated particles before and after heat treatment has been used as a measure of the rate of uranium migration. Heat treatment temperatures range from 1600° to 2600°C, and times range from 1/2 to 20 hours. The results obtained on NUMEC and 3M coated particles were similar.

Two changes in the rate controlling process were indicated, one at 2150°C and another at 2450°C. The activation energy calculated for the migration rate between 2150° and 2450°C is approximately 135 kcal/mole. The rate controlling mechanism in this temperature range is believed to be one of uranium vaporization from the coated UC₂ particle. The heat of vaporization for UC₂ has been reported to be 133 kcal/mole at 2000°K.

A possible rate controlling process for uranium migration at temperatures below 2150°C is the ordering transformation which takes place in the PyC coating. This transformation is known to be rapid at temperatures above 2150°C. The transformation products appear as a zone which progresses through the coating thickness. The structure of this zone appears to be transformed PyC, as indicated by electron microscopic examination. Electron micro-probe analysis of the pyrocarbon coating indicates a nearly uniform uranium concentration in the transformed zone. This zone appears in electron micrographs of both 3M and NUMEC particles; however, the grain size of this zone in the NUMEC particles is so small that it is difficult to see under the optical microscope at 400X.

Micro-radiography of coated particles and of fueled graphite has shown the progress of uranium migration in the PyC coatings, through the graphite matrix, and into the NbC liners. However, the method has not yet been developed to obtain a quantitative measure of migration. The micro-radiographs also provide information on the distribution of fuel particles within the fuel cross-section. Stereo-micro-radiographs have been obtained in thin (30 mil) sections of fueled graphite. The three dimensional distribution of the particles in the matrix is visible by means of a stereo-viewer. Clumps and stringers of coated fuel particles can be seen.

Reactor Shutdown Studies

The feasibility of using nitrogen and air as coolants for reactor shutdown cooling or as interim storage atmospheres was studied. A detailed analysis of the results may be found in WANL-TME-564, "A Preliminary Report on the Use of Nitrogen and Air for Reactor Shutdown."

The test results reveal a relatively constant pattern. Oxygen and or moisture were observed to react extensively with fueled material where uranium migration had occurred extensively. The rates of these reactions were relatively slow at 100°F (38°C), but they increased to significant proportions at 300°C. Severe reactions were characterized by cracking and crumbling of the graphite matrix and a general discoloration and separation of the NbC liner from the channel wall. Further studies are in progress to define the type of atmosphere required during reactor shutdown from full power operation.

FUEL ELEMENT DEVELOPMENT:

FUEL PARTICLE DEVELOPMENT

Several existing processes for preparation of development quantities of fuel particles have been studied. The general methods used for preparing spherical fuel particles were reviewed. Based on an evaluation of equipment requirements and characteristics of the fuel product, two fabrication methods were selected: the agglomeration-packed bed method, and the arc melt-plasma spheroidizer method. Test batches of UO_2 -C agglomerates in the proper size range have been prepared by the first method and a large induction furnace for packed-bed sintering and spheroidization has been used to produce laboratory quantities of uncoated UC_2 fuel particles for physical and chemical property evaluation. Results indicate that the packed-bed method will be suitable for the fabrication of both reference and advanced fuel particles. An arc plasma apparatus already installed in the laboratory is being modified to permit its use as a particle spheroidizing apparatus. Powder pressing and arc melting facilities for the production of homogeneous fuel composition ingots are in operation and studies have been initiated to utilize this equipment in conjunction with the arc plasma unit to produce fuel particles by the second method.

Concurrent with this effort, studies have been made of possible means to produce improved types of fuel systems with high temperature properties superior to the reference UC_2 system used at the present time. Toward this end, a number of lots of uncoated

particles in single and mixed carbide systems have been purchased. Of particular interest are hyperstoichiometric UC_2 and UC-NbC and UC-ZrC solid solution binary systems. Test procedures for evaluating the properties of the base particles have been formulated.

FUEL PARTICLE COATING

To achieve coating capability, a fluidized bed system for intermediate and high temperature fuel particle coating has been constructed. This equipment was designed to permit pyrocarbon coating at both low and high temperature deposition temperatures. The equipment also has the capability of forming refractory-metal carbide coatings by halide vapor reaction. Further, co-deposition of complex carbide-carbon coatings may be carried out simultaneously by introducing the proper reactants into the system under controlled conditions.

EVALUATION OF PRODUCTION FUEL PARTICLES

The evaluation of the coated fuel particles used in the production of NERVA fuel elements is required to insure satisfactory behavior in fabrication and reactor operation.

A number of fuel evaluation tests have been studied in detail and recommendations have been made as to which techniques may be the most reliable and economical for determining high temperature fuel particle properties. In addition, the studies have led to suggestions for modifications in the present fuel particle procurement specification, to improve the performance capabilities of the fuel elements.

FUEL ELEMENT COATING DEVELOPMENT

This work has been divided into two categories: initiation of laboratory experiments to study parametric variations on the basic coating process to permit optimization of production fuel element coating, and a study of alternate coating processes to achieve improved coating capabilities.

A vapor coating apparatus has been constructed to investigate the production coating process. Short lengths of fuel element sections can be treated. Preliminary runs have been made using production flow conditions to produce NbC coatings in channels of unfueled

element sections. A series of parametric studies have been initiated involving the relationships of reactant concentrations, flow rates, temperatures, pressures, and substrate condition to the production and properties of NbC coatings. Raw materials also have been obtained to conduct vapor deposition with Zr, Ta, W, or Mo halide vapors.

To meet the requirements of the second phase of effort, a special quartz reaction tube apparatus has been designed to permit study of a reduced pressure thermal decomposition process wherein only the halide vapor is introduced to the reaction chamber and reacted at the hot element surface. The apparatus is designed to provide information on reaction kinetics, about which little is known at the present time. Additional use of this apparatus will be aimed toward study of refractory-metal organo-metallic compounds for low temperature carbide coating studies.

To improve control of the production coating process, a gas analysis apparatus has been constructed to monitor the effluent gases in the coating furnace. Provisions for quantitative analysis of salt, hydrogen, hydrogen chloride, and helium have been incorporated in this equipment. By knowing the concentrations of these gases as a function of operating conditions, the efficiency of the coating process and the maintenance of required coating profiles and thickness should be controlled in an improved fashion. The analytical equipment has been put through a number of coating runs and has demonstrated its usefulness.

FUEL ELEMENT BINDER DEVELOPMENT

The furfuryl alcohol resin (Varcuum 8251), used as a binder in the manufacture of fuel elements, was studied from the standpoint of quality control to determine resin characteristics that must be controlled in order to assure satisfactory performance during processing. Acceptance criteria for the resin were established based on measurement of solids content, viscosity, gel time, pH and boron content. Test methods for these determinations were specified. A report describing this work has been prepared, WANL-TNR-132, "Graphite Binder Studies".

IRRADIATION EXPERIMENTS:

Studies of the irradiation behavior of NERVA fuel material are being conducted in steady-state and transient irradiations in the General Electric Testing Reactor (GETR) and in the Transient Test Reactor (TREAT), respectively. The evaluation of NERVA fuel at equivalent full power operation (2-4 kw/cc) requires thermal neutron fluxes of $3-6 \times 10^{14} \text{ n/cm}^2 \times \text{sec}$ for times up to 30 minutes. This combination of flux and time is not available in existing irradiation facilities. The reactors used for the program bracket the two requirements. The GETR provides the required burnup and 20% of the neutron flux requirement; whereas, TREAT provides the neutron flux for periods up to 15 seconds. The GETR can be used to obtain preliminary information regarding the steady-state, high-temperature fuel material behavior with respect to fission product release, dimensional stability and fuel migration. On the other hand, TREAT is useful to study the effects of thermal stress on fueled graphite matrix behavior. A detailed summary of the results of the irradiation program is given in WANL-TNR-139, "Irradiation Studies of NERVA Fuel Material".

THERMAL STRESS EXPERIMENTS IN TREAT

Thermal stress experiments were carried out on NERVA reference fuel material employing three sample configurations. The first consisted of a 1/2 inch diameter 7-hole NbC lined, 3/4 inch long cylinders in standard autoclaves. The 7-hole fuel element sections were machined out of 19-hole hexagonal fuel elements. The second consisted of 19-hole NbC lined, 1/2 inch long fuel samples in autoclave capsules. The third consisted of the same fuel material as the second but irradiated in photographic capsules. Table 8 lists the irradiation conditions used in these experiments.

The experimental data indicates that the 7-hole cylindrical specimens exhibit cracking at energy inputs of 1300-1800 cal/gm; whereas, the 19-hole hexagonal specimens remained intact up to energy inputs of 2500 cal/gram. This represents anomalous behavior, since for the same energy input, the hexagonal samples with larger radial dimensions develop larger temperature gradients than the cylindrical specimens.

The explanation for the difference in behavior may involve the internal stress configuration within the fuel material. There is evidence that the as-manufactured hexagonal fuel material contains residual internal stresses which result from radial variations of preferred orientation. X-ray studies indicate a high degree of preferred orientation (basal planes parallel to the axial extrusion) at the surface of the fuel element. The radial variation of preferred orientation gives rise to residual compressive stresses at the surface due to the differences in thermal expansion coefficient with preferred orientation. Thus, an hexagonal cross-section sample with residual compressive stress at the surface would appear to be stronger than a sample cut from the interior containing smaller residual compressive stress. The results of mechanical property studies carried out on fuel elements are consistent with the proposed explanation.

A second series of highly instrumented transient experiments are being assembled. The 6-inch long fuel samples used in these experiments will be instrumented with 12 tungsten-rhenium thermocouples located in selected radial and axial positions both in fuel holes and in the webbing between the holes. Temperatures will be recorded as a function of time during transients. Pre-and post irradiation examination will serve to define material behavior as a function of the thermal gradients. Following the thermal stress experiments in a static helium environment, the work will be extended to an investigation of the effects of coolant flow (helium) passing through the fuel holes. This work will simulate the fuel-stress pattern during NERVA operation.

The studies completed indicate that unconstrained NERVA fuel material (19-hole, hexagonal cross-section) is resistant to thermal stresses resulting from energy inputs up to 2500 cal/gram. This energy input gives rise to a maximum radial temperature gradient of 600°C or more between fuel center and surface during the cooling cycle. The fuel material is remarkably resistant to thermal shock, having been heated from 25-3000°C in 0.5 seconds at maximum heating rates of 28,000°C per second without damage to the graphite matrix. Microstructural changes occur in the coated particles in the NbC liner under conditions which do not affect the graphite matrix. Experiments on 7-hole sections of the fuel material reveal anomolous behavior which appears to be related to the residual stress distribution within the fuel elements.

TABLE 8

Irradiation Conditions and Post-Irradiation Observations of Thermal Stress Experiments in TREAT

TREAT Transient Number	Sample* Config- uration	Energy Mw-sec	Period ms	Energy Input cal/gm	Peak Temp. ** °C	Maximum Center to Surface Temp. Difference	Post-Irradiation Observations
CEN-126	1	430	59	980	1700 (E) 2650 (A)	90 (C)	Intact no visible cracks.
CEN-127	1	580	50	1320	2025 (E) 3350 (A)	105 (C)	Several large longitudinal cracks in webs
CEN-128	1	720	42	1660	2275 (E) 3625 (A)	120 (C)	Fractured longitudinally into three segments
CEN-129	1	820	40	1860	2450 (E) 3625 (A)	130 (C)	One small longitudinal crack on surface
CEN-148	2	490	57	1130	2380 (M) 3000 (A)	180 (M)	No visible cracks
T-15	3	300	76	1370	2360 (M) 3400 (A)	190 (M)	No visible cracks
T-16	3	388	70	1750	2600 (E) 3625 (A)	345 (C)	No visible cracks
T-18	3	548	52	2520	3100 (E) 3625 (A)	610 (C)	No visible cracks
T-17	3	644	48	2960	3350 (E) 3625 (A)	700 (C)	Radial crack on surface in webbing near apex of the hexagon

- * Sample configurations: 1 - 1/2" diameter, 3/4 inch long fuel in standard autoclave
 2 - 3/4" hexagon, 1-1/2 inch long fuel in standard autoclave
 3 - 3/4" hexagon, 1-1/2 inch long in photographic capsule

- ** (A) Adiabatic sample temperature (maximum temperature of 3625°C).
 (B) Estimated from calibration curves of measured temperature versus energy input (cal/gm).
 (M) Measured with W-Re thermocouple and optical pyrometer.
 (C) Estimated from calculations. Cylindrical samples were adjusted in proportion to the square of the diameter to put on the same basis as the hexagonal samples.

STEADY-STATE IRRADIATION EXPERIMENTS

An experimental program was initiated to ascertain the irradiation behavior of NERVA fuel material as a function of irradiation temperature and time. Of particular interest is the effect of neutron irradiation upon fuel migration, NbC liner integrity, mechanical properties, thermal conductivity, and restart capability. The initial phase of the experiments involved examination of the irradiation behavior of NERVA fuel material irradiated in capsules FP-3 and FP-4 at the General Electric Testing Reactor. The conditions of the irradiations were as follows: FP-3 was irradiated for 15 minutes at a neutron flux of 1.4×10^{14} and a maximum temperature of 1800°C ; FP-4 was irradiated for 15 minutes at a neutron flux of 1.2×10^{14} and a maximum temperature of 2175°C . Preliminary metallographic examination of the irradiated pellets suggested that some migration occurred as a result of irradiation of fuel material at 1800°C for 15 minutes. Migration of uranium in fuel material was not expected at as low a temperature as 1800°C . Additional metallographic studies of the irradiated material are in progress.

The second phase of the program involves six capsule assemblies (see Figure 8). Two of the six capsules will be used to collect data related to the in-pile thermal conductivity of the fuel material. Measurements will be made at centerline temperatures of 1000, 1500, 2000 and 2500°C . Each of the remaining four capsules will contain pellets which have received special heat treatments. The irradiation samples will include: as-fabricated samples; samples exposed to the hydrogen corrosion environment; and samples exposed to a hydrolyzing environment. The irradiation conditions for the four capsules are outlined in Table 6.

TABLE 6 - IRRADIATION PARAMETERS

<u>Fuel Temperature, $^{\circ}\text{C}$</u>	<u>Fuel Burnup Fissions/cubic centimeter</u>
2500	3.7×10^{16}
2500	1.5×10^{17}
2100	1.5×10^{17}
2100	3.0×10^{17}

The burnups listed in Table 1 correspond to those in NERVA for 5, 30, and 40 minute operations with a maximum power density of 4 kilowatts per cubic centimeter.

The capsules will be irradiated within the 2-1/2 inch diameter trail-cable facility of the GETR. The maximum perturbed flux available in this facility is approximately 1.5×10^{14} n/cm² sec. The advantage of the GETR trail-cable facility is that the capsules may be inserted and removed during a reactor cycle, thereby facilitating short term irradiations.

PRELIMINARY CONSIDERATIONS OF DYNAMIC IRRADIATION EXPERIMENTS

Preliminary studies were made to establish the reactor and system requirements for a dynamic fuel material test in a neutron environment. It was established that the reactor requirements were 5 to 60 minute cycles at thermal neutron fluxes of $3-6 \times 10^{14}$ n/cm² sec. It was established that these conditions can not be readily met by existing reactors.

The design of a once through in-pile hydrogen or helium loop, which would accommodate a 3 foot long section of NERVA fuel material, appears feasible. The major problem areas were determined to be the design of a removable in-pile test section and design of piping and heat exchangers capable of withstanding the high gas temperatures. A report summarizing the needs for dynamic irradiation studies was prepared, WANL-TME-475, "Justification and Preliminary Program Description for Dynamic Irradiation Testing of NERVA Fuel Material".

CONCLUSIONS:

Technical goals were achieved in all areas defined by the project objectives. All milestones were met during the contract year. A small amount of carry-over work will be required to complete the quality control evaluation (destructive testing) of NRX-A2 fuel elements.

THERMAL EXPANSION OF A NERVA FUEL ELEMENT

THERMAL EXPANSION OF NERVA FUEL ELEMENTS

--- No. 3994, 444 loading	--- No. 1637,
..... No. 1757, 310 loading	 No. 4005, 190 loading
--- No. 1749, 264 loading	--- No. 4638, unfueled, single hole

FIGURE 5: EFFECT OF STRESS ON PROBABILITY OF RUPTURE, IN FLEXURE, OF SPECIMENS CONTAINING UP TO 444 MG/CC NAT. U .

FIGURE 6: EFFECT OF TEST VOLUME
ON FLEXURE STRENGTH

DEPENDENCE OF SURVIVAL RATE OF COATED FUEL
ELEMENTS ON TEMPERATURE OF HYDROGEN CORROSION TESTS

APPROXIMATE TEMPERATURE DISTRIBUTION IN AN NRX-A FUEL ELEMENT

DURING A HYDROGEN CORROSION TEST

Test temperature - 2450°C, surface temperature measured at 48 inches. The shaded zone is the region where a corrosion band appears over the surface temperature range 1500°-1700°C (internal temperature 1670°-1890°C).

595853C

IRRADIATION CAPSULE

NE-1361

MANUFACTURING ENGINEERING AND SERVICES

PURPOSE: "To provide Manufacturing Engineering support for Design activities to insure manufacturability of reactor components, test rigs, associated tooling, and handling equipment; provide manufacturing engineering guidance and support to fabrication and procurement; investigate supplier qualifications and carry out necessary process development programs.

"Based on overall project direction, analyze and prepare detail ordering requirements for procurement. Initiate procurement on all hardware, tooling, and services. Establish and maintain priority scheduling and inventory control on all work in process. Receive, store, dispatch, and maintain records of finished parts, raw materials, and other Government Property."

INTRODUCTION:

In order to develop the proper vendors it was necessary to visit approximately 280 sources. Visits were made in conjunction with Quality Control and Procurement. The present approved list consists of 88 approved and 38 conditionally approved vendors. Sixteen (16) vendors were disapproved. In many instances vendors who were approved for specific components were re-evaluated. This was required because of additional specialty or broader manufacturing process requirements.

During the past year, all original design engineering drawings were signed by Manufacturing Engineering. Cost estimates were developed on each component released. The estimate consists of detailed material, labor and tooling cost, and related lead time schedules. Reactor components manufactured of Pyrographite, Graphite (both extruded and molded), Beryllium, Aluminum (2219), Inconel (718; X-750), Titanium, and various Stainless Steels require a more diversified processing approach than normal. The difficulties encountered in predicting material behavior and tolerance limitations necessitated

close vendor follow up for determining the proper methods, tooling, and processing techniques required.

To review each and every detail part would take considerable time. However, here are a few parts that required considerable input from Manufacturing Engineering to aid in developing methods, tooling, and fabrication techniques.

There are three items on which the development of manufacturing procedures will be covered in detail: poison wires, orifice jets, and tie rods.

POISON RODS:

The filling of Boron Carbide into a stainless steel tube (figure 1) with an extremely large length to diameter ratio (54" to .067 O.D. x .048 I.D.) with a straightness requirement that it must fit freely in an .090 inch diameter hole made it imperative that a manufacturing development program be initiated.

The general manufacturing steps were:

1. Fill SST Tubes with B_4C - Vibratory Compaction.
2. Reduce tubes to finished size.
3. Straighten.
4. Cut to finished length.
5. Weld bottom end closure.
6. Braze head to rod.
7. Coat with Polyethylene.

1. Filling stainless steel tubes with Boron Carbide by vibratory compaction

The initial efforts were directed at determining the starting dimensions of stainless steel tubing and mesh size of Boron Carbide.

Vibratory compaction experiments showed that a usable density (approximately 55% of theoretical) could be obtained using a 1:1 ratio of 180 and 400 mesh B_4C in a stainless steel tube (.125 O.D. x .012 wall).

WANL-TME-573

2. Reduce tubes to finished size - drawing or swaging

In order to reduce the stainless steel tube to finished size (.125 O.D. x .012 wall to .067 O.D. x .0095 wall) and at the same time increase the theoretical density of B_4C (55% to 80%), both swaging and drawing were tried.

Seven (7) reduction passes were made in a rotary swager and a finished rod of the desired size with the correct density was obtained. However, in the process the stainless steel tube was subjected to an excessive amount of cold work so that longitudinal cracks appeared. Although this could be overcome by in-process annealing, it was decided that overall reduction by drawing would be superior. Drawing trials showed that final size and percent density could be obtained by 11 reductions made in 8° entry angle 1/2 BS gage draw die.

3. Straightening

In order to meet the final requirement that the poison rods (with the polyethylene coating) fit freely in an .090 diameter tube, full length, several approaches were tried. Initially, the tubes were straightened in an offset guide straightener after drawing. This produced poor results due to the stress in the rods during the 11 drawing operations. Neither stress-relieving nor annealing the rods prior to straightening proved successful. Next, an Abramsen rotary straightener was tried. This produced the straightest rod but it still did not meet the requirement. This requirement was eventually achieved by annealing the rods, making an initial pass through the offset guide straightener and a final pass through the rotary straightener.

The offset guide straightener puts enough cold work back in the rods after annealing so that the rotary machine can then be effective.

4. Cut to length

Each tube is cut to length with a diamond cut-off wheel. Note: Each filled and drawn tube makes five (5) poison rods.

5. Weld bottom end

A 1/8" long 308 stainless steel plug is inserted into the bottom end of the tube and set square with the end. The end is then TIG welded without filler metal.

6. Braze head to rod

The head is resistance brazed to the end of the rod using Handy & Harman Easy-Flo Three braze alloy.

7. Coat with Polyethylene

A thin coating (.005 to .008) of polyethylene is applied to the rods. To accomplish this, four methods were initially considered: Spraying, Hot Dip, Fluidizing, Shrink Tube.

Spraying was immediately eliminated due to the small O.D. of the rods. Inserting the rods in molten polyethylene and pulling them through a die was a successful method for coating some 15" long experimental poison rods. However, when longer lengths were tried, separation of the polyethylene occurred and the coating thickness was out of a controllable tolerance.

The fluidize bed method was also tried. The poison rods were too small in mass to retain sufficient heat to provide a uniform, completely melted coating. Manufacturing Engineering working with an outside supplier developed thin wall small diameter shrinkable tubing. This tubing is slipped over the rod and shrunk to it by hot air (approximately 200°F). The heat must be continually applied from one end since this tubing expands in length as it shrinks radially. This proved to be successful.

ORIFICE JET DEVELOPMENT AND MANUFACTURE:

The change of material from aluminum (LASL requirements) to 303 stainless steel necessitated manufacturing development.

Considering size, quantity, precise requirements and specifications, the orifice is one of the most critical parts manufactured (figure 2). Manufacturing specifications are so critical only one vendor has been approved from at least two dozen companies which have attempted to produce this part. It should be pointed out that there are twelve (12) different hole sizes in the orifices ranging from .0225 to .0700. The three (3) smaller sizes are considered deep holes since the length to diameter ratio exceeds 10:1.

In the operations, Swiss Automatic Screw Machines, Petermann variety, sizes No. 4 and No. 7 are used and the developed manufacturing sequence of operations is as follows:

1. Face bar stock. (Centerless ground 303 Stainless Steel)
2. Drill inside diameter per applicable size requirement. Allow .002 stock for final reaming to size.
3. Turn $.135 + .003$ and $.125 \pm .001$ diameters.
- .000
4. Turn $.0925 \pm .0005$ diameter.
5. Cut off and chamfer outside diameter $.005 \times 45^\circ$.
6. Ream inside diameter and face inlet end to required length to help maintain sharp corner requirements.
7. Ultrasonically clean.
8. Place orifices (350) in fixture and lap inlet end to guarantee sharp corner requirement.
9. Ultrasonically clean and final inspect. Parts were randomly selected and sectioned to verify process maintenance of the 32AA surface finish, inside diameter, and the sharp corner requirement of the inlet end.

Operations 1 to 5 are considered precise close tolerance machining operations.

The development work began with the sizing of the inside diameter which has a tolerance of $\pm .0002$. The ballizing method of sizing the inside diameter was attempted, but was extremely slow. The ball would get stuck and the plungers would bend. Each time this occurred a lengthy tear down and set up would be necessary.

Lapping of the inside diameter was also tried in conjunction with the ballizing method in order to improve the surface finish and maintain the correct diametral tolerance. The diametral tolerance of the lapping wire was not precise enough to guarantee maintaining the required tolerance of the inside diameter of the Orifice. Also, the lapping wire continually broke and created serious setup and production problems. This operation along with the ballizing hole size method had to be abandoned in order to meet projected program schedules.

It was then decided to go from a drilled through hole into a reaming operation. Machine speeds and reaming feeds were developed for the required diameters and the stainless steel material. After the reaming operation was successfully developed to meet the drawing requirement, another problem was created by this specific operation. The sharp corner requirement could not be met. The drilling and subsequent reaming operation left a burr around the inlet end of the hole. In order to correct this problem, a light machine cut was taken across the inlet end after reaming. This improved but did not completely resolve the problem so several hundred pieces were fixtured and machine lapped on a wooden block with lapping compound. After lapping, the pieces were thoroughly cleaned and the sharp corner requirement was checked with 100 X magnification and found to meet the drawing requirements of 32AA surface finish on the inside diameter with a sharp corner on the inlet end. The sharp corner requirement on the inlet end was further improved by changing .0005 radius maximum to a .0005 maximum x 45° chamfer at the inlet face. The chamfer offers less opportunity for creation of burr.

Hole concentricity requirement of .001 T. I. R. was also extremely difficult to maintain. Manufacturing Engineering requested that a study be made of the requirement in order to relax the tolerance. This request was approved by Design Engineering as follows:

Relaxation of hole concentricity from .001 T. I. R. to .005 T. I. R. on the six (6) smallest sizes and from .001 T. I. R. to .003 T. I. R. on the six (6) largest sizes. This relaxation reduced the rejection of Orifices from 70% to approximately 20% of production and still met the functional requirement of the part.

TIE RODS - INCONEL X-750:

1. Purpose

A development program was initiated by Manufacturing Engineering for the purpose of defining a suitable fabrication process for the production of tie rods. The greatest obstacle to overcome was forging the stand of material required for the head end and at the same time build in the best grain flow distribution possible, particularly at the junction of head or shoulder and shank where shear loads are greatest (figures 3, 4).

2. Forging Sequence

Material is X-750 - .110 diameter wire.

Starting with the first upset, the material is formed into a cone extending above the die line. Split dies are used to hold the material, built into the dies is the contour to fit the diameter of the wire plus the form for the .125 radius that blends into the shank of the tie rod. The coil used for heating is designed to surround the material and is mounted on a retracting mechanism which, when released automatically, lifts the coil backward, just as the metal has reached the exact forging temperature. Split second timing is required here between press, heating unit and coil before the downward stroke of the punch has started.

The forging process continues in much the same manner gradually reducing the height of the material, maintaining the cone shape through the first fifteen (15) blows. The final forging of the head and the fillet under the head requires a second set of dies plus ten (10) additional forging operations. (figure 5).

Following each hit the forged area is cleaned and polished to remove scale, oxide and all apparent surface defects.

The thread end is forged in very much the same manner except that twelve (12) progressions are used to bump up the materials for the threaded area (figure 6).

Final machining and polishing of the part is required at the completion of the forging in order to meet the drawing dimensions. Beginning with the head end the part is form ground on a horizontal grinder. The shank of the tie rod is gripped in a collet just below the lower radius under the head with the remainder of the tie rod supported by an outboard fixture. The total amount of metal removed is approximately .005".

The thread portion of the tie rod is centerless ground and here again special fixturing is required. The grinding of all surfaces is employed for the purpose of producing the best surface finish that can be acquired prior to roll threading and liquid penetrant inspection. Thread rolling is accomplished using flat dies mounted in a horizontal thread rolling machine. The tie rod is fixtured in a vertical position until the threading is completed and requires one pass only through the dies. The tie rod is then checked dimensionally and for surface indications and if acceptable is ready for heat treatment.

3. Tie Rod Heat Treating

Inconel X-750, the tie rod material, because of its titanium and aluminum content, is most difficult to bright solution treat. The required treatment is heat to 1900°F, hold 30 minutes, and cool below 1000°F in five minutes. The titanium and aluminum at 1900°F act as scavengers of minute traces of oxygen, resulting in a most difficult to pickle scale. Eighteen inch length samples were solution treated in a two chamber vertical vacuum furnace. Samples were clean, bright, and properly solutioned. Unfortunately, the furnace was not long enough to treat full length tie rods. A search was conducted nationwide for vertical vacuum equipment with cooling capacity for treating these parts. No unit meeting the cycling requirements was located. The other alternative was the use of super dry gas atmosphere, namely hydrogen or cracked ammonia purified to a few parts per million by passing through a silver-palladium purifier.

An experimental program was initiated at the Bennett Heat Treating Company, using available facilities, on a best effort basis. Argon and various combinations of argon and hydrogen as atmospheres showed little improvement. Wrapping the individual tie rods in tantalum foil resulted in a bright treatment, but the resulting cost was prohibitive, as the foil became oxidized and brittle on one cycling, and could not be re-used. Purification of the hydrogen and argon in an externally heated generator by flowing these gases through tantalum powder at 1300 - 1500°, resulted in treated tie rods with a decreased film of oxide, and more importantly, an oxide which is readily removed by liquid honing. This process is producing acceptable tie rods.

PROCESS REFINEMENTS:

There are items which require further process refinement. A few examples are:
(figures 7, 8)

1. Deposition and machining of Pyrolytic Graphite.
2. Brazing and fabricating of screening to curved surfaces.
3. Deep hole drilling of various materials including Beryllium.
4. Hot upsetting of stainless steel tubing for instrumentation tie rods.

MANUFACTURING SERVICES ACTIVITIES:

Ordering and Stores

Manufacturing Services analyzed and prepared detail ordering information and issued requisitions for material and services required to meet the program procurement schedule.

The requisitions, work orders, change notices and L-Orders processed totaled in excess of 6500. In addition, variation, approval and disposition requests were analyzed and processed for ordering effect. In excess of 4700 such transactions were completed after determination of effect and applicability on usage and schedule.

Master records are maintained in the department and used for ready reference in processing of new information. A continuous review of cost history, both detail estimates and purchase requisition, is maintained. This information, for example, is used in preparation of budget proposals and program planning.

Inventory control is maintained on a continuing basis. On direction from program management, material is assigned priority and expedited from the time it arrives at the receiving dock through final inspection and issuance from stores.

Store rooms are maintained at WANL and WANEF for receipt and storage of materials, components, and subassemblies. Reactor and non-reactor material are segregated. In addition, the stores are responsible for all records, receiving, cleaning, packaging, storage and issuance of materials. For example, 1743 Quality Control Material releases have been processed and over 9300 stores requisitions handled.

Tool Engineering

Tool Engineering provided technical assistance to Design, Purchasing and Suppliers for special tools, gages, and fixtures associated with the manufacture of detail hardware, test components, assembly and dis-assembly equipment at WANL, WANEF, and NRDS. Detail cost estimates are prepared, and supplier bids evaluated prior to procurement. Vendor surveillance during manufacture and providing accounting with verification of special tooling acquired is on a continuing basis. Over 2000 ordering and approval requests have been processed during the past year.

Government Property Control and Accountability

Procedures for Government Property Control and Accountability were initiated and INSMAT approval obtained. A system has been established of reporting on continuing basis all special tooling and capital equipment items acquired by the NERVA Project.

YEAR END STATUS:

In the past year we have gone through our first phase of procurement. Tooling for reactor assembly and component testing has been designed and procured. Test rigs and

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WANL-TME-573

other special equipment have been completed. Specification grade hardware has been received for reactor assemblies and various component tests such as vibration, mechanical and cryogenic. During the procurement of this hardware, considerable experience was gained by our technical personnel. We are essentially a service group and, while we have no specific milestones to meet, Manufacturing Engineering and Services have a continuous input to all other departments.

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POISON ROD

595894A

ORIFICE JET

595898A

NE-1363

REACTOR ASSEMBLY, PACKAGING, AND SHIPPING

PURPOSE: "To assemble reactors and associated components to the required configuration; to prepare assemblies for shipment; and to transport assemblies to the required test site."

ACCOMPLISHMENTS:

1) Ground was broken for the construction of the Reactor Assembly Laboratory late in September, 1962, and the Laboratory was completed in March, 1963. The installation of the assembly stands and fixtures was completed in June, 1963.

During this period, the Reactor Assembly Laboratory was staffed with personnel having an average of five years' nuclear experience. This includes the supervisors, engineers, technicians, and clerical staff.

The Engineering Mock-up Reactor was used to train the personnel and verify the assembly procedures commonly called RAPO's (Reactor Assembly Process Outlines). RAPO's were prepared during Contract Year 1963 by the Reactor Assembly Engineering Group for the Phase I Vibration Test Core (ND-201) and the NRX-A1 Cold Flow Reactor.

The Reactor Assembly Laboratory's twenty-thousand square feet are located in Buildings 6 and 6A, and are divided into two major areas: the Preassembly Area (Figure 1), which contains orificing, clustering, instrumentation, and cluster storage; and the Final Assembly Area (Figure 2), which contains core stands, inner and outer reflector stands, and a rotating fixture. Figure 3 depicts an orificing operation in the Preassembly Area; and Figure 4, a clustering operation in the Preassembly Area. The ND-101 Engineering Mock-up is shown on one of the two core stands in Figure 5.

With the exception of a few items in the locker rooms, only non-combustible materials of construction were used throughout the Assembly Area. The facility is completely air-conditioned with provisions for humidity control. Cleanliness being one of the important factors in a Reactor Assembly Area, an air-filtration system was

installed that is capable of supplying dust-free air to less than 0.5 microns by means of pre-filters and absolute filters.

Criticality being another important factor in a Reactor Assembly Area, such parameters as concentration and quantity of fissionable material, geometry, moderators, poison, reflector, and effect of interaction were taken into account. In addition, it was assumed that all of the fuel elements would be loaded to 500 mg/cc, and that it would require two unlikely and unrelated accidents to cause a critical mass to be assembled.

The fixtures, stands, platforms, tables, storage containers, etc. that are in intimate contact with any reactor component are made of non-ferrous materials to preclude the possibility of chips of paint adhering to a reactor component.

Sufficient floor area, assembly stands, tooling and fixturing, and storage capacity are available so that two NERVA Reactors can be assembled concurrently and six reactors' worth of fuel can be stored.

2) The Phase I Vibration Test Core (ND-201) also known as the G-I Test (Figure 6) was assembled and delivered to the vibration test stand in the Engineering Mechanics Laboratory on September 14, 1963.

3) A shipping container and special trailer (Figure 7) that will be used to transport NERVA Reactors to the required sites were designed and made available during this contract year.

4) A demonstration run by special train to the MAD Building at NRDS, Nevada, (Figure 8) and return to Large, Pennsylvania, was completed in early July of this year. This shipment was conducted to demonstrate the ability of the container to provide adequate protection for the assembled NERVA Reactors during handling and transportation. The railroad equipment, shipping container, and simulated NRX Reactor were highly instrumented. A report issued by Ground Support Engineering indicated the container and all equipment performed in a satisfactory fashion and that the accelerations to be imposed on a reactor are well within the allowable limits.

5) A demonstration run to the Critical Facility at WANEF and to the Hydrogen

Corrosion Facility at WANHEF to obtain verification of shipping techniques, handling techniques, and fixturing was completed during September, 1963.

CARRY OVER:

- 1) The milestone to complete the assembly and to ship the NRX-A1 Cold Flow Reactor to NRDS will carry over into Contract Year 1964 through the first of November. Presently, the core is completely bundled on the assembly stand and its inner reflector installed.
- 2) A three-week carry over is required to complete the PAX Assembly for NRX-A2.

SUMMARY:

Contract Year 1963 saw the following accomplishments:

- 1) Design, build, and staff a Reactor Assembly Laboratory.
- 2) Design, build, and install the necessary assembly stands, fixtures, and tooling to assemble two NERVA Reactors concurrently.
- 3) Complete the assembly and deliver the Phase I Vibration Test Core (ND-201) for the G-I Test.
- 4) Perform a shipping demonstration of an instrumented container and simulated reactor from Large, Pennsylvania, to the MAD Building at NRDS, Nevada, and return.
- 5) Perform a shipping and handling demonstration from Large, Pennsylvania, to the Critical Facility at WANEF and to the Hydrogen Corrosion Facility at WANHEF, and return.

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WESTINGHOUSE ASTRONUCLEAR LABORATORY

NEGATIVE #43301
RAL-076

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WESTINGHOUSE ASTRONUCLEAR LABORATORY

NEGATIVE #43301
RAL-070

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WESTINGHOUSE ASTRONUCLEAR LABORATORY

NEGATIVE #43301
RAL-078

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WESTINGHOUSE ASTRONUCLEAR LABORATORY

NEGATIVE #43301
RAL-079

WESTINGHOUSE ASTRONUCLEAR LABORATORY

NEGATIVE #43301
RAL-075

WESTINGHOUSE ASTRONUCLEAR LABORATORY

NEGATIVE #43301
RAL-071

WESTINGHOUSE ASTRONUCLEAR LABORATORY

NEGATIVE #43301
RAL-072

WESTINGHOUSE ASTRONUCLEAR LABORATORY

NEGATIVE #43301
RAL-073

NE-2600

RELIABILITY

PURPOSE: "To ensure that the nuclear subsystem design and development program will achieve the required reliability objectives for the design and to ensure that the reliability will not be degraded in manufacturing, shipping and installation"

PROGRAM PLAN

Project 2600 started approved only through March 1963, by which time it was required that a Program Plan be established to describe the reliability program for the remainder of the Contract Year. In line with this, WANL was required to submit a Program Plan for approval by January 31, 1963. WANL-TNR-088, NERVA Nuclear Subsystem Reliability and Quality Assurance Program Plan, was issued on January 31, 1963. It should be noted here that the work on the Quality Control portion of this plan was carried out under Project 2610. Following the initial issue a series of meetings were held with Aerojet and SNPO-C representatives to discuss the plan and resolve differences. A revised plan was written to incorporate the agreements reached. The revision was issued on June 24 and submitted for approval. As of the end of the Contract Year, and as of this date, no approval has been received. It is intended that this Plan be revised and up-dated as necessary so as to completely describe the Reliability and Quality Control programs with the exception of schedules.

FAILURE MODE ANALYSES

A failure mode analysis was made for the NRX-A reactor in January and February. The results of this analysis were issued in a draft form and later in a final form, WANL-TNR-085, to highlight the critical failure modes to assist design and design review efforts. The first slide shows a typical page from the failure mode analysis report. In the columns at the left, each component is identified and the modes of failure for the component are then

listed. The effects of the failure on the engine performance is also shown, and it is noted whether or not the failure could constitute a nuclear hazard. All possible causes of each failure mode are shown as are recommended methods to eliminate or prevent the failure. The next slide shows a failure mode severity index which was used to establish the relative severity or seriousness of each failure mode. In this analysis, ratings were established for the probability of failure for each mode, the effect of failure, and the confidence in the probability. These were combined as shown to obtain a severity index. The higher numbers indicate a more serious condition. The severity index was used to review the component test program to ensure that wherever feasible, the component test program included the most serious of the failure modes.

Failure mode analyses were also made for the Bendix and the General Electric control drum actuators. In addition, a potential failure rate index was applied to the results of the failure mode analysis to compare the potential reliabilities of the two actuators. The factors considered in this analysis can be seen by referring to the next slide. The failure rate index for each part is obtained by a summation of the indices associated with each failure mode cause. The failure rate index is a product of the following items:

- a) The number of failure mode causes
- b) A factor rating the difficulty of removal of the failure mode cause
- c) An environmental severity factor
- d) A factor rating the effect of the failure on performance

The following slide shows the results of such an analysis for the control drum actuators. We have in the left hand columns the failure rate index values and in the right hand columns a comparative index ratio. The columns "Original" and "Improved" refer to the actuators in their original form and in a form in which it was assumed that design changes were made to remove the weak spots. This chart boils down, looking at the number in the lower right hand corner, to the fact that the Bendix actuator was estimated to have a potential failure rate almost twice that of the G. E. actuator. These results were a factor in the selection of the G. E. actuator for further development.

DESIGN REVIEW

A continuing informal design review was conducted throughout the contract year. Reliability engineers reviewed and signed off all engineering drawings. Where necessary, recommended changes to improve the reliability were incorporated prior to signoff. Reliability engineers also reviewed and commented upon drafts of Process and Purchasing Department Specifications. They also reviewed and participated in the approval of all engineering change orders.

A formal design review of the NRX-A2 reactor was conducted by Reliability with members of Engineering, Quality Control, Manufacturing, and Materials participating. For this design review, the comments and recommendations from the informal design reviews and the results of the failure mode analyses serve as a check list. The results of the review and the recommendations were documented and reported to Project Management. Toward the end of the contract year, a formal design review was held on some of the post-test disassembly equipment. Results of these design reviews were also documented and recommendations were incorporated into the design.

Reliability Engineers participated in review board activities conducted to consider the acceptability of discrepant material. This type of effort increased toward the end of the contract year in support of the assembly operations.

PROGRAM AUDITS

A series of audits were conducted in selected areas to determine conformance with the Reliability and Quality Assurance Program Plan requirements. Where deficiencies were found, recommendations were made to the appropriate Management personnel for corrective action. Examples of these audits include the following:

- 1) A check of the assembly area for proper documentation of procedures and instructions for both the Assembly and Quality Control personnel.
- 2) A check of procurement packages for the inclusion of proper Quality Control requirements.

- 3) A detailed check of the accuracy of the drafting checking operation.
- 4) A check of the adequacy of test result records from the component test program.

Toward the end of the contract year, efforts were started on the outlining of a formal audit program to define all functions to be audited and the characteristics to be checked under each function.

DATA SYSTEM

A considerable effort was devoted to establishing elements of the Reliability and Quality Data System to provide historical data as well as data for analysis and recommended corrective action. A list of the data system reports developed during the contract year is included in the pass-outs which will be supplied to you. In addition, assistance was given to Astrofuel in developing process control reports. This work was actually done under Astrofuel Project No. 1350. An end item configuration log was prepared for the KIWI-B5A Shield. This report, WANL-DSR-001, was issued to serve as a data link between the shield and the Quality Control records. It allows the history of any shield component to be traced back to the source for use in post-test analysis. This is typical of the end item configuration logs which will be used for the reactors.

A failure trouble reporting system was developed and added to the Reliability and Quality Assurance Program Plan.

Monthly Reliability and Quality Status Reports were issued starting with April. These reports are prepared from inputs by both Reliability and Quality Control.

Close Reliability monitoring of the actuator test program was provided. Reliability monitoring of the component test program and the review of the test plans was initiated during the second half of the contract year. The plans and the hardware for the reactor shipping test were reviewed by Reliability with a number of changes being made to improve the reliability of the test.

SUMMARY:

The seven milestones scheduled for this project were all completed by the end of the contract year. The Reliability support effort required for the NRX-A1 assembly continues, however. This is the effort required for reliability participation in design changes proposed during NRX-A1 assembly and for the Reliability participation in Engineering Review Board activities on discrepant material.

FAILURE MODE ANALYSIS SUMMARY						PAGE <u>1</u> of <u>54</u> PAGES			
NERVA NRX-A1 REACTOR						DATE <u>March 15, 1963</u> Rev. <u>-</u>			
PART NAME	PLANNING PARTS LIST NUMBER PREFIX 1.2.1	MODE OF FAILURE	HAZARD	EFFECT ON THRUST*	CAUSE OF FAILURE	RECOMMENDED METHODS TO ELIMINATE OR PREVENT FAILURE		DISPOSITION	
						TESTING			DESIGN
						MFG.	DEV.		
PLATE, Core Support (934J705)	.01.00.01	Circumferential break at support flange	X	1	Vibration	-	E	Design analysis to determine extent of sag due to non-linear thermal expansion.	
					Thermal shock	-	E		
					Defect	QC	-		
		Break at fuel cluster support hole	X	2	Vibration	-	E		
					Defect	QC	-		
		Sag	X	2, 3	ΔT	-	E		
					Pressure	-	E		
		Support flange breakout	X	2, 3	Vibration	-	E		
					Defect	QC	-		
		Break at dome seal flange	X	2, 3	Vibration	-	E		
			Defect	QC	-				
PIN, Core Support Plate (963B405)	.01.00.02	Transverse break	-	None	Vibration	-	E		
					Thermal shock	-	E		
					Defect	QC	-		
NUT, Fuel Cluster (945C463, 945C464, 945C478)	.00.00.03, .00.00.04, .00.00.05	Shear at threads	X	2, 3	Vibration	-	E	Proper assembly procedure is required.	
					Thermal shock	-	E		
					Excessive loads	-	E		
					Overtorque	QC	-		
					ΔT	-	E		
					Defect	QC	-		
		Break at shoulder	X	2, 3	Vibration	-	E	Design analysis.	
					Thermal shock	-	E		
					Excessive loads	-	E		
					Overtorque	QC	-		
					Defect	QC	-		

* EFFECT ON THRUST

1. IMMEDIATE DESTRUCTION
2. EVENTUAL DESTRUCTION
3. DECREASE MORE THAN 10%
4. INCREASE
5. DECREASE LESS THAN 10%

FAILURE MODE SEVERITY INDEX NERVA NRX-A1						DATE <u>3-12-63</u> PAGE <u>1</u> OF <u>37</u>
PART NAME	PLANNING PARTS LIST NUMBER	MODE OF FAILURE	PROBABILITY OF FAILURE P	EFFECT OF FAILURE E	DESIGN CONFIDENCE C	SEVERITY INDEX $S_i = (P)(E) + C$
PLATE, Core Support (934J705)	1.2.1.01.00.01	Circumferential break at support flange	1	10	0	10
		Break at fuel cluster support hole	1	5	0	5
		Sag	3	5	0	15
		Support flange breakout	1	5	0	5
		Break at dome seal secure flange	1	5	0	5
PIN, Core Support Plate (963B405)	1.2.1.01.00.02	Transverse break	1	0	2	2
NUT, Fuel Cluster (945C463, 945C464, 945C478)	1.2.1.00.00.03, 1.2.1.00.00.04, 1.2.1.00.00.05	Shear at threads	3	10	4	34
		Break at shoulder	1	5	2	7
DEVICE, Nut Locking (944C562)	1.2.1.00.00.07	Transverse break at tang	3	5	4	19
		Fuel cluster nut rotates and unlocks	1	5	4	9

SYSTEM: _____

SUBSYSTEM: _____

DATE: _____

WANL-FORM NO. 1074

REVISED FAILURE RATE INDEX SUMMARY				The following summary reflects the potential decreases in "Failure Rate Indices" as stated in WANL-TMI-445 and compares them with the original "Failure Rate Indices" presented in the same TMI.				
Control Drum Actuators Bendix Model ND-D1, P/N 2150750 General Electric Model AG-14, P/N 1076510-761								
Component or Section	Failure Rate Index				Comparative Index Ratio			
	G.E.		BENDIX		G.E.		BENDIX	
	Original	Improved	Original	Improved	Original	Improved	Original	Improved
1. Torque Motor Assembly	5.8	5.8	6.6	5.4		1.07	1.14	
2. Servo Valve Assembly	9.2	5.2	11.3	7.5			1.23	1.52
3. Power Section	87.7	62.3	61.0	58.7	1.44	1.06		
4. Transmission & Output Shaft	28.9	24.2	99.7	78.9			3.45	3.26
5. Feedback Section	7.2	7.2	33.3	26.4			4.62	3.67
6. Electrical Recept. & Wiring	9.1	9.1	18.5	17.3			2.03	1.90
7. Cooling Gas Circuit	3.0	2.1	4.4	0.1		21.0	1.47	
8. Supply Gas Circuit	0.02	0.02	2.68	0.92			134.0	46.0
9. Housing, Covers & Seals	3.3	3.3	2.1	2.1	1.57	1.57		
10. Mounting Face Clamp & Seals	1.1	1.1	1.2	1.2			1.09	1.09
11. Safety Lock	~	~	8.9	2.2	~	~	~	~
12. Scram Solenoid	~	~	2.3	1.2	~	~	~	~
13. Miscellaneous Hardware	~	~	14.4	0.0	~	~	~	~
	155.32	120.32	266.28	201.92			1.71	1.68
Excluding items 11, 12 & 13, which are unique to the BENDIX actuator, the actuators compare thusly:	155.32	120.32	240.68	198.52			1.55	1.65
Considering Item #3, Power Section, for the G.E. Actuator, a reduction in the number of piston rings from 8 to 4 will change the numbers as follows: - Item #3	87.7	44.5	61.0	58.7	1.44			1.32
- Total of Items #1 thru #10	155.32	102.52	240.68	198.52			1.55	1.94

NE-2610

QUALITY CONTROL

PURPOSE: "To provide for the inspection and control on nuclear subsystem components and assemblies; to ensure conformance to all design specifications."

ORGANIZATION

The organization of the Quality Control Department can best be explained by means of an organization chart (Figure 2610-1). The Quality Control Department consists of five sections reporting directly to the Quality Control Manager, who, in turn reports directly to the NERVA Program Manager. The Quality Control Manager is responsible for reporting to project management on the adequacy and status of the overall Quality Control Program. He is responsible for the conduct of the program and for recommending changes or improvements to the Quality Control Program to ensure conformance to design specifications.

Although the Astrofuel and Nevada Test Operations Quality Control Sections are separate entities, there is close coordination between them and the WANL Quality Control Department to ensure that the overall quality objectives of the program are met.

A detailed explanation of the responsibilities and duties of the five sections within the Quality Control Department will be given later.

The increase in activities in the Quality Control Department is shown by this graph showing the build-up of Quality Control Manpower during CY'63 (Figures 2610-2 and 2610-3). In October 1962, this department consisted of a workforce of 14 persons. The Receiving Inspection area was non-existent. The Supplier Surveillance Section was only beginning to make initial surveys of potential suppliers. The Quality Engineering Section was formulating the basic procedures to be used to ensure complete quality control. As the workload increased and facilities were completed, additional

manpower was needed. In June, the Nondestructive Testing and Metallurgical Release Section was broken out of the Inspection Section to provide the highly specialized services essential to the proper performance of this work. A separate Reactor Assembly and Critical Experiment Section was organized in September.

By the end of September, the manpower had increased from the original 14 persons working on a one-shift basis to 106 persons providing control on a 24-hour-a-day basis, seven-days-a-week basis.

QUALITY ENGINEERING SECTION

A. Basic Responsibilities

The basic responsibilities of the Quality Engineering Section are shown below:

1. Establishing and maintaining the WANL Quality Control Manual.
2. Reviewing and recommending changes to WANL specifications.
3. Reviewing and approving or disapproving supplier and WANL special manufacturing procedures.
4. Preparing, coordinating, and issuing inspection instructions, for the verification of quality characteristics at receiving, in-process, and final inspections.
5. Defining, approving the design, and ordering special inspection tools, fixtures, and gages required for the verification of necessary quality requirements.
6. Coordinating, controlling, and documenting all Engineering Review Board and Materials Review Board actions when required for NERVA hardware.
7. Initiating and coordinating corrective action for internal WANL quality problems.
8. Establishing and maintaining the WANL statistical sampling program.
9. Coordinating the resolution of quality problems with AGC and SNPO representatives.

10. Analyzing supplier-originated Variation Requests and QANL Field Trip Reports.
11. Analyzing WANL originated Variation Requests and receiving, in-process, and final inspection records for recurrent quality problems, and coordinating corrective action efforts with other WANL activities.
12. Reviewing drawings, specifications, requisitions, purchase orders, and other documentation to ensure implementation of Quality Control requirements.
13. Performing periodic audits of Supplier Surveillance activities.
14. Developing and maintaining a training program to increase the proficiency of Quality Engineering personnel.
15. Preparing and issuing Procurement Quality Requirements defining necessary quality requirements to be included in procurement actions.

B. Summary of Accomplishments

The majority of the responsibilities of the Quality Engineering Section are accomplished by initiating, reviewing, processing or approving various forms of documentation. The following summarizes the accomplishments in this area:

Prepared 1,900 Procurement Quality Requirements (This includes only the original issues and does not include revisions or re-issues.)

Reviewed 4,700 Purchase Orders and /or Purchase Requisitions

Reviewed 2,900 Changes to Purchase Orders and/or Purchase Requisitions

Contributed to and approved 70 Reactor Assembly Process Outlines

Prepared 20 Quality Control Operating Instructions

Approved 950 Approval Requests

Approved 1,100 Variation Requests

Reviewed 65 Purchasing and Process Specifications

Prepared 275 Inspection Instructions

Produced 75 Tooling Fixtures and Gage Designs

Reviewed 300 Engineering Change Requests or Engineering Change Orders

Approved 510 Nonconforming Material Disposition Requests

During Contract Year 1963, the ND20101 reactor was completed and assembly was in process on the ND10101 and ND207 reactors. Procedures were established and implemented to ensure all necessary quality requirements were maintained during the assembly of these reactors. Quality Control Operating Instructions, Quality Control Inspection Instructions, Reactor Assembly Process Outlines and other established methods of quality control were utilized.

SUPPLIER SURVEILLANCE SECTION

A. Basic Responsibilities

The Supplier Surveillance Section has been established to ensure that material and parts manufactured by suppliers meet all the design requirements. To accomplish this goal, the following basic responsibilities have been assigned to this section:

1. Conduct surveys of prospective suppliers to evaluate the capabilities of their Quality Control System. Make specific recommendations concerning the approval or disapproval of prospective suppliers based upon survey evaluations.
2. Assist Quality Engineering in designating meaningful Procurement Quality Requirements.
3. Advise the supplier of his responsibilities as defined in WANL Quality Control Specification No. 101, "Quality Requirements for Suppliers"
4. Assist the suppliers in interpretation of specifications, drawings, and WANL quality requirements, and provide assistance in the solution of quality problems which arise at the supplier's facility.

5. Verify the information and adequacy of documentation on Variation Requests and Approval Requests originated by a supplier and recommend dispositions to Quality Engineering.
6. Provide source inspections on NERVA parts and material as directed by the Purchase Order.
7. Direct the supplier corrective action program including the initiation of Requests for Corrective Action and follow-up action.
8. Periodically issue WANL Quality Control Supplier Surveillance Reports detailing the supplier's overall program status and describing existent or potential trouble areas.
9. In coordination with the WANL Purchasing Department, issue work stoppage orders to suppliers because of serious quality problems.

B. Summary of Accomplishments

During Contract Year 1963, the Supplier Surveillance Section participated in the initial evaluation of approximately 140 suppliers. As a result of these evaluations, 39 suppliers were conditionally approved, 15 were disapproved, and the remainder were approved as suppliers for parts or materials.

During these initial surveys, the following areas were investigated and evaluated by Supplier Surveillance representatives:

1. Condition of manufacturing facilities and equipment
2. Quality Control organization and procedures
3. Receiving, in-process, and final inspection practices
4. Inspection and test equipment
5. Tool and gage control
6. Engineering organization (if applicable)
7. Quality Control Records
8. Past performance on similar jobs - workmanship

The number of Supplier Surveillance source inspections continually increased during Contract Year 1963. A total of approximately 500 source inspections were performed during the year with 104 inspections being performed in August. The following is a summary of the most important areas covered during these inspections:

1. Review Purchase Orders and Change Notices
2. Review for Compliance to Approval Requests
3. Review for Compliance to Variation Requests
4. Review the accuracy and completeness of material releases, test reports, certifications, and special test results.
5. Inspection records, sampling plans, evidence of inspection, and evidence of the completion of necessary rework reviewed on dimensional and visual inspections.
6. Review Quality Certificates and Shipping Releases to ensure all necessary portions are completed accurately.
7. Make shipping inspections to check shipping containers and to check for cleanliness and damage of parts.

Thirty-three Requests for Corrective Action were initiated during CY'63. The following is a sample of supplier deficiencies that resulted in the issuance of this form:

1. Shipping defective parts which may need to be scrapped because of supplier's negligence.
2. Material or parts not cleaned properly.
3. Material received without proper documentation.
4. Work being performed prior to WANL approval of necessary manufacturing or inspection procedures.
5. Material being used by a supplier prior to the receipt of a WANL Quality Control Material Release.
6. Necessary inspection operations not being performed.

7. Review supplier certifications to verify that quality requirements have been met.

B. Summary of Accomplishments

I. Radiographic Inspection

During Contract Year 1963, the WANL radiographic laboratory was made functional. Construction of the 17 ft. x 19 ft. exposure room shown in Figure 2610-4 was started in November, 1962, and work on this project was completed in January, 1963. A 300 KV X-ray unit and dark room facilities were installed in February and radiographic work commenced on March 15, 1963. Figure 2610-5 shows the X-ray facility in operation on the evaluation of incoming raw material. A total of 580 square feet of area is presently being utilized for radiographic inspection.

The volume of X-ray work necessitated the installation of increases film processing capacity. A Carr Master Tank was installed in June, 1963. A portable low KV X-ray unit to be used for thin graphite and/or beryllium parts was placed into operation in August.

Because of the increasing volume of radiographic work, it was necessary to revamp the record system in this area. Consequently, an improved system was put into operation in July which covered serialization and history of parts on each radiographic film, the identification of the film itself, and the packaging and filing of the radiographic film.

During the Contract Year, the following technical progress was accomplished in the radiographic area:

Effort was expended to improve radiographic techniques for beryllium and graphite parts. WANL Quality Control NDT personnel made a number of trips to beryllium suppliers to evaluate their prevailing

techniques. Recommendations were made concerning the type of film and masking being used, spacing of film to object, and energy levels being employed.

A study was made on the effect of various grades of extruded and molded graphite on radiographic inspection capabilities. A program has been initiated to define the optimum techniques for the greatest sensitivity (resolution) in three thicknesses of several grades of graphite. The results of these studies will be used to determine the optimum techniques to be used when radiographically inspecting graphite.

The following is a summary of the work output for the last two months of the Contract Year showing only the work completed within WANL:

	1963	
	<u>August</u>	<u>September</u>
NERVA Parts Released	1,248 film	1,754 film
Experimental Work	227 film	230 film
Interpretation of supplier's film	<u>480 film</u>	<u>1,541 film</u>
Total	1,955 film	3,525 film

In August, the Quality Engineering Procedures on qualification and certification of X-ray inspection personnel were approved. Training courses were prepared by the appointed WANL NDT examiner, and the permanent X-ray laboratory staff was certified.

2. Liquid Penetrant Inspection

The installation of the penetrant inspection line shown in Figure 2610-6 was completed in April, 1963.

The tanks were filled with the dye penetrant materials conforming to MIL-STD-271B, Type II. Additional quantities of fluorescent penetrant

inspection materials conforming to MIL-STD-271B, Types III and IV, were provided for special inspections. In September, a decision was made to convert the dye penetrant line to a fluorescent inspection system. This revision will standardize routine inspection to Types III and IV with limited capability for inspection to Type II.

3. Ultrasonic and Vidigage Inspection

Two ultrasonic units, one a narrow band receiver and one a wide band receiver, were placed in operation in February, 1963. Several technicians attended the manufacturer's training school to receive instructions on the operation and maintenance of this equipment. Considerable development work is scheduled in this area to develop procedures for the inspection of tie rod material, graphite, thin aluminum, stainless steel and beryllium. Figure 2610-7 shows an ultrasonic unit in use on a comparison standard.

A Curtiss Wright Model SR-153-3 manual scanning bridge and instrument platform has been ordered for use in immersion testing. A Vidigage was placed in operation in January, 1963 for use in making thickness measurements.

4. Eddy Current Inspection

The design and manufacture of the basic electronic and transport equipment for use in eddy current inspections on assembled cores was initiated during Contract Year 1963.

5. The Metallurgical Release activity was initiated as a separate function in August, 1963. Prior to this date, the function was being performed by Inspection personnel. Because of the specialized technical information contained in certifications and test results, it was decided a more thorough and efficient study could be made by persons with a metallurgical background.

The Metallurgical Release Section maintains a file of certifications and test reports applicable to the material release. Any deviations in material requirements provides a basis for material rejection and subsequent review by an Engineering Review Board.

The Metallurgical Release Section also provided a constant check on the suppliers report test results and on materials shipped by initiating periodic destructive tests. In addition to the above responsibilities, this group controlled Request for NDT Testing instructions, distribution and replies.

RECEIVING AND IN-PROCESS INSPECTION

A. Basic Responsibilities

The Receiving and In-Process Inspection Section provides for all WANL inspection activities other than those performed in Reactor Assembly or on Critical Experiments conducted at the Waltz Mill site. The following are the basic responsibilities of this Section:

1. Verify that received material documentation references the applicable purchase order number. Ensure that correct quantities are received and material has not been damaged in transit.
2. Perform receiving and in-process inspections per established inspection instructions.
3. Maintain records of all data submitted by WANL suppliers. Prepare, maintain and distribute records of all inspections performed. Prepare and distribute necessary documentation indicating acceptance or rejection of material, parts or assemblies.
4. Provide Quality Status Identification of all material, parts and assemblies through the use of stamps, tags, or similar devices.
5. Maintain the Quality Control Mechanical Standards Laboratory which performs initial and periodic calibration of gages, and inspection tools.

6. Order standard gages and inspection equipment, and maintain all Quality Control tools, gages, jigs and fixtures required to verify the compliance of parts and material to quality requirements.
7. Inspect, store, issue and recall manufacturing tools, gages, jigs and fixtures which directly determine the conformance of NERVA hardware.
8. Report any condition or practice observed in WANL manufacturing or assembly areas which could in any way jeopardize the quality of a WANL product.
9. Periodically audit the measuring and test equipment in the inspection, manufacturing and assembly areas to assure that the equipment is within current calibration period.
10. Assist in source inspections at supplier's facilities when necessary.
11. Assist Quality Engineering in the preparation of detailed inspection instructions.

B. Summary of Accomplishments

At the beginning of the Contract Year, the Inspection Section consisted of only 4 persons. The Receiving Inspection area was not as yet in existence. Only a bare minimum of inspection tools and gages were on hand and operating procedures were in the formulative stage.

The present Receiving Inspection area, consisting of approximately 3,000 square feet, was ready for occupancy in February, 1963. During the next few months, large quantities of inspection equipment were ordered, received, and placed in operation. A total of \$60,500 worth of special inspection equipment was procured with NERVA project funds.

A Gage Laboratory approximately 400 square feet in area was established for the calibration of tools and gages. Facilities were installed to maintain a constant temperature of $68^{\circ} \pm 2^{\circ}$, a maximum relative humidity of 50%,

and a dust-free atmosphere in the Gage Laboratory. Upon receipt of all inspection tools, gages, and fixtures, cognizant Gage Laboratory personnel conducted an initial inspection and calibration to determine the accuracy of each item of inspection equipment. All measured standards used in calibrating equipment was certified and directly traceable to the National Bureau of Standards. A Quality Control Gage Record was completed and placed in the Tool and Gage Inventory Files for all inspection tools, gages, and special equipment. All such equipment was assigned and marked with an identification number.

A procedure was established and implemented detailing the frequency of calibration of inspection tools, gages, and special equipment and establishing methods of maintaining records of such calibrations.

Incoming NERVA hardware was inspected in the Receiving Inspection area to ensure compliance with necessary quality requirements prior to acceptance from the supplier. A sample of one of the larger parts, as shown in Figure 2610-8, is the Outer Reflector, Nozzle End, Support Ring mounted on the 48" Rotary Face-Plate Table. At the other extreme in size are the jet orifices shown in Figure 2610-9. The Toolmakers Microscope and Flow Tester used to inspect these small orifices is shown in Figure 2610-10.

The following is a summary of the number of parts received and inspected during the last six months of CY' 63:

<u>Month</u>	<u>Parts Received and Inspected*</u> <u>(No. of Pieces)</u>
April	30,113
May	47,872
June	82,620
July	66,060
August	81,744
September	114,478

*Includes entire lot on sample plan inspections.

A summary report of Receiving Inspection activities was initiated using a monthly IBM runoff. Included in this report are the part number, part name, quantity received, quantity inspected, quantity rejected, Receiving Inspection Report number, supplier, purchase order number, applicable Variation Requests and Nonconforming Material Disposition Requests, and date of inspection. On parts found nonconforming, the nonconforming characteristics and quantity rejected are included.

Quality Control Inspectors performed the Pre-Operational inspection defined in the Test Planning Program for the NDI0101 reactor. This consisted of detailed inspections of specific parts. After the reactor has been tested, these same parts will be reinspected to determine the effect of the test on the parts.

In-Process inspections were conducted on parts manufactured at WANL and on component testing and other experimental activities. Quality Control surveillance was maintained in such areas as the Machine Shop (Model Shop), Materials Development Laboratory, Flow Test and Analysis and High Temperature Laboratory, Engineering Mechanics Laboratory and Instrument Calibration Laboratory.

REACTOR ASSEMBLY AND CRITICAL EXPERIMENT INSPECTION

The Reactor Assembly and Critical Experiment Inspection Section was established as a separate activity in September, 1963. The primary responsibility of this group is to exercise necessary quality surveillance in the Reactor Assembly area and at WANEF (Waltz Mills, Pa.).

Upon release from Stores, all reactor hardware is under continuous Quality Control monitoring during all phases of assembly, instrumentatin, and test. The Inspection operations are planned for each assembly and integrated into a Reactor Assembly Process Outline (RAPO). In the event that the Inspection operations are complex,

they are referenced in the RAPO and defined by a Quality Control Operating Instruction (QCOI). The Quality Control personnel ensure that only qualified parts are used in the reactor and that the instructions in the RAPO and QCOI are followed.

During Contract Year 1963, Quality Control was exercised during the complete assembly of the ND20101 reactor and during the assembly of the ND10101 and ND207 reactors to their present state. The ND20101 was used to "de-bug" our initial inspection procedures.

SUMMARY

The milestones listed in our Work Program for CY'63 have been met.

FIGURE 2610-1
WANL QUALITY CONTROL ORGANIZATION CHART

FIGURE 2610-2
BUILD-UP OF QUALITY CONTROL MANPOWER
C Y '63
(TOTAL MANPOWER)

562320A

FIGURE 2610 - 3
BUILD-UP OF QUALITY CONTROL MANPOWER
CY '63

WANL
X-RAY FACILITY LAYOUT
FIGURE 2610-4

NE-I340

NRX REACTOR CONTROL SYSTEM DESIGN

PURPOSE:

"To design and develop the reactor control systems for NRX-A1 and A2 reactor tests in Test Cell 'A'. Design and develop system components meeting the functional and reliability objectives of the NRX Control System".

Based on the work done in CY 62 and an evaluation of the future NRX testing requirements, work was continued in the development of a NRX control system supplement to be incorporated in Test Cell "A" for use during the NRX Test Program. In November, 1962, (Drawing 574F012) W Block Diagram - NRX Reactor Control System was issued completing the preliminary NRX-A Control equipment design. An updated version of this block diagram is shown in Figure 1. This design incorporated the following capabilities:

- a. Manual gang control of drums.
- b. Manual control of individual drums.
- c. Automatic startup from source level to full power operation.
- d. Flexibility to test advanced reactor control system concepts.
- e. Operation with present KIWI control system.

In December of 1962 the detailed design of NRX systems and components was begun. At this point reliability studies of the system and its major components were initiated, and were continued throughout the design fabrication and testing stages. Concurrent with the design, as sufficient information became available, parts were ordered so that fabrication could begin upon the completion of the design. Fabrication of portions of the system was initiated in April, using preliminary drawings. Final sign-off of all drawings was accomplished in June, completing the design phase. By July, 1963, the WANL designed NRX control equipment was complete and the Acceptance Tests, Quality Control and Reliability inspections were initiated. Upon satisfactory completion of the Quality Control and Reliability Acceptance Tests, the NRX control equipment was shipped to NRDS on August 9, 1963.

Figure 2 shows these equipment racks mounted for shipment to NRDS. Of particular interest is the controller/sequencer which is in effect a group of computing amplifiers and function generators and associated switching and logic devices which provides the flexibility to handle a large range of NRX test requirement. Various patch panels can be provided to give the required control and sequencing of the several NRX tests as needed.

Because of the need to perform drum worth measurements and other low power physics experiments, individual drum control has been provided as part of NRX-A control. The controller for this is shown in Figure 3. This equipment was designed to permit independent control of the position of selected drums for control drum worth tests as required.

The design of the NRX Reactor Control System required the design of system components to meet the functional and reliability objectives of the NRX control systems. The majority of the components were available as standard items, however there were some components which were fabricated at WANL. Figure 4 shows these components along with the commercially procured Donner computing amplifier.

These equipments were subjected to Receiving Acceptance Tests at NRDS and performed satisfactorily. The individual drum positioner was installed in August. The complete description of this system and its operation is contained in TME 530 and 531.

NE-1342

DESIGN AND DEVELOPMENT CONTROL ACTUATORS

PURPOSE: "To develop rotary control drum actuators capable of meeting the functional safety and reliability requirements of the NERVA engine while operating in all phases of the flight environment."

This program had five major objectives for CY 63.

1. Selection of the basic electro-pneumatic actuator design configuration which is to be used on the NERVA engine.
2. Development of an improved design (second generation) of the selected electro-pneumatic actuator.
3. Initiation of the fabrication of a second generation electro-pneumatic actuator.
4. Testing and evaluation of components of the selected electro-pneumatic actuator design.
 - a. Environmental tests of actuator components to prove their performance under various temperatures .
 - b. Testing of actuator components in a combined cryogenic and radiation environment.
5. Provide specifications and follow the procurement and testing of shielded hydraulic actuators for the NRX tests.

Significant engineering effort was also entailed in the design, procurement, fabrication, installation, and operation of special test equipment needed to accomplish these objectives.

In October, 1962, models of two basic design concepts were available for comparative evaluation. These actuators were built to WANL Specification CD-147. In October the actuators were installed and operated with ambient temperature gas and simulated loads. The testing was performed using helium for the drive gas and nitrogen for coolant flow. Cryogenic temperature tests were conducted in November and December to show the effect of temperature on actuator performance. During this period both actuators were adjusted and modified to obtain performance which was in accordance with the specification.

Figure 5 shows the equipment used to test both actuators under ambient conditions with the actuator test stand at the right side of the photograph. Figure 6 shows the cryogenic facility partially installed, and Figure 7 shows the top of the cryogenic test chamber with the simulated drum load being lowered into it. Figure 8 shows the Bendix actuator undergoing warm gas testing. The amplifier, controls and power supplies are rack mounted alongside the test stand. Figure 9 shows a typical frequency response for the GE actuator at ambient conditions, with and without coolant gas. The performance of the actuator system is only slightly affected by the presence of coolant gas. Figure 10 shows the GE actuator mounted to the adapter fitting used in testing. Figure 11 shows the Bendix actuator during cryogenic tests.

After the ambient and cryogenic evaluations were complete, vibration testing of both actuators under ambient conditions was undertaken. The shaker used was an MB 2500 lb table, and the controls permitted automatic frequency sweep and table acceleration control. Figure 12 shows the GE actuator arranged for both axial and transverse vibration. This particular photograph shows the load fixture used to enable actuator operation during vibration. Figure 13 shows the performance of the Bendix actuator in the vicinity of 200 to 400 cps. The command signal is a $3^\circ/\text{sec}$ range with a peak to peak amplitude of 18° . The vibration excitation is 5 g's . Tests like this were run over a frequency range of from 20 to 2000 cps. A similar test result is shown for the GE actuator on Figure 14, while Figure 15 is the step response of the GE actuator in the vicinity of a vibration frequency of 270 cps (one of its resonant frequencies). The performance of the actuators was not detrimentally affected by any of the vibration tests. However, an electrical connection failed in the Bendix test causing loss of control and random oscillation.

As a result of the tests conducted both at ambient and cryogenic temperatures on both actuators and also as a result of the comparative reliability evaluations, it was decided to concentrate further development effort on the General Electric design.

Redesign was started in May of 1963 to meet a new actuator specification E-675378. Six second generation models will be procured as a result of the current contract with GE. A design review was held in August of 1963 at which time the basic design approach was approved. As a result of the new specification requirements and changes in the allowable space envelope, the second generation actuators will look somewhat different from the first generation actuators. Figure 16 shows two views of a wood mockup of this actuator. Very evident is the lock which was not a part of the first generation actuator. This new design is shorter across the piston axis but is of greater dimension in overall height. As of the end of CY 63 the main housing forgings have been procured and initial machining begun at the vendor's plant. Most of the other component parts are on order with lower tier suppliers.

During CY 63 non-radiation and radiation tests were conducted on component parts and subassemblies of the first generation actuators. Non-radiation tests were also initiated in this period on selected second generation parts.

Ambient tests to measure torque motor characteristics have been conducted and analyzed. Characteristics such as spring gradient, static gain, and frequency response were examined. The overall characteristics of the torque motor servo valve combination were also examined at both ambient and cryogenic temperatures. Parameters affecting the performance have been evaluated when subjected to pneumatic load and driving point conditions as expected in this actuator system. Figure 17 shows the torque motor servo valve assembly prior to test at cryogenic temperatures.

Special fixtures were designed and tests were conducted to evaluate the error and dead band characteristics of the piston seals. The seal material was tested at ambient and cryogenic temperatures, with both small amplitude oscillation and with full stroke cyclic runs. The feedback potentiometer was also evaluated at ambient and cryogenic temperatures for linearity, resolution and noise.

The drive assembly consisting of the rack pinion, output shaft, bearings, backup bearings, and piston seals was tested with measurements taken of the drag torque levels at both ambient and cryogenic temperatures.

As a means of evaluating the performance of the actuator components in a combined environment of radiation and low temperatures, tests were performed at General Dynamics, Forth Worth, Texas. To date two 100-hour runs have been made.

In the first 100 hour test the torque motor, piston seal, connector, and potentiometer were run. Figure 18 shows these assemblies mounted to a cryostat cover ready for the irradiation test.

In the second 100 hours at GTR, the servo valve and torque motor combination, drive assembly, and bearing tests were run. Figure 19 shows the assemblies prior to insertion into GTR. The dynamic piston assembly with its load springs are evident at the bottom of Figure 19.

At the conclusion of the tests the components had been maintained at approximately 170°R and had been subjected to the total dose of 10^8 Rads. In each test the component was loaded and driven as expected in the actuator, and the measured parameters were those expected to be the most influencing on performance. For example, the torque motor was loaded by a spring equivalent to the nozzle force and a bias spring was used to place it in its operating range. Then it was driven with a voltage drive identical to that which would be supplied by the actuator amplifier. The torque motor voltage and current were recorded along with the output displacement.

NE-1345

DEVELOPMENT ENGINE REACTOR CONTROL SYSTEMS DESIGN

PURPOSE: "To design and develop the reactor control system for the Block 1 engine leading toward the reactor control system for the flight engine; to design and develop system components meeting the functional, safety and reliability requirements for ground and flight environment."

Prior to the beginning of CY 63 several reactor control system concepts had been investigated. Early in the year a system was adopted and a design completed. The operating cycle of this control scheme was discussed in an external memorandum, WANL-TME-188, "Operation of the Reference Reactor Control System." A system schematic was prepared and released as WANL Drawing No. 936J100. This is shown as Figure 20.

The Reference Reactor Control System was simulated on an analog computer by Systems Analysis so that its operation could be observed while controlling the simulated reactor in a complete engine simulation. The results of this computer study were reported in WANL-TME-270. In general, the Reference Reactor Control System was found to be a feasible method of control, and functioned properly in maintaining the proper relationships among the reactor system parameters. Various gain and compensation adjustments, which were anticipated in the initial design of the Reference Reactor Control System, were determined using the analog computer simulation.

Upon establishing the Reference Reactor Control System as a feasible method of control which would meet the requirements, as specified in WANL-CD-450, "Specification for the Reactor Control System for the Hot Bleed Nuclear Engine," a breadboard of the controller was built. The breadboard, Figure 21, designated Reactor Controller Breadboard II was used to confirm:

- a. That the controller built in accordance with the control cycle of the Reference Reactor Control System would operate satisfactorily outside the clean environment of the analog computer.

- b. That the chosen control cycle would operate satisfactorily with standard solid state amplifiers of less sophisticated design than in the analog computer.
- c. That transients encountered in the system during the shift from the sub-power range of operation into the power range could be controlled or tolerated.

After the results of the tests and analyses of the Reference Reactor Control System and a Reactor Controller Breadboard II were completed, work was commenced on an advanced version to be known as Reactor Controller Breadboard III as shown in Figure 22. This controller is to operate at a reduced computing voltage and with amplifiers which were not stabilized.

Specifically, the information to be studied using Reactor Controller Breadboard III is as follows:

- a. Can the accuracy of the control system be maintained using a computing voltage of + to - ten volts.
- b. Will a controller operate satisfactorily, without problems due to amplifier drift if the amplifiers are not chopper stabilized.
- c. Can a controller for this system be built completely solid state without relays.
- d. Will manufacturing tolerances of individual components, especially diodes, adversely effect the operation of the controller.

Reactor Controller Breadboard III was designed and constructed, and various open loop tests were made on the circuitry and components. The results of this preliminary testing indicate that the ability of Reactor Controller Breadboard III to control may be comparable to that of the previous controller. A much more extensive test is scheduled

using the reactor and engine simulation on the analog computer. Preparation for this test was underway at the end of September, 1963.

To meet the environment requirements, a series of fifteen preliminary E-Specs (675460 to 675475) were issued covering special designs of resistors, capacitors, transistors, and silicon diodes which will be used in the design of amplifiers and switching circuits.

A detailed survey of the mechanisms of failure in transistors indicates that there should be a significant correlation between $1/f$ noise in transistors and abnormalities in manufacture which cause high current density or high potential gradients in the transistor, and therefore may lead to early failure. To evaluate the nature of $1/f$ noise in transistors so that suitable noise limits may be specified, data has been gathered on several hundred transistors. This work will continue through CY 64.

The donner amplifier was selected as a representative type of circuit for packaging and for use as an operational amplifier requiring high voltage output and reliability. The primary concern in this effort was the heat transfer from heat producing components in the absence of convection cooling. The mechanical design required the use of a conduction type of printed circuit board, which consisted of an aluminum sub-base coated with epoxy on which the printed circuit was applied. This forms a printed circuit board which is a good conductor of heat and is very rugged and reliable. Using this board, transistors were mounted without heat sinks. The rugged construction is shown in Figure 23. Tests indicate that this type of packaging could be used in areas where convection cooling by air or other gases is not possible. The dynamic response of this development model was excellent.

NE-134I

NRX REACTOR CONTROL SYSTEM DYNAMIC ANALYSIS

PURPOSE: "To provide for the NRX control system development program, specifications of control loop configuration and controller functions and to analyze NRX reactor behavior under normal off-design and emergency conditions by means of computer studies. "

NE-1346

DEVELOPMENT ENGINE REACTOR CONTROL SYSTEM DYNAMIC ANALYSIS

PURPOSE: "To provide for the control system development program, specifications of control loop configurations and controller functions and to analyze development engine reactor behavior under normal, off-design and emergency conditions by means of computer studies. "

In providing analytical support to NE 1340 NRX Reactor Control System Design, emphasis was concentrated on an analysis of the existing KIWI-B4A controller for NRX-A2. The results of the system analysis were reported in TME-525. Figure 24 shows a typical low power startup transient through three decades of power, with a hold at 1% power. Figure 25 shows the power range transient from 1% power up to 100% power using the startup flow program from TME-386. The power and flow profiles were set to give a 25°R/sec exit gas temperature ramp with a power, temperature and flow hold for 10 seconds at 25% power.

Other activities carried on by the analysis group in connection with NRX control system analysis included:

- a. Hybrid simulation of the NRX reactor core, both at WANL and at P. C. C.
- b. Tie Rod Channel Simulation in order to provide a basis for NRX control limitations.

- c. Development of a signal generator required to make cross correlation measurements of various reactor and control system transfer functions during future NRX tests.
- d. Analysis of the Propellant Feed Control System to be used in NRX testing including the cold gas mixing chamber controls.
- e. Malfunction studies of the NRX control system for NRX-A2.

Figure 26 shows typical transient responses of the NRX core to both step changes in power and core inlet plenum pressure using a hybrid representation of the core. The computer (using time as analogous to length) computes the spatial temperature variation at discrete instants of time. The curves in Figure 26 were drawn through these points in a smooth manner.

Project NE 1346 has provided analytic support to the design effort on the Engine Reactor Control System and for the actuator project. Included in this effort are:

- a. Analysis of the Reference Reactor Control System using a complete engine simulator.
- b. Testing of the breadboard of the Reference Reactor Control System in conjunction with the analog computer.
- c. Studies on the effect of velocity limiting of the control drum actuators.
- d. Studies of the interaction between the Reactor Control System and the Propellant Feed System.
- e. Development of Reactor Core, Reflector, Turbopump and Actuator models.

NE-2200

GROUND SUPPORT EQUIPMENT, CHECKOUT AND TEST

PURPOSE: "To provide for the checkout and test equipment required for the nuclear subsystem at NRDS and AMR. To design the required equipment and to provide for the procurement, installation and test of the equipment. "

A study was made of the testing and checkout equipment required for the testing of the NRX reactors. A survey was made of the equipment now available at the test site, and it was determined that in order to increase the reliability and to meet checkout schedules of the NRX program, the following equipment was required:

1. Cart Wiring Transducer Checker
2. Control Drum System Checker
3. Nuclear Channel Criticality Checker Monitor

The cart wiring transducer checker, Figure 27, was designed and constructed and delivered to NRDS. The cart wiring transducer checker is capable of checking all wiring within the cart and reactor as to the continuity and grounds. It will help ensure that the cart and reactor are wired according to the drawings and it will give a printed record of continuity and grounds for future reference in case of any difficulties during testing. The cart wiring transducer checker will also have the capability of energizing any strain gauge bridge and checking the bridge for balance and continuity before the core is installed at the test cell. It will also give a printed record of the voltage applied to the strain gauge bridge and will give a record of the amount of unbalance within the bridge.

The control drum system checker, Figure 28, has the capability of checking each control drum actuator before it is installed within the reactor. This will give a recording of the signal applied to the actuator and a record of the position of the actuator during

its movement. The control drum checker will record the temperature of the actuator during this checking period. Various types of signals at various frequencies can be applied to the actuator to insure its proper operation. Provisions have also been made so that the actuator can be checked again after it is installed in the reactor. Similar records will be made during this second check. The control drum system checker has its own hydraulic supply, stand for checking the actuator out of the reactor, all necessary signal sources, and a recorder, giving a self-contained system.

NE-3300

NRDS SUPPORT ETS-1 I&C SYSTEMS

PURPOSE: "To provide assistance and support in establishing Instrumentation and Control (I&C) system functional requirements; to provide technical follow of detailed design and analysis of interfaces between the various I&C systems; to assist in review of detailed system designs and equipment specifications in preparation for equipment procurement. "

Under NE 3300 assistance was given to REON with regard to the I&C contract for ETS-1 instrumentation and control system. Design information and data were furnished in the following areas to ensure that the facility instrumentation, controls and data acquisition equipment were adequate for the requirements of the reactor during test.

1. Safety System
2. Facility Control System
3. Timing System
4. Power System
5. Signaling Conditioning System
6. Analog Recording System

7. Digital Data System
8. TCP System
9. Facility Modifications

NE-3600

NRDS SUPPORT, TEST CELL "A"

PURPOSE: "To provide criteria and design information for adaptation of Test Cell 'A' for use as an NRX reactor development facility."

Under NE 3600 a study was conducted to determine the type and number of channels of wideband magnetic tape recording necessary for NRX testing. This study revealed that approximately 200 data channels with frequency response above 10 cycles per second would be required. Therefore, it was requested that 195 channels FM/FM recording equipment be installed in Test Cell "A". It was further requested that this equipment be identical to the equipment installed by LASL in Test Cell "C", to ensure compatibility and reduce the complexity of the wideband data recording system. Discussions were held with LASL on the timing system and it was agreed that a new time code generator would be installed in the central control building for the use of Test Cells "A" and "C". The discussions have been held with LASL, AGC (REON) on the narrow band data reduction system and it was agreed that the system as proposed would provide adequate reduction facilities for such data at NRDS for the NRX program. Future studies were made on the measurement requirements for NRX-A1.

Information has been given to ACFI to insure that the test car will be compatible to the measurement requirements as developed. It has been necessary to add additional wiring from the test cart to the test cell for the wideband data recording system.

2 Control Cubicles

MANUAL DRUM POSITIONER

112 VOLT POWER SUPPLY

150 VOLT BIAS SUPPLY

TRIGGER CIRCUIT

COINCIDENCE CIRCUIT

DONNER AMPLIFIER

CONTROL DRUM SIMULATOR AND LOWER
ENVIRONMENT CHAMBER

COOLANT PRESSURE EFFECT ON
FREQUENCY RESPONSE

REVISIONS			
NO.	DESCRIPTION	DATE	BY
A	ADDED COMPONENT VALUES & GENERAL CHANGES	12-1-54	JEP
B	GENERAL CHANGES	12-1-54	JEP
C	GENERAL CHANGES	12-1-54	JEP

LIST OF MATERIALS			
ITEM NO.	DESCRIPTION	QUANTITY	REMARKS
1	REACTOR CONTROL SYSTEM	1	
2	NEVA ENGINE	1	
3	14683	1	
4	936J100	1

VISION OF CLEVITE CORPORATION

CLEVELAND, OHIO

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STARTUP - SUBPOWER RANGE - THREE SECOND PERIOD

STARTUP TRANSIENT - POWER RANGE

FIGURE 2

HYBRID REACTOR CORE SIMULATION
REAL TIME FUEL TEMPERATURE RESPONSE

NE-1344

NUCLEAR INSTRUMENTATION DEVELOPMENT

PURPOSE: "Develop, design, test and calibrate, as required, the nuclear instrumentation systems to be used for diagnostic and control purposes on ground and flight test reactors. These systems shall be suitable for accurately measuring the nuclear parameters in the high radiation flux and extreme environment of the NERVA engine. Satisfactory reliability and performance of the developed instrumentation systems and components shall be established by proof testing under simulated NERVA environmental conditions prior to use in the NERVA engine."

NEUTRON DETECTORS:

We started out this task by establishing the types of neutron detectors to be developed for use in Block I reactors for measurement of the neutron flux from source level to reactor full power and during restart of a reactor. Simultaneously, we developed specifications for amplifier systems suitable for use with these neutron detectors and with response fast enough to meet control requirements.

We have designed and partially tested a first generation set of neutron detectors. A compensated ion chamber, designated the WX-5362, is nearing completion. An early version of this ion chamber using titanium for the electrode structural material collected up to 10^{-2} amperes using only 250 volts of polarizing potential. This is a current ten times greater than typically collectable by previous ion chambers which are five times bigger than our WX-5362. The second version of this detector now being completed should retain this high current feature and show improved gamma compensation due to use of beryllium electrodes.

The prototype intermediate range detector, designated the WX-5406, is a low sensitivity fission counter. This detector has been designed to solve the formidable restart problem caused by high gamma ray levels with low neutron

levels which will be present a few minutes after shutdown of one of our reactors. This detector has been checked out with low speed electronics and the results indicate that when a fast electronics system is developed we will have a system capable of operating in a gamma field of 10^6 R/hr with negligible gamma pileup counts.

The prototype source range detector, designated the WX-5436, is a high sensitivity fission counter. This counter is a fairly conventional design which has been slightly improved based on design principles discovered in the intermediate range detector development.

AMPLIFIER SYSTEMS:

In the area of amplifier systems for use with these detectors, we have made considerable progress also. We have specified, procured and evaluated a nuclear instrumentation system for use with NRX-A reactors. (See Vue-Graph.) This is essentially a breadboard system that is quite flexible. We hope to be able to incorporate design improvements and circuitry for use with future developmental detectors into this system as testing on NRX-A reactors becomes available.

We have developed and tested a first generation radiation resistant preamplifier for use with the fast intermediate range detector. A pulse amplifier with a rise time of 0.04μ sec has been breadboarded but we have not solved the packaging problem to avoid high frequency oscillations. This amplifier is more than twice as fast as previous systems.

Progress has been made in all phases of the amplifier systems for the source, intermediate and power range systems but we have a long way to go. Besides solving several problems we have encountered so far, we must greatly reduce the packaging size and power consumption and obtain high reliability.

Six and two-thirds of the seven milestones have been met for project 1344. The single carryover is reactor testing of the source range prototype detector. This detector is the least critical of the three neutron detectors since it incorporates no

radical changes from previous fission counters. We are holding up the reactor test pending completion of a new low noise radiation resistant preamplifier designed for the source range. This design has been breadboarded and bench-tested. It is now being built out of radiation resistant materials and mounted on the detector.

NE-1347

NON-NUCLEAR INSTRUMENTATION DEVELOPMENT

PURPOSE: "To select, develop, design, test and calibrate, as required, the non-nuclear instrumentation systems to be used for diagnostic and control purposes on ground and flight test reactors. These systems shall be suitable for reliably measuring the non-nuclear parameters in the NRX-A-1 Cold Flow Test and in the high radiation flux and extreme temperature environment of the NRX-A-2 and NRX-A-3 power tests. Satisfactory reliability and performance of the developed and/or selected instrumentation systems and components shall be established by proof-testing, as required, under simulated NERVA environmental conditions, prior to use in the NERVA engine developmental tests."

ENVIRONMENTAL TESTING:

We are carrying out similar programs in all of our transducer areas. In general we use the requirements established by the NERVA Program Plans and the Development Plans in the NERVA Instrumentation Data Book to specify and select or develop if necessary, and it usually is, instrumentation for use on the various reactor, cold flow and vibration tests. A large amount of our effort goes into simulating the environment in which these instruments will have to operate. We have learned through experience that a transducer that operates properly in separate environments of cryogenic temperature, radiation and vibration will often fail in a combined environment of all three.

A device which has greatly advanced our program is a radiation-resistant vibration table for use at cryogenic temperatures. (See Vue-Graph.) This type of table was essential for our accelerometer program but is also being used to vibrate other transducers.

Another high point of our year's activity is the size reduction we have obtained in some of the transducers at the same time that radiation hardening has been done. (See Vue-Graphs.) We have qualified four types of pressure transducers for use in the NRX-A-2 test through combined environment tests.

TRANSDUCER DEVELOPMENT:

We have made considerable progress on accelerometers but the road is long and difficult for this type of transducer since conventional miniature accelerometers made out of piezoelectrics are quite sensitive to radiation. Our effort in this area has been to evaluate and try to solve the problems of the piezoelectrics as well as to develop miniature electromechanical types of accelerometers. We have developed two such devices, one based on an LVDT principle and the other based on a variable reluctance pressure transducer. Further size reduction will be required after we radiation harden these models. Although the first versions of these devices have not been successful under combined environment tests, the failures appear to be due to mechanical problems similar to ones we have solved in other transducers.

We have developed temperature compensated LVDT's which have passed our combined environment tests. The Vue-Graph shows one of the fixtures built for irradiation of 16 LVDT's in the Ground Test Reactor.

Our efforts on strain gages have been concentrated in the area of bonding techniques. We have successfully bonded gages to stainless steel, aluminum, graphite, titanium and beryllium for use at cryogenic temperatures.

THERMOCOUPLES:

One of our most formidable problems was the failure of W/W-26% Re

thermocouples at 4500° R in the KIWI tests conducted by LASL. We have completed and put into use a facility for testing thermocouples at temperatures up to 5300° R in a combined hydrogen-graphite environment. Our initial testing confirmed the failure of the tungsten-rhenium thermocouples at 4500° R and also showed degradation above 4000° R. We have determined the reason for the failure of the thermocouples and have had considerable success in solving the problem by using protective sheaths of tantalum coated with tungsten. Some of these thermocouples were recently tested for over 30 minutes at temperature between 4500° R and 4900° R. Then they were cooled to room temperature and vibrated for 90 minutes at between 5 and 10 g's. Finally they were retested for another 30 minutes at 4500-4900° R with the total degradation amounting to only 6 - 7%. The thermocouples are still intact.

EDDY CURRENT PROBE:

One of the many miscellaneous items our laboratory worked on during the year is of particular interest and is already in use on vibration tests. This is the portable eddy-current fuel element flaw detector which will be used to probe fuel elements in the NRX-A-1 cold flow test. (See Vue-Graph.) This device can be used to locate cracked fuel elements on either an alarm basis or by recording the output signal on an oscillograph. With practice it should be possible to probe coolant channels on an alarm basis at a rate of approximately one every five seconds.

The design has been modified for use in niobium-carbide coated fuel elements. In this case the detector locates rapid changes in coating conductivity such as would be caused by cracks or chips.

SUMMARY:

We have completed all of our milestones in Project 1347 with the exception of about two week's work in approving the instrumentation installation designs for the NRX-A-2 and A-3 power tests when these designs become available.

Considerable development is still required in the non-nuclear instrumentation area. Much of this effort will be directed towards improved reliability and accuracy.

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Atomic Energy Act - 1954

WANL-TME-573

More thorough proof testing under conditions that more closely simulate the NERVA environment is badly needed. Use of the Plum Brook Reactor with its higher flux and proposed cryogenic facility are expected to greatly aid our program. We have much to do in improving the high temperature tolerance of nearly all transducers so that more diagnostic tests can be made on hot reactor tests.

CONFIDENTIAL
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Atomic Energy Act - 1954

WANL ACCELEROMETER TEST SET-UP (23/W002) MAY, 1963 SNPO-C-GTR#4

GULTON
MAN-MADE
CRYSTAL
(QTY-2)

STATHAM
STRAIN
GAGE
GAS-DAMPED
(QTY. 2)

PHYSICAL
SCIENCE
LVDT
GAS-DAMPED
(QTY. 2)

WIANCKO
VARIABLE
RELUCTANCE
GAS-DAMPED
(QTY. 2)

ENDEVCO
NATURAL
QUARTZ
CRYSTAL
(QTY-2)

LVDT JIG

LVDT JIG FOR G.T.R. RADIATION TEST

NE-1391

REACTOR ASSEMBLIES FOR NON-NUCLEAR TEST SYSTEMS

PURPOSE: "To provide for the engineering study to determine nuclear subsystem requirements for simulators (reactor flow path and heat capacity mockup assembly for the cold flow development test system, a functional reactor mockup assembly for the NERVA engine hard functional mockup, and a core assembly mockup for training purposes in the E-MAD building)."

This project was inserted in the 1963 work program in order that Westinghouse perform analytical studies and establish a design for a number of reactor simulators to be employed in the NERVA development program. WANL-TNR-087 is the result of this effort. I will discuss briefly the recommendations made in that report, and the evolution of the program subsequent to its publication.

COLD FLOW DEVELOPMENT TEST SYSTEM:

The first of these simulator tasks was to design a nuclear subsystem for incorporation into the Cold Flow Development Test System. Considerable work with transient heat transfer digital computer codes led to a system design which incorporated an Al reflector, a shield consisting of double wall steel tubes, and a graphite core with heaters to allow a restart simulation. Figure 1 compares Be and Al in chilldown; figure 2 compares our simulated shield with the LiH, and figure 3 is a system chill-down study.

This design was not chosen for procurement and present plans call for rework of NRX-A1 and its incorporation into the CFDTs. It is our opinion, however, that the core should be replaced in the future by one with heaters to allow consideration of restart, and that a simulated flight-type shield should be put in when there is a more firm definition of its configuration.

SIMULATORS FOR NRDS:

Regarding ETS, we concluded that some type of cold-flow core would be desirable, but that detailed design would be premature.

NON-NUCLEAR TEST ASSEMBLY:

In the course of the above work, it became evident that a non-nuclear test assembly would be desirable to check core mechanical behavior under cold-flow conditions. Although a full-scale cold flow machine was already contemplated for use in NEVADA, a more accessible and readily changeable device was felt to be desirable. This device would be built to actual specifications except for access parts and instrumentation. This thinking led to the incorporation of the plugged core flow test in the component test program, as discussed by Dr. DeZubay this morning.

SUMMARY:

The work called for was accomplished as planned, and the milestones met.

NE-2710

NUCLEAR ACCIDENT ANALYSIS AND SAFEGUARDS SUMMARY REPORT

PURPOSE: "To study and specify provisions to prevent accidental criticality during storage, processing, shipment, and use of NERVA fuel material, to prepare safeguards reports and other documentation required to fulfill contractual obligations, and to determine the feasibility and study the design of a flight version of a full core nuclear poison system. The objectives were accomplished in the areas described below."

ASTROFUEL:

The fuel fabrication process and equipment was reviewed prior to activation of the facility with enriched uranium material. Based on this review the criticality limits and the criticality control procedures were established for the facility. Subsequently, modifications in the fabrication process have required additional substantiation of new methods and equipment such as new fixtures for the baking furnace, material traps for

the leaching and coating furnaces, and expanded requirements for the storage of fueled process scrap.

REACTOR ASSEMBLY:

Assembly of NERVA reactors by Westinghouse will be done at the Astronuclear Laboratory in Large, Pennsylvania. To insure a high degree of safety during these operations special facilities were constructed to permit the safe storage and handling of fuel material and the assembly of the NRX reactors. Special effort was exerted during the design of these facilities to make them self-sufficient, to essentially isolate them from other activities at WANL, and to exclude water. Final assembly of the reactor is accomplished with the fuel poisoned, which eliminates all possibility of accidental criticality of the assembled reactor which otherwise could possibly result from accidental moderation or reflection, malfunction of the control drum system, or errors in the fuel loading or fuel loading specification. WANL-TME-437 is a safety review of these operations.

SHIPPING NRX REACTORS:

Since the fully assembled NRX reactors must be shipped from the assembly area at WANL to NRDS, provisions for safe shipment were examined and have been incorporated into the equipment and procedures which will be used. The shipping container is a doubly sealed system which has been designed to protect the reactor from damage due to normal handling and transportation; contain the reactor material in the event of an accidental collision which is equivalent to a 30 foot fall and maintain its seal integrity. As an additional safeguard against accidental criticality, the core is shipped with sufficient nuclear poison to maintain an adequate shutdown margin in the event of total immersion or flooding with water or collapse of the core void volume.

A review of the safeguards for reactor shipment is given in "Safeguards Against Accidental Criticality During NERVA Core Shipment" - WANL-TME-524.

WANEF SAFETY ANALYSIS REPORT:

The Safety Analysis Report required for operation of reactors was prepared for the Westinghouse Astronuclear Experimental Facility (WANEF). This report, WANL-TNR-099, covers the safety aspects of operation of the Pre-Assembly Critical Experiments (PAX) and the Proof Test Critical Experiments (PTX) in Westinghouse facilities located in Waltz Mills, Pennsylvania. It established the adequacy of the facilities design, the safety equipment, the planned mode of operation, the experimental plan and the site location for safe conduct of the NRX critical experiments. Since the initial issue, an addendum has been prepared as a result of comments from reviewers.

NRX SAFETY ANALYSIS REPORT:

The preparation of an NRX Safety Analysis Report similar to that provided for WANEF was commenced early in the contract year. A June 6 letter from Aerojet, reflecting SNPO desires, caused a redirection of this effort aimed at preparing a broad SAR for the whole series of planned tests, plus a specific safety evaluation report for each hot test. An outline for the new SAR has been prepared and agreed upon; of the eight chapters called for, four have been prepared in draft form and submitted as a TME. The preparation of the remaining four chapters will be carried over through January 15.

FLIGHT COUNTERMEASURES:

The full core nuclear poison system is a countermeasure being designed to safeguard against accidental criticality in the vicinity of the launch pad during preparation or launch of the RIFT vehicle. The present concept employs a system of nuclear poison wires which are installed in the reactor during assembly of the engine. These remain in the core until after launch and are removed just prior to startup of the nuclear stages.

The nuclear poison considered for use is Boron-10 contained in a 0.050 inch stainless steel tube which is coated with teflon. Approximately 3000 of these "poison wires" are uniformly distributed throughout the core. The shutdown provided by this system is

sufficient to maintain the reactor subcritical in the event of total immersion and flooding with water or complete collapse of the core coolant channel voids.

Experimental equipment (figure 1) which simulates the reactor and nozzle configuration was fabricated to study problems involved in the design such as support block damage and determination of force required for removal. After initial unsuccessful attempts with the uncoated poison wires it was found that coating the wires with teflon eliminated the damage to the support block and reduced the force required for their removal. It is felt that this work has established the basic feasibility of this system.

WANL participated in the ROVER Flight Safety Panel program and developed information as required. The major formal effort was in preparation of our contribution to the "RIFT Hazards Summary Review" - WANL-TME-203.

SUMMARY:

All milestones were met with the exception of the NRX-SAR, mentioned above.

NE-2731

FISSION PRODUCT DIFFUSION STUDIES

PURPOSE: "The purpose of this program was to determine the rate of fission product diffusion by isotope as a function of operational history, temperature, and pressure. These data will be utilized to evaluate the extent of fission product diffusion for an entire NERVA reactor. With this information it will be possible to modify the fission product inventory and source term data as a function of time after shutdown. In the case of ground tests, the diffusion data will be utilized to evaluate environmental hazards arising from fission product release."

EXPERIMENTAL:

The approach taken in this study was to irradiate segments of NERVA fuel in the

GETR reactor at burnups anticipated during NERVA operation. A picture of the capsule utilized in these irradiations is shown in figure 1. After irradiation the fuel samples were heated at various temperatures in the controlled atmosphere diffusion apparatus shown in figure 2. This bell jar assembly is capable of attaining pressures as low as 10^{-6} mm of Hg and, with the use of induction heating, temperatures as high as 2500°C may be achieved. After heating, the samples were dissolved and analyzed radiochemically for ten isotopes which were selected for analysis since they constitute greater than 90% of the total activity present in a fuel sample, and represent in terms of general chemical and physical characteristics all fission products formed during operation of a NERVA reactor. The following variables have thus far been examined: time, temperature, environment and sample size.

Preliminary studies have been made on three fuel types:

1. Type I fuel which consists of 10μ of UC_2 dispersed in a graphite matrix.
2. Type II fuel which contains pyrocarbon covered UC_2 beads dispersed in a graphite matrix.
3. Type III fuel which is prepared by thermal treatment of Type II fuel at 2600°C .

In addition to the above experiments, low level irradiations have been made at the Westinghouse Reactor Evaluation Center and gross activity release fractions have been determined as a function of time, temperature, sample size and thermal pretreatment. One experiment has also been run in the TREAT reactor to determine the release fractions of the various isotopes when pre-irradiated NERVA fuel is subjected to rapid nuclear transients.

As a result of these studies a few general conclusions have been made:

1. The order of release rates for the nuclides studied has allowed a division into three main groups; fast diffusers, intermediate diffusers, and those whose diffusion rate is very slow or negligible within the limits of our experimental parameters.

2. The diffusion rate for all nuclides other than Zr^{95} and Mo^{99} is much slower from Type II fuel than Type I.
3. Thermal pretreatment at $2800^{\circ}C$ for 30 min. results in a degradation of the pyrocarbon coating and migration of UC_2 throughout the graphite lattice. The degradation of the beaded fuel (Type III) results in an increase of the fraction of gross activity released from 4% to 13% at $2100^{\circ}C$ in 15 minutes.
4. A helium environment retards the release of Sr89, Ba140, Cs137, 1131 and Te129. These results indicate that greater release rates will be achieved during a flight mission and correspondingly the release rate may be beneficially inhibited during operation of a ground test.
5. The release of some fission products is independent of the gross physical dimensions of the fuel particle. This effect simplifies the method used to devise an analytical model to describe the fission product release rates.
6. An 800 megawatt second transient in the TREAT reactor is sufficient to volatilize 15% of the test sample and release 7% of the gross activity present before the transient. This suggests a mechanism for reduction of the fission product inventory of the core after power operation in flight.

The results examined to date were obtained from the first five capsules of the twenty-one capsules which comprise the first phase of this program. These results are reported in TME-542, issued September, 1963.

ANALYTICAL:

A fission product inventory computer program was written, which includes: five hundred individual isotopes, multiple periods of reactor operation at constant power and of

decay, loss of isotopes limited to radioactive decay. The output of the code gives activity of the individual isotopes and of several energy group gamma and beta decay power.

SUMMARY:

All milestones were met with the exception of one on re-entry simulation runs on irradiated fuel, missed because of delay in negotiations between AGC and GETR.

NE-2732

MECHANICAL AND PHYSICAL ASPECTS OF REACTOR EXCURSIONS

PURPOSE: "To analyze and verify where possible by experiment the physical conditions leading to nuclear excursions of the reactor; to analyze the shutdown mechanisms which terminate these excursions and develop methods for calculating the energy released in the excursion."

WATER IMMERSION:

The results of both the experimental and analytical effort on the water immersion of the NERVA core are given in WANL-TNR-117, entitled "Hydrodynamics of NERVA Core Immersion." The basic objective of this program was to predict the density and the relative flooding rate of water into a NERVA core under a rapid immersion.

This study enables us to predict the initial water rise rate to within $\pm 20\%$ for hydrodynamic impact where the relative velocity between the core and the water surface lies between 30 ft/sec. This $\pm 20\%$ predictability of the initial rise rate would indicate about $\pm 25\%$ predictability of initial mean density.

ELECTRICAL TRANSIENT EXPERIMENTS:

A series of simulated nuclear transient tests using electrical heating was initiated this year. These tests were conducted to obtain better information on the threshold energies required to cause sublimation, fragmentation, and surface degradation of NERVA fuel.

The tests involved the resistance heating of 7.5 inch samples of unfueled NERVA geometry graphite for transients of 16 to 100 milliseconds duration. The effective energies deposited into the samples ranged from 500 to 5600 cal/gram. The energy required to cause fragmentation was between 1790 and 2000 cal/gram.

By comparing results with the conclusions from the TREAT experiments, the advantages of each method of testing will be utilized to enhance the overall understanding of conditions during reactor excursions, nuclear destruct and startup.

EXCURSION ANALYSIS:

The WANL excursion analysis program is based on two computer programs (RTS and RAC) developed by LASL. RTS is a neutron kinetics code; RAC studies disassembly due to internal pressures generated during very rapid heat generation. WANL has made certain modifications and additions to these programs where necessary to allow treatment of special problems of interest, such as control drum scram as a shutdown mechanism, tie rod spring constraint as an additional boundary condition, and additional power distributions.

The possible types of excursions are divided into three classes distinguished by the manner of reactivity insertion.

1. Abnormal control drum motion
2. Core deformation and compaction
3. Water immersion and hydrogen propellant flooding

Energy deposited in the core versus various step reactivity insertions typical of the above types of accidents is shown in figure 1.

NUCLEAR TRANSIENT EXPERIMENTS:

Forty-seven transient irradiation experiments were performed during this period on beaded fueled graphite specimens in the TREAT reactor in Idaho. The various types of fueled stock tested were as received, back-reacted to 3%, and back-reacted to 22%. TREAT transients were performed at sample heat generation levels ranging from 425 to 6460 calories per gram with reactor periods ranging from 77 to 39 milliseconds. The transients produced

peak sample temperature ranging from 1000°C to greater than 3500°C. The data are summarized in figure 2. These experiments provided information on the behavior of the fueled material in the following areas of interest:

1. Structural integrity of as-manufactured fuel stock, back-reacted 3% and 22%.
2. Damage to NbC liners.
3. Threshold of carbon vaporization.
4. Dimensional stability.
5. Damage to fuel beads.
6. Reactivity and fragmentation in water environment.

SUMMARY:

The work that was planned for this project during calendar year 1963 was completed and all of the milestones met.

NE-2733

RE-ENTRY SAFETY STUDIES R&D

PURPOSE:

"To evaluate the heating, vaporization and oxidation characteristics of reactor and engine fragments during re-entry and to determine the length of time required for these fragments to re-enter the earth's atmosphere following post-operational destruct. This information is required to evaluate source term data and to calculate the potential hazards associated with the random re-entry of reactor fragments. It further serves as a basis for establishing the functional requirements of a destruct system. In addition, this program will investigate the feasibility of utilizing non-explosive destruct system."

RE-ENTRY TIME:

With the finding that complete burn-up would not occur during re-entry, it became increasingly important to consider the length of time which fragments would remain in space. Initially the orbital decay time for fragments in circular orbits was examined. The decay times evaluated are shown in figure 2. An examination of these data indicates that a slight increase in orbital altitude offers a substantial increase in orbital lifetime.

SOURCE TERM PROGRAM:

Initial studies were made to determine the post operational source strength of NERVA fuel fragments following destruct action. The cases examined include sub-orbital start with destruct prior to reaching orbit, sub-orbital start with destruct accomplished in orbit, and orbital start in 100 and 200 nautical mile orbits. The results of this study were presented at the Rover Flight Safety Panel Meeting in April of this year. As a result of this study the following conclusions were drawn:

1. The higher the orbit, in which the reactor is started, the smaller the resulting radiological hazard.
2. Large destruct fragments (greater than one inch in diameter), present a radiological hazard from the point of view of whole body γ -doses.
3. Small destruct fragments, i.e., those in which the diameter is less than 1000 μ but greater than 100 μ present a hazard from an ingestion point of view.
4. The smaller the velocity vector added by destruct action the smaller the hazard. It was found that, in general, it was not possible to assign a destruct size criterion to encompass all types of missions. Therefore, in the later part of the contract year, the development of an integrated source term computer program was started, the purpose of which is to determine source term information for all types of NERVA missions.

CHEMICAL DESTRUCT:

A laboratory screening program was carried out for the purpose of evaluating the feasibility of a chemical destruct system for the post-operational disposition of the NERVA core. Tests were conducted in the temperature range from 20 μ to 2450 $^{\circ}$ C. A list of the reagents examined is shown in figure 3 along with the weights required to fragment the entire core. Of all the chemicals tested, chlorine trifluoride appears to be the most promising from the point of view of weight. As you can see, at low temperatures only 600 lbs. of ClF₃ are required for core destruction. However, the toxic nature of ClF₃ presents problems in developing a delivery system usable under flight conditions. This program was terminated because of the unlikelihood that a useful flight system could be developed.

SUMMARY:

All milestones in NE 2733 were met.

NE-2734

STUDY OF FUEL ADDITIVES FOR ENHANCEMENT OF FLIGHT SAFETY

PURPOSE: "An investigation of NERVA fuel containing additives and determination of the extent to which integral fuel additives can be used to improve the safe use of the NERVA reactor type in missions involving re-entry at uncontrolled locations. The primary purposes of additives in the fuel will be to accelerate the mechanical, thermal, or chemical destruction of the core by passive means, and to increase the rate at which core fragments are burned up during re-entry."

FABRICATION:

Seventeen candidate additives were chosen as generic types from a list of 52 possibilities with melting points 4000 $^{\circ}$ F. These were:

WC	B ₄ C	Ta ₂ N
Mo ₂ C	VC	ZrN
NbC	WB	TaSi ₂
ZrC	ZrB ₂	WSi ₂
TiC	TiB ₂	Re
TaC	NbB ₂	

These were incorporated into unfueled and fueled graphite mixtures (along with blank control samples), and processed in a standard fashion into rods.

TESTING:

The rods were subjected to mechanical property and oxidation tests over a wide range of temperature. Four additives, WC, Mo₂C, VC, and Re showed enhanced oxidation rates (C. 100%) while actually increasing tensile (15-50% and flexural strength (10-30%) of unfueled graphite. However, in the case of fueled graphite, it was found that the above effects were reduced or eliminated, and the particles themselves provide the additive effect, enhancing oxidation by approximately 50%.

SUMMARY:

This project was terminated with the agreement of SNPO and AGC. It is intended that further work on the enhancement by additives of the strength of unfueled graphite be pursued under a Materials Department Program.

Comparison of Beryllium and Aluminum Chilldown.

Shield Temperatures with Stainless Steel Liner

Exit Fluid Temperatures Vs Time

TRAIL CABLE IRRADIATION CAPSULE

WANL FP-1

TABLE 1

ENERGY RELEASES AND PEAK POWERS
FOR SEVERAL REACTIVITY STEP INPUTS

Prompt ΔK_p Step $(\Delta K_p = K - 1 - \beta)$	Total Energy (MW-Sec)	<u>-RAC-</u>		Energy Into Liquid (MW-Sec)	Peak Power (MW)
		Energy Into Temp. (MW-Sec)	Energy Into Vapor (MW-Sec)		
0.01	3005	3005	0	0	2.665×10^5
0.015	4924	4924	0	0	6.546×10^5
0.02	6925	6907	18	0	1.241×10^6
0.03	9728	8276	119	1333	3.081×10^6
0.05	16,833	10,365	2625	3843	1.036×10^7
0.07	23,866	11,522	7800	4544	2.526×10^7

<u>-RTS-</u>		
ΔK Step $(\Delta K = K - 1)$	Total Energy (MW-Sec)	Peak Power (MW)
0.006	1380	80
0.01	1475	1.42×10^4
0.013	2090	7.56×10^4
0.016	2800	1.88×10^5
0.02	3490	4.20×10^5

(continued):

<u>Kinetic Parameter</u>	<u>Description</u>	<u>NRX-A Value</u>
β	Effective delayed neutron fraction	0.0078
ℓ^*	Prompt neutron lifetime	2.5×10^{-5} sec.
$\Delta K/\Delta T^\dagger$	Temperature reactivity coefficient	$-1.1 \times 10^{-5} \text{ }^\circ\text{C}^{-1}$

† Doppler contribution is $0.26 \times 10^{-5} \text{ }^\circ\text{C}^{-1}$.

TABLE II

RESULTS OF TREAT EXPERIMENTS ON LINED BEADED FUEL

Area of Interest	Calories/Gram & Est. Temp.	Type of Fuel	Comments
Structural Integrity	6460 ($>4500^{\circ}\text{C}$)	As-manufactured	Intact Webbing Cracks
"	2720 ($\sim 3200^{\circ}\text{C}$)	Back-reacted (3%)	Intact Webbing Cracks
"	2930 ($\sim 3500^{\circ}\text{C}$)	Back-reacted (22%)	Fragmented (49.3 mils)
"	2345 ($\sim 2700^{\circ}\text{C}$)	Back-reacted (22%)	Fragmented (137 mils)
NbC Liner Damage	2000 ($\sim 2600^{\circ}\text{C}$)	As-manufactured	Melts (Threshold)
Carbon Sublimation	2000 ($\sim 2600^{\circ}\text{C}$)	As-manufactured	Threshold ($\sim 2\%$ weight loss)
Dimensional Stability	1800 ($\sim 2500^{\circ}\text{C}$)	As-manufactured	Radial & Long. (0.13%)*
Fuel Bead Damage	1600 ($\sim 2400^{\circ}\text{C}$)	As-manufactured	Threshold of Coating Damage
"	6460 ($>4500^{\circ}\text{C}$)	As-manufactured	Bead Dispersed
Fragmentation in H_2O	2060 ($\sim 2600^{\circ}\text{C}$)	As-manufactured	Intact
"	2495 ($\sim 2800^{\circ}\text{C}$)	Back-reacted (22%)	Fragmented (41.6 mils)

*

The growth in the radial dimension was approximately 2 to 3 times that in the longitudinal dimension at higher energy levels.

Per Cent Burnup vs Initial Particle Size and R-entry Angle

Angle	Initial Particle Size								
	1"	1/2"	1/4"	1/8"	1/16"	1/32"	1/64"	1/128"	1/256"
0°	7.0%	12%	20%	33%	40%	52%	43%	42%	41%
-1°	6.9%	12%	20%	33%	40%	53%	43%	42%	41%
-3°	6.0%	10%	19%	33%	40%	52%	43%	42%	41%
-5°	5.2%	10%	17%	33%	40%	52%	43%	42%	41%
-10°	4.3%	8%	14.5%	25%	38%	49%	43%	42%	41%
-30°	3%	7%	10%	18%	34%	43%	43%	42%	41%
-60°	2.3%	7%	8%	15%	25%	41%	43%	42%	41%
-90°	2.2%	7%	7.4%	14%	25%	40%	43%	42%	41%

Initial Conditions:

Altitude = 400,000 ft

Velocity : 25,000 FPS

DECAY TIME TO 400,000 FT. FROM CIRCULAR ORBITS
FOR GRAPHITE SPHERES

<u>Orbital Altitude</u>	<u>Particle Size</u>	<u>1956 Atm.</u>	<u>1959 Atm.</u>
1.8×10^6 Ft.	1/64"	56 days	3.3 days
	1"	3584 days	210 days
1.6×10^6 Ft.	1/64"	21.5 days	1.3 days
	1"	1375 days	87 days
1.4×10^6 Ft.	1/64"	7.8 days	0.5 days
	1"	495 days	33.5 days

CHEM. DESTRUCT LAB. TESTS

R. T. $\longrightarrow$ 2000°C

REAGENTS TESTED AND ESTIMATED MINIMUM

FULL SCALE SYSTEM REAGENT WEIGHT

NON-HALIDES

O₂ - 6000 Lb.
H₂O - Prohib. *
H₂O₂ - Prohib. *
HNO₃ - Prohib. *
N₂ - Prohib. *
K -1000 to 2700 Lb.

HALIDES

BBr₃ - 4000 Lb.
Br₂ - 6000 Lb.
HF - Prohib. *
HCl - Prohib. *
Cl₂ - Prohib. *
BCl₃ - Prohib. *
CrO₂Cl₂ - Prohib. *
FeCl₃ - Prohib. *

FREONS

CF₄ - Prohib. *
CCl₃F - Prohib. *
CCl₂F₂ - Prohib. *

FLUORINE & FLUORIDES

F ₂	6,000 Lb.	Prohib. * by itself
BrF ₃	14,000 Lb.	Prohib. *
BrF ₅	10,000 Lb.	2,000 Lb.
ClF ₃	8,000 Lb.	600 Lb.
IF ₅	14,000 Lb.	4,000 Lb.
UF ₆	Prohib. *	9,000 Lb.

HIGH TEMP > 400°C
1200°R > LOW TEMP

* Prohib. = Reagent weight equal to sample weight left >90% intact.

NE 1380

DEVELOPMENT TEST PLANNING AND ANALYSIS

PURPOSE: "To prepare an overall test program for NRDS test operations; to coordinate, schedule and issue detailed test plans and specifications; and to provide the technical coordination required for test conductance, test analysis and issuance of test reports."

It is the intention to first discuss special test devices which have been incorporated in the NRX-A1 test plan. These special test devices in general will be covered by discussing the cold gas supply system, the efforts to provide through nozzle photography, and arrangements to use recently developed eddy current probe. Following this, a discussion of the NRX-A Block I overall programs and of the NRX-A1 program and the NRX-A2 program both of which have been initiated during Contract Year 1963.

POSSIBLE COLD GAS SUPPLY SYSTEMS:

Early in 1963 it was recognized that there were advantages to the use of a cold gas supply system in testing power reactors. The principal reasons for this were that supplying a gas at a nominal temperature of 90°R would permit coolant supply at low temperature (close to the liquid hydrogen temperature) but without the effects of two-phase hydrogen being present. Secondly, it was recognized that the use of this kind of a coolant system would permit operation at low power steady state without the possibility of liquid hydrogen entering the reactor core. This is not possible with a liquid hydrogen feed in that, in this case, it would be necessary to operate at power levels of the order of 20% full power before it can be assured that no liquid hydrogen would enter the reactor core. It was the intention at that time and it remains so to test early power reactors under conservative conditions in which the power level is carefully increased with several holds at low power in order to provide time for an assessment of the performance of the reactor during test. A study was made to determine what kind of physical system would be required to supply cold gas over a wide range of flow rates in the temperature range from 60 to 100°R . Figure 1 indicates

at least three possibilities which were considered. The first is a system which utilizes ambient gas from the tank farm and cools the ambient gas before it enters the reactor nozzle by passing the gas through a counter current heat exchanger using liquid hydrogen on the other side as the coolant. One advantage of this system is that there is no transition of liquid to gas in the line directly connected with the reactor. However, the disadvantage far outweighed this advantage in that this concept required extremely large amount of heat exchange and an extremely large heat exchanger. The second alternative is a mixing chamber in which ambient gas and liquid are mixed directly in the chamber at a ratio that will provide the required temperature and that a total flow rate equivalent to that demanded. One of the principal advantages of this scheme is that there is no wasted coolant in the production of the cooled gas. The major disadvantage which was recognized at the time of the study was the possibility of instability occurring during the mixing process. The third method was the opposite of the first in that liquid hydrogen would be heated to the desired gas temperature before entering the nozzle. Again this would be accomplished in a counter current heat exchanger. This system had the minimum coolant wastage of the two systems in which a heat exchanger would be utilized and had the disadvantage of a phase change in the feed system as well as possible heat exchanger space problems in Test Cell "A". As a result of this study it appeared, at that time that the third method, that is the heating of liquid had the best possible combination of advantages and deserved some further study. It should be pointed out that the major disadvantage that was recognized for the mixing chamber was the concern over instability and was an intangible which resulted in a shift of interest at that time from the mixing chamber concept to the heated liquid concept. The heat exchanger was conceptually designed and its characteristics are shown in Figure 2. This device exchanged energy at about 1/10 of the rate that would have been required had the first method mentioned above been used. Note that the length of the tubes for the heat exchanger was about 7 feet and an overall size of practically 8-1/2 feet would have been required which was of the order which may have been feasible for Test Cell "A" providing certain arrangement problems had been worked out. Following the conceptional design a meeting was held between WANL,

REON, SNPO, and LASL in which the entire problem was discussed. It was noted, at that time that a mixing chamber had been purchased by LASL some time prior to this but had never been flow tested or used. Information available at that time on the mixing chamber indicated that it had been sized for duty which would have been acceptable for the NRX Program and that the area of principal concern was as mentioned previously, possible instabilities during the mixing process. It was proposed that the mixing chamber be activated and installed in Test Cell "A" and that tests be performed on it to determine whether or not its performance was adequate for the NRX Program. It was decided to attempt this rather than go into the detailed design phases for a heat exchanger as well as the long lead procurement efforts required. In this way it was the hope to determine in a reasonably short time whether or not a cold gas supply system using the mixing chamber could be employed since it was felt that this would result in an early solution to the problem if the mixing chamber performance was satisfactory. The mixing chamber was installed in Test Cell "A" and a number of tests have been performed since installation. A very condensed summary of these tests is shown in Figure 3. Six experimental plans have been carried out and a few remarks on these will be made. The first experimental plan was carried out in mostly open loop operation. That is, both the liquid hydrogen flow rate and the gaseous hydrogen flow rate were independently programmed. In general the performance was quite stable but the range of flow tests was rather limited. It was decided at this point to perform several other tests to determine if it were possible to automatically supply total flow demand and temperature demand. Tests 2, 3, and 4 were carried out in which efforts were made to test the system utilizing these automatic control systems and several instabilities were encountered in each test. It was decided that the most important point to determine initially was whether or not these instabilities arose due to a physical instability in the mixing process or whether they were due to control system induced instabilities. Consequently, the last two tests, that is experimental plans 5 and 6, were carried out mainly in open loop control to ascertain the degree of stability of the mixing chamber. In general it can be said at this time that the mixing chamber appeared to be

quite stable when operated in open loop phases and that, in addition, progress has been made toward the development of an automatic temperature control. It has been concluded that the mixing chamber appears to be capable in general in meeting the NRX program requirements and efforts in the near future will be directed towards testing of the mixing chamber control system complex in order to improve the control system performance.

NOZZLE PHOTOGRAPHY:

A few remarks, on the other special test devices should be made. The nozzle photography on the KIWI B-4A was found to result in extremely valuable information and it was decided to attempt to provide similar capabilities for the NRX-A1 cold flow test. Consequently, Aerojet has modified the NRX-A1 nozzle design to permit three windows 45° apart which will provide access for TV and motion picture photography through the central window and lighting through the windows on either side. The windows being used utilize the hot bleedport design for the engine and have the advantage of flow bypass around the windows thus permitting coolant flow through all tubes in the nozzle. The use of a television camera in addition to the movie camera will, it is hoped, provide additional information during the test. New techniques for recording the time and event on the processed film are being worked out in order to improve the availability of information during the viewing of the movies. This is also being handled by Aerojet.

EDDY CURRENT PROBE:

The eddy current probe which will be used on the NRX-A1 has been developed at the Astronuclear Testing Labs. This probe is usable in fuel that is either the coated or uncoated and has been found to be quite sensitive, in that cracks of any reasonable size in the element may be detected even though the crack does not extend in the coolant channel being probed. This probe will be used prior to test in order to provide an accurate record of the condition of the condition of the fuel being used, following each block of profiles to determine the effects of the testing on the fuel element integrity and will be used subsequent to all of the tests to provide a final comparison with the measurements made prior to testing.

NRX-A BLOCK I TEST PROGRAM:

Figure 4 outlines the major objectives of the NRX-A Block I program. Figure 5 gives the major objectives of the Block Program for each reactor. This is reported in draft form in WANL-TNR-138, and is presently under review by Aerojet. At the same time it is forming a basis for the early planning of A-1 and A-2.

NRX-A1 TEST PROFILES:

The NRX-A1 program has been documented by the issuance of a Test Plan and a Test Specification, numbers TNR-097 and TNR-121. Figures 6, 7, and 8 summarize the test profiles that are planned for use in the NRX-A1 test program. Phase I ambient gas test will utilize gaseous nitrogen, hydrogen and gaseous helium depending on the outcome of the other two tests. The profiles shown in Figure 6 are for ambient hydrogen and have the same shape as the nitrogen and helium except for the absolute values which are different. The flow rates for hydrogen may be multiplied by a factor of approximately 1.3 for helium and 3.7 for nitrogen in order to achieve the same system pressure and pressure drops. The profiles have been designed to provide pressure drops equivalent to the maximum pressure drops expected in the cold flow tests which follow and are shown in Figure 7. These cold flow profiles are essentially composed of ramps at several rates and are planned to be used for between the cold gas and liquid tests. These ramp rates are chosen to cover a range of equivalent power startups utilizing chamber temperature rates of change from 15°R per second to 100°R per second. Following the cold flow profiles a second phase of ambient gas tests are planned. These are composed of hydrogen and helium tests which are intended to provide information on reactor instructional integrity under conditions similar to emergency cooldown conditions. These are shown in Figure 8 for the hydrogen profile and the shape is similar for helium profile. The last portion of this phase of testing is the split flow test also shown in Figure 8. In these tests hydrogen will be introduced into the dome and nozzle simultaneously. It is intended to simulate as closely as possible the pressure drop relationship between the reflector and core expected for an actual full power run.

NRX-A2 POWER TEST PROFILE:

The NRX-A2 test program has been initiated by development of a rough draft of the test plan which is presently under revision having been reviewed by cognizant technical personnel. The power test profile presently planned is summarized in Figure 9. The points to be noted are the holds at low power level during the approach to 25% power, followed by a hold at low power (approximately 10 megawatts) for approximately 15 minutes during which it is intended to review the quick look data in order to permit a decision as to whether or not to go to a higher power level. While the figure shows 100% power for the second part of the profile, it is intended to mean at this time that planning is being carried out so that nothing will be done to preclude the possibility of obtaining full power. It should be noted that the present plans include the use of cold gas coolant in the lower portions of the power test, with a switchover to liquid feed at 42% of full power.

SUMMARY

In summary, the following were areas receiving major emphasis under Project 1380 during Contract Year 1963:

Conceptual design and analysis studies of a cold gas supply system were completed as well as development of performance criteria for the mixing chamber tests begun in Contract Year 1963.

Requirements for cold flow nozzle photography were spelled out and plans were made to incorporate the recently developed eddy current probe in the NRX testing program.

An over-all Block I Program was developed and has formed a basis for development of detailed test plans and test specifications.

The Test Plan and Test Specification documents for NRX-A-1 have been completed with the exception of final revisions to the presently issued NRX-A-1 Test Specification. Review of NRX-A-1 site procedures is underway on a continual basis as they are received.

The NRX-A-2 Test Plan has been developed and reviewed as a rough draft. A smooth draft for more formal review will be issued in November.

POSSIBLE COLD GAS SUPPLY SYSTEMS (Ref. TP-191, CRD)

1. Cooled High Pressure Gas -

Gas from tank ~~form~~ cooled by counter current flow of LH₂ in heat exchanger.

Advantages - no transition from liquid to gas.

Disadvantages - large heat exchanger, waste of LH₂.

2. Mixing Chamber -

Ambient gas and liquid mixed in chamber to supply total flow at temperature required.

Advantages - no waste of coolant.

Disadvantages - concern over stability of mixing process.

3. Heating of Liquid -

Liquid feed is heated in heat exchanger by counter current flow of ambient hydrogen.

Advantages - minimum coolant wastage for non-contact cooling.

Disadvantages - heat exchanger possible space problem in Test Cell A.

COLD GAS SUPPLY
HEAT EXCHANGER PERFORMANCE
(Reference TP-258, CRD)

Duty:	4.5×10^7 Btu/hr, or 13.4 megawatts
Gas Flow Rate:	8.4 lbs/sec
Liquid Flow Rate:	70 lbs/sec
Gas Stream Inlet Temperature:	540°R
Gas Stream Outlet Temperature:	90°R
Liquid Stream Inlet Temperature:	43°R
Liquid Stream Outlet Temperature:	90°R
Gas Film Coefficient:	1540 Btu/hr ft ² °R
Liquid Film Coefficient:	4650 Btu/hr ft ² °R
Overall Heat Transfer Coefficient:	1160 Btu/hr ft ² °R
Pressure Drop (Liquid Side):	28 psi
Pressure Drop (Gas Side):	2.8 psi
Length of Tubes:	7.0 ft
Dia. of Heat Exchanger :	0.72 ft
Approximate Weight :	1200 lbs
Average Heat Flux:	190,000 Btu/hr ft ²
Heat Transfer Area:	235 ft ²
Log Mean Temperature Difference:	165°R

SUMMARY OF MIXING CHAMBER TESTS

Exp. Plan No.	Flow Range	Control Mode	Remarks
I	30-50 lb/sec	LH ₂ flow rate and PCV-41 position	Stable
II	30	LH ₂ flow rate control, PCV-41 position	Stable
	30-50	LH ₂ flow rate, temperature	Stable at about 90°R Instabilities at about 124°R and above
III	15-35	LH ₂ flow rate, temperature	Some stable points but general instability in flow encountered
IV	20-35	LH ₂ flow rate, temperature	Similar to EP III
V	0-20	Programed dewar pressure, PCV-41 Temperature control	Unstable
	0-20	Programmed dewar pressure & PCV-41	Stable
	0-35	Programmed LH ₂ and PCV-41	Stable
VI	0-5	LH ₂ & GH ₂ programmed	Stable
	0-5	New temperature loop	Oscillated
	0-5	New temperature loop, reduced gain	Slight drift
	0-30	LH ₂ flow rate, temperature control	Stable for two ramps, 0.4 and 1.2 lb/sec ²

NRX-A BLOCK 1 PROGRAM
(Reference WANL-TNR-138, CRD)

Main Objectives -

1. To assess the suitability of the NRX-A design as a basis for development to meet the NERVA Engine requirements.
2. To establish the suitability of the Block I design for meeting the requirements of the initial engine test program.

NRX -A BLOCK I PROGRAM
(Reference WANL-TNR-138, CRD)

<u>REACTOR</u>	<u>TEST TYPE</u>	<u>OBJECTIVES and GOALS</u>
NRX-A-1	Cold Flow	<ol style="list-style-type: none"> 1. Assess the mechanical stability of the reactor under various flow conditions. 2. Obtain data useful in analysis.
NRX-A-2	Power	<ol style="list-style-type: none"> 1. Operate at steady-state power levels and temperatures of such magnitude that significant information for checking the steady-state design analysis can be obtained. 2. Operate the reactor at as high a power level as design analysis and test data indicate will not compromise the structural integrity of the reactor.
NRX-A-3	Power	<ol style="list-style-type: none"> 1. Operate at steady-state power levels and temperatures corresponding to full power conditions. 2. Integrated full power runs of approximately 20 minutes.
NRX-A-4	Power	<ol style="list-style-type: none"> 1. Establish capability of Block I reactor to meet initial engine test program by examining typical power transients. 2. Establish restart and recycle capability.

NRX-A1 TEST PROFILES

PHASE I

AMBIENT HYDROGEN (540 °R)

FLOW PROFILES

NRX-A1 TEST PROFILES

COLD FLOW PROFILES

NRX-A1 TEST PROFILES

PHASE II
AMBIENT HYDROGEN (540 °R)
FLOW PROFILES

PHASE II
AMBIENT HYDROGEN (540 °R)
SPLIT-FLOW PROFILES

NRX-A2 POWER TEST PROFILES

INTRODUCTION: NE 1381, 1382, 2800, 3000 (NRDS OPERATIONS)

I am going to discuss the effort performed by WANL personnel at NRDS operated under Projects 1381, 1382, 2800 and 3000.

During the course of the year, the NERVA Test Operations team has been manned with a group having a background qualifying them for nuclear test work. On-the-job training brought this group to the required state of readiness at the end of the contract year. Although NRX-A1 was not delivered for testing in contract year 1963, other important additional tasks fully occupied the available manpower. Preparations for NRX-A1 plus the extra work brought the year to a close at full NRDS activity. Certain work which had been planned for contract year 1964 was moved up into contract year 1963 due to the slip in NRX-A1 delivery to NRDS.

In general, it was originally planned that the NTO organization would be in a state of readiness such that NRX-A1 testing could begin the first week of contract year 1964. Consistant with the facts that an accepted mixing chamber system was not available and that some areas of NRX-A1 were not firm, this goal was met.

NE 1381

NRDS TEST OPERATIONS

PURPOSE: "To provide for the preparation and overall checkout for the NRX-A series tests; to conduct the reactor tests; to secure the necessary test date and to conduct the maintenance supervision of the facility."

TEST CELL "A":

During contract year 1963, WANL-NERVA Test Operations personnel underwent on-the-job training in the Test Cell "A" complex. In the Test Cell proper, at year's end, were an active Test Cell manager and five systems engineers. All had participated in an indoctrination program covering all components and systems at the Cell. These personnel had participated in several LASL reactor tests, and facility checks including numerous mixing chamber runs. A formal preventive maintenance program was instituted primarily

by NERVA personnel to document component failures and initiate corrective action to prevent future failures. Active participation in Test Cell modifications and preparation of standard operating procedures for the various systems was also accomplished.

CONTROL POINT "A":

At Control Point "A", the Control and Data Acquisition Groups were active in preparation of the Data Acquisition requirements, wide band instrumentation planning, and Data Reduction Codes for NRX-A1. Modifications to various control circuits, design input and analysis of the mixing chamber temperature and flow loop controllers was also performed. The Reactor and Neutronics Group and Test Equipment Engineering designed and instituted procurement of the neutronics positioning system with associated water tank, position indication system and Test Cell face cement work.

Preparation of mixing chamber Experimental Plans and Control Point check lists, analysis of the mixing chamber test results, review of the NRX-A1 test plan and test specifications, review of the NRX-A2 test plan draft, and preparation of inputs to the Operational and Program Requirements Documents were completed by Test Planning and Reporting with Test Equipment Engineering.

As of June and at the request of LASL, NERVA Test Operations personnel became increasingly involved in testing of the LASL designed and installed mixing chamber. This involvement was such that the last two experimental plans have been handled completely by NERVA personnel under the supervision of LASL. This included planning the run, preparing check lists, setting up the systems, furnishing the operating team, issuing the data and analyzing the results. Assistance was provided by NERVA engineers to LASL resulting in the issue of drafts of KIWI B-4A and B-1B Test Reports as well as the data reports covering the last few mixing chamber tests.

At the end of the contract year a staff of engineers, scientists and technicians had received on-the-job training in most phases of preparation of documentation required for conducting nuclear test operations, design and engineering work to accomplish modifications to the various facilities at NRDS and actual test operations at the Test

Cell "A" complex. This staff was in a state of readiness to conduct NRX-A1 tests.

NE 1382

NUCLEAR SUBSYSTEM ASSEMBLY, DISASSEMBLY, AND POST-OPERATIVE
OPERATIONS AT NRDS AND LOS ALAMOS SCIENTIFIC LABORATORY

PURPOSE: "To perform assembly, disassembly and post-operative examinations on the nuclear subsystems at NRDS and Los Alamos Scientific Laboratory."

TRAINING:

During contract year 1963, WANL-NERVA Test Operations personnel underwent on-the-job training in Reactor and Test Car Assembly, and Hot Cell operations in preparation for NRX-A testing. WANL technicians were integrated with the ACFI technicians to form an augmented group capable of handling the increased R-MAD Building work load. NRX-A1 Test Car Assembly was started and progressed as far as possible based on the information and components available at the end of the contract year.

An advanced technician training program using mockups and Ground Support equipment was initiated prior to NRX-A1, rather than as planned before NRX-A2, due to the late arrival of NRX-A1.

DOCUMENTATION:

Review of the NRX-A1 Test Specification and Test Plan, NRX-A2 Test Plan, ACFI Quality Control Inspection Procedures and the Test Car Assembly Manual was completed.

Procedures for Receiving Inspection, Test Car Assembly, and Mating of Reactor Pressure Vessel, Nozzle and Test Car were prepared and submitted for comments.

Some NRDS-R-MAD Standard Operating Procedures were written. A test mate-up of the Pressure Vessel Closure Head, Control Rod Guide Tubes, Reactor Support Stand and Adapter Ring was completed with no problems evident.

MODIFICATIONS:

Engineering studies of modifications to the Test Car, Test Cell Movable Shed, Test Cell Flare and redesign input to Ground Support Equipment on the cylinder slitting stand was accomplished.

POST-OPERATIVE:

In the post-operative area, data sheets for post-operative examinations were prepared, and remote handling facilities and equipment at NRDS and Los Alamos Scientific Laboratory were reviewed in connection with NRX-A2.

Seven items of Ground Support equipment were received and checked out. This picture shows the reactor trunion fixture.

At the end of the contract year, a staff of well trained R-MAD technicians and engineers was engaged in document preparation, receipt of ground support equipment, design and modification work, test car assembly and R-MAD maintenance. The capability to assemble and conduct post-operative examinations has been augmented by addition of qualified personnel and additional equipment. Test Car T-2 was 80% complete on September 30.

During the latter half of the contract year, several items of large magnitude were accomplished under this project which had not been planned or budgeted. Among these were: inventory of R-MAD equipment; input to Operational Requirements Document; liaison at other locations to procure information required for NTO prepared documentation; NRX Task Force; Safety Analysis Report for NRX-A2; upgrading NRX-A1 Test Car drawings; Advanced R-MAD technician training; and NRX-A1 dummy shipment.

NE 2800

TRAINING PROGRAM

PURPOSE: "The purpose of this project was to provide for a training and qualification program for operating and test personnel as required for the Nuclear Sub-System Development Program."

CONTROL POINT "A"

During Contract Year 1963, WANL-NERVA Test Operations personnel were engaged in the preparation of a formal Control Point "A" Operator Training program. The Operator Training Guide was written and the training carried out with both classroom lectures, simulator training and participation in actual test runs (to a major extent in mixing chamber tests). A large amount of reference material was collected to form a training manual.

At the end of the year training was essentially complete although formal certification of individual operators was not to be made until just before NRX-A1 testing starts.

Post-Operational qualification was not completed because advanced training was undertaken due to NRX-A1 slippage. This advanced training had been intended originally to be conducted after NRX-A1 and prior to NRX-A2.

NE 3000

NRDS GENERAL SUPPORT FACILITY

PURPOSE: "To provide assistance in planning and in establishing broad criteria and information for support facilities and services at the NRDS."

Effort was expended under this project in the preparation of the two issues of the Program Requirements Document. Assistance was provided to WANL and REON in the form of furnishing information to aid in establishing criteria for support facilities and services required at NRDS from the standpoint of reactor system operation, radiation environment and hazards consideration.

LIST OF REPORTS ISSUED:

Project 2800

1. Training course content.
2. NRDS Test Operator Qualification Program.
3. NRDS Test Operator Qualification Guide.

Project 1381

1. Data report for LASL is EP V and VI Test Cell "A" Mixing Chamber.

NE-2100

REMOTE HANDLING EQUIPMENT

PURPOSE: "To develop methods, procedures and equipment for remote assembly, disassembly and handling of nuclear subsystems at NRDS. To coordinate the equipment requirements for nuclear subsystems with those of the engine. To provide equipment required in the operation of the MAD, E-MAD, and Test Cell A facilities."

HOT DISASSEMBLY EQUIPMENT

During the early part of the contract year the design of the dome end loaded reactor was reviewed, as it evolved, from the standpoint of remote disassembly. Suggestions were made for minor changes in reactor design to facilitate disassembly.

The existing facilities and equipment at the MAD Building were reviewed for applicability to remote disassembly of NRX.

Operation sequences were determined for the remote disassembly operations on NRX-A2. Lists of equipment required to carry out these operations were drawn up. The functional requirements for this equipment were written and issued.

Midway through the contract year, a change in the approach to reactor disassembly was considered necessary. Changes in reactor design to improve its ability to withstand service conditions made remote disassembly by conventional techniques a virtual impossibility. In addition, the failures experienced in KIWI tests made it doubly important to be able to view the fuel elements, filler strips, etc. as they ended the test. The disassembly method adopted to answer these needs is a radical departure from conventional techniques. It is based on cutting away sections of the metal support barrel and inner reflector. The individual core parts are exposed for viewing, photographing and easy disassembly. The machine in which this core disassembly takes place is called the cylinder slitting stand. It is shown on Figures 1 and 2.

The Operation Support Plan for NRX-A2 was written and issued early in the contract year. This is currently undergoing updating and will be reissued.

The remote disassembly equipment for NRX consists of 38 separate tools, stands, fixtures and items of handling gear. Twenty of these have been designed and released for fabrication. Another fourteen were near completion and the remaining four have been started. These four are simple, short lead time items such as the lateral support screw removal tool. Nineteen items of disassembly equipment have been completed or are in fabrication at present.

POST-OPERATIVE EXAMINATION EQUIPMENT

Engineering support was furnished for the planning for post-operative testing.

Trips were made to several hot cell facilities to appraise their capability to do post-operative testing for NERVA reactors.

Functional requirements were written for fifty-one pieces of post-operative equipment. Due to the fact that the post-operative testing program was not firm, these were not issued. The ones pertaining to NRX-A2 have since been issued.

Specifications were written for eight items of post-operative testing equipment.

Design of thirteen items of equipment have been started and one has been completed.

The preparation of the part of the NRX-A2 Operation Support Plan dealing with post-operative testing has been started.

An equipment study has been performed to determine the equipment required for support of the NRX program in the E-MAD facility.

TEST CAR

During the contract year, WANL has maintained liaison with the various groups interested in the test car and the test assembly (i.e. NRX reactor mounted on the test car) to assure that the various subassemblies are compatible and properly assembled to make possible efficient carrying out of the tests. One result of this is the test assembly drawing which depicts

the entire test assembly, references drawings of the various subassemblies which make it up, and calls out the pertinent assembly procedures to be used.

Design of the equipment above the privy roof has been carried out by WANL. This includes equipment for supporting the reactor and maintaining proper alignment between it and the control actuators, assembly tooling, various items of piping for connecting the reactor to the sources of gas, instrumentation boom, personnel stands for assembly and test and supports and enclosures for instrumentation mounted on the test car.

Design of a system to provide split flow gas injection into the pressure vessel head was accomplished.

In the course of work for the NRX-A1 reactor test, 92 drawings have been completed. Figure 3 shows an isometric view of the completed test assembly.

Technical supervision of the ACFI work on the test cars has been maintained. The adequacy of the existing KIWI (T2) car for NRX-A1 testing was evaluated and the modifications needed for the basic car were established. In the course of this work, over 800 drawings were reviewed. Follow-up of design of these modifications was maintained. The ACFI test car assembly manual was reviewed.

Work commenced on design of the car for the NRX-A2 reactor test. Calculations of activation of the various parts of the test car were made and evaluated. Conceptual designs for the purpose of reducing activation were prepared for some parts in case of future need. Radiation heating and the resulting equilibrium temperatures were calculated for many of the parts of the test car and attached equipment.

Drawings of the reactor support stand, vessel adapter ring, personnel platforms, instrumentation boom, stand outlet water system, and privy roof modifications have been prepared and released to ACFI for fabrication for NRX-A2. As of October 1, 1963, 53 drawings have been prepared for the NRX-A2 test car.

The section of the hazards report dealing with the test car was written.

MISCELLANEOUS

Certain tests were conducted to confirm the design basis of disassembly tooling.

The first of these tested abrasive cutting of graphite-aluminum-stainless steel for designing the cylinder slitting stand. Another series of these tested abrasive wheels for cutting heavier sections of aluminum and stainless steel cluster hardware.

A number of reactor parts were procured and fabricated for use by NRDS in operator training.

A brief study of on stand hot maintenance of the reactor was conducted. It was concluded that such maintenance was not practical.

NE-2300

GROUND SUPPORT EQUIPMENT, HANDLING AND MAINTENANCE

PURPOSE: "To design, procure and install the reactor and reactor subsystem support equipment of a portable, semi-portable or mobile nature required for handling, assembly, maintenance and storage of the reactor and reactor subsystem in the test and operating areas of NRDS and AMR."

GENERAL: There are several groups of equipment included under this heading. The general pattern of work for most of these was similar.

Planning for equipment requirements for NRX-A1 was carried out. Operation sequences, equipment provisioning lists, and functional requirements were written and issued. The pertinent parts of the Operation Support Plan were written and issued.

Designs of equipment were prepared and released for fabrication. Orders were placed and equipment was fabricated.

REACTOR RECEIVING EQUIPMENT

The equipment listing calls for eight items consisting of stands, tooling and handling equipment. The design and fabrication of all of these was completed. Figure 4 shows a typical item, the shipping container lifting fixture.

REACTOR INSPECTION EQUIPMENT

This category consists of 120 items of equipment. Technical release was given on all of these items and they were placed on order. Procurement of twenty-five of them was completed.

REACTOR NON-REMOTE DISASSEMBLY EQUIPMENT

The provisioning list calls for twelve stands, racks and fixtures. The design and fabrication of these items were completed. Figure 5 is a typical item. It shows the reactor cold disassembly stand as the outer reflector is raised.

REACTOR PARTS STORAGE CONTAINERS

The design of the NRX-A1 shipping and storage containers was completed. Figures 6 and 7 show a typical reactor parts storage container.

ASSEMBLED REACTOR SHIPPING

Planning for shipment of assembled reactors was carried out. It was decided to use

rail shipment as the primary method and to develop backup capability for truck shipment.

Shipping equipment was designed, procured and installed for shipment of a simulated reactor to MAD Building. This shipment was successfully carried out. Figures 8 and 9 show the reactor flat car and special trailer used in the shipment.

ACFI HANDLING EQUIPMENT

ACFI was assigned the responsibility for design and fabrication of handling equipment for installing the reactor in the vessel, positioning the assembly on the test car mounting stand, removing the assembly after test and breaking it down into the major subassemblies such as reactor, vessel, and nozzle. WANL is responsible for technical supervision of this work. WANL has established requirements for equipment, written functional requirements, prepared provisioning lists of equipment, written the sections of the Operation Support Plan pertaining to the equipment and approved ACF designs.

The equipment required for NRX-A1 totals nine separate pieces of equipment. All of these have been fabricated and delivered.

For NRX-A2, in addition to the above equipment, nine other pieces of equipment are required. Of these, five have been designed.

SUPPORT EQUIPMENT FOR VIBRATION AND SHOCK TESTING

The Ground Support Group designed the items of equipment necessary to take the last assembly stand, deliver and position it on either the vibration test stand or the shock test stand. There were five pieces of equipment involved. Twenty drawings were prepared in designing the equipment.

NE-3200

NRDS SUPPORT, ETS-1

PURPOSE: "To provide assistance and support in establishing functional requirements, detailed design and technical follow of the fabrication, installation and activation of the ETS-1 facility.

ACTIVITIES:

Drawings and criteria reports were reviewed for general familiarity with the facility. Fluid and gas systems were reviewed briefly from the standpoint of protecting the reactor against overheating after shutdown.

NE-3400

NRDS SUPPORT, ETS-2

PURPOSE: "To provide assistance and support in establishing ETS-2 facility functional requirements, review of detailed design and integration with ETS-1."

ACTIVITIES:

None.

NE-3500

NRDS SUPPORT, E-MAD BUILDING

PURPOSE: "To provide for technical review of the E-MAD facility designs considering operational requirements of reactor maintenance, assembly and disassembly and post-operative examination."

ACTIVITIES:

The Vitro design of E-MAD, Phase I, was reviewed. This review included detailed perusal and comments on approximately 250 drawings (reviewed an average of four times each) and 50 specifications (reviewed an average of three times each).

Plans were made for post-operative testing in E-MAD Building. Requirements for hot cells and other facilities were established. Conceptual layouts were prepared for the hot cells and materials handling equipment. This work was summed up in WANL-TNR-171, "Criteria for Phase II E-MAD Facility Hot Cells". Subsequent to this, a large number of memos have been sent to REON giving detailed requirements for the E-MAD hot cells.

The Vitro designs of E-MAD, Phase II, both preliminary and final have been reviewed and comments have been forwarded to REON. This review covered approximately 150 drawings (reviewed an average of four times each) and 75 specifications (reviewed an average of three times each).

A study was made of the capability of E-MAD Building for handling NRX reactors. This study was summed up in three study memos, WANL-GSE-128, WANL-GSE-138 and WANL-GSE-180. In these memos are described the facility additions required to accommodate NRX and E-Engine reactors both singly and concurrently.

NE-3600

NRDS SUPPORT, TEST CELL A

PURPOSE: "To provide criteria and design information for adaptation of Test Cell A for

use as an NRX reactor development facility."

ACTIVITIES:

LASL specifications for a proposed liquid nitrogen system for Test Cell A were reviewed. Comments on the suitability of this system for NRX testing were transmitted to REON in WANL-TME-268.

Calculation of gas tank farm requirements, allocation of gas storage space and the timing and sequencing of gas usage during and after various proposed run profiles have been a regularly recurring activity during the year. Requirements and utilization of cryogenic fluids have also been included in this activity.

Contact has been maintained with the Reactor Analysis Group to secure continuing, up to date information on fission product decay heat to serve as a basis for design of cooldown facilities.

Work was done on conceptual design and analysis of a liquid hydrogen system for emergency cooldown.

Ground Support Group participated in the preparation of WANL-TP-258, "Feasibility Study of a Single Phase Propellant Feed System for Test Cell A". Subsequently when the mixing chamber type of single phase feed system was selected, WANL-TME-374, "Performance Criteria for an NRX Propellant Feed System Using the Mixing Chamber Principle" was prepared and issued. Subsequently the work at the Test Site in assembling and testing the mixing chamber system has been followed.

The facility requirements for NRX testing have been analyzed and compiled in the following memos:

WANL-TME-279, "Recommended Modifications to Extend Test Cell A Capabilities for NRX Testing"

WANL-TME-340, "Request for Test Cell A Modifications"

Assistance was given in determining requirements for movie and TV camera coverage through the nozzle in cold flow tests. WANL-TME-401, "NRX-A1 Specification for Cold

Flow Core Observation" was prepared in cooperation with Test Engineering Group.

Calculations of neutron activation of the parts of the test cell adjacent to the reactor were obtained. From these calculations an evaluation of the problems of personnel exposure in work at the test cell was made.

Gamma and neutron heat generation rates were calculated and from these, temperatures of the pad and rails at the cell were determined. Preliminary analysis of the transient temperatures of the test cell shield wall due to neutron, gamma and flame radiation heating was made.

Test Cell modifications to provide split flow gas injection into the vessel head were worked out with NTO personnel. These modifications are to provide capability for testing with design pressure drop in both the reflector and core using ambient temperature gas.

Contributions were made to the Program Requirements Document in relation to Test Cell A.

NE-3800

NRDS SUPPORT - RADIOACTIVE MATERIAL HANDLING, DECONTAMINATION
AND STORAGE COMPLEX

PURPOSE: "To provide support in establishing functional requirements and reviewing the design. To provide technical follow of the fabrication, installation and activation of the above facility."

ACTIVITIES:

The major effort in this project was directed toward defining the reactor subsystem requirements for the storage and disposal complex. WANL-TME-287, "Radioactive Materials Handling, Decontamination and Storage Complex, Design Criteria", stated WANL's basic functional and operational requirements for the storage and disposal complex. Subsequently WANL-TME-117 and WANL-GSE-073 amplified the requirements set forth in the earlier document.

AGC report No. 2592, "Radioactive Materials Handling Decontamination and

Storage Complex, Design Concept and Criteria" and AGC report No. 2603, "Engine Transport System Maintenance Building Design Criteria" were reviewed from the reactor subsystem standpoint. Comments were forwarded to AGC in WANL-GSE-151, "Review and Comments on AGC Reports No. 2592 and No. 2603".

Calculations of spent fuel cooling requirements and reactor monitoring requirements in the hot engine hold facility were presented in WANL-GSE-220.

561844

CYLINDER SLITTING STAND NT650060

**NRX-A TEST
ASSEMBLY**

7. DYE PENETRANT INSPECTION OF ALL WELDS REQUIRED AFTER PROOF TEST & BEFORE PAINTING UNLESS OTHERWISE NOTED

Lonstronuclear

SECTION-AA

FLAT CAR WITH EQUIPMENT MOUNTED

NE-1980

RADIATION EFFECTS PROGRAM

PURPOSE: "To plan, coordinate and schedule the overall WANL-NERVA Radiation Effects Program to assure that the technical requirements are adequately considered, and to assist in test performance. This project is part of the overall NERVA Radiation Effects Program which is established with the objective of providing assurance by experimental testing and analytical examination that the NERVA components will perform their intended functions under the environment to be encountered during NERVA operation. "

Project 1980 comprises four main phases:

INTEGRATION AND TECHNICAL MANAGEMENT:

It is the function of the Integration and Technical Management effort to assure that radiation tests on this project are soundly conceived and designed; that experimental data are sound and meaningful, and that useful auxiliary data are taken where appropriate in connection with each test. To accomplish this phase of the program, it is necessary to prepare and submit:

1. Preliminary and final test plans
2. Test specifications
3. Test reports

The total number of documents of various types which were issued to fulfill this phase of the subtask is illustrated in the next Slide (1980-2):

Total Preliminary and Final Test Plans, Issued	
Preliminary and Final Test Specifications, Issued	54
Reports, Various Types	28
Drawings	67

EXPERIMENTAL DESIGN AND TESTING:

The main function of the Experimental Design and Testing effort is two-fold:

1. To provide the necessary engineering designs, specifications, and installations for conducting the required radiation tests.
2. To conduct experiments on mechanical, electrical and instrument components and structural materials.

During the year, four series of tests were conducted at GTR: 4, 7, 7a, and combined tests 8 and 9. Test 4 is shown in Slide 1980-1, and the irradiation position of each test is illustrated. Test 4 consisted of the following tests: Transition Joint, Neutron Detectors, Neutron Detector Preamps and Cables, Accelerometers, Resistance Thermometers, Strain Gauges, Pressure Transducers, Electronic Components, and Control Drum Bearing. The transition joint and the neutron detectors were positioned in the reactor pool. The neutron detector preamplifiers and cables and electronic components were attached to the reactor face of the north dewar. Pressure transducers, resistance thermometers, strain gauges and a control drum bearing were located in the east dewar.

The transition joint was allowed to remain in the pool for 1000 Mw-hr or irradiation. The other test items were irradiated for a period of 100 hours during reactor operation at 3 megawatts, except for the neutron detectors which were left in the pool for approximately four hours for a series of transient tests.

Early in the test, detonations in the LN_2 dewars occurred, making it necessary to continue the test with gaseous, rather than liquid nitrogen.

The aluminum-to-stainless steel transition joint is used in the design of the in-pile, water-cooled capsule in the HT-1 tube of the experimental installation at PBRF, and is located about 8 ft. from the reactor core midplane. Extrapolated measurements indicated the transition joint would be exposed to a fast flux of about 3×10^{14} nvt. Since little was known of the effects of irradiation, a WANL experiment was designed and performed at GTR to prove the integrity of the joint for PBRF use. The transition joint was of 2-in. diameter 3003 aluminum and 304L stainless steel.

The transition joint assembly and its pressurization system, as shown in 1980-2, was placed in the water pool of the GTR, close to the reactor core, and pressurized with gaseous N_2 internally and externally in alternate 8-hr. cycles. There was no evidence of failure after 300 hrs. of reactor operation. During irradiation, the temperature of the transition joint ranged from about $70^{\circ}F$ to $125^{\circ}F$ while exposed to an integrated fast flux of 3.7×10^{14} nvt. This exposure is equivalent to about 1600 full power hours of PBRF service.

From post-irradiation experiments which were conducted, it was concluded that the joint could withstand the maximum combination of shear and tensile stresses anticipated in the W-2 loop at PBRF.

Neutron detector pulse preamplifiers were irradiated in GTR Test 7. They were placed in a dummy fuel element and irradiated for 211 Mw-hr.

The test specimens which comprised GTR Test 7a, and their positions during irradiation are illustrated in 1980-3. They include: Pressure Transducers, Accelerometers, Resistance Thermometers, Strain Gauges, Linear Displacement Transducers, Thermocouples and Connectors, Cables, Control Drum Actuator Components.

The total run time was 114 hours and total power generated was 300 Mw-hr.

The components included in this test were shown in 1980-4, 5, and 6.

GTR Tests 8 and 9 were combined as illustrated in 1980-7. Tie rod components were irradiated in the north dewar for 150 hours, tie rod materials in the east dewar for 150 hours, and actuator components in the west dewar for 300 hours. Test items included:

Tie Rod Components, shown in 1980-8

Tie Rod Materials, shown in 1980-9

Actuator Components, shown in 1980-10

Post-irradiation testing is being conducted now. Results are not yet available.

GETR TESTS:

Irradiation experiments were conducted at the General Electric Test Reactor hydraulic shuttle facility or the trail cable facility to investigate the radiation stability

WANL-TME-573

of various materials considered for NERVA reactor service. The specimens irradiated included candidate NERVA shield capsules, dry lubricant bearing component materials, and two series of NERVA fuel tests for fission product studies.

A series of 14 capsule irradiations of dry lubricant bearing materials was made using the shuttle facility. Each material was encapsulated in both a low pressure and a hydrogen environment. A test capsule of the type used is shown in 1980-11. It was generally found that materials containing Teflon as a binder experienced extreme dimensional instability during irradiation. There were also several indications of Teflon Decomposition. Materials with DuPont poly-imide binders and the stainless steel coated with a solid lubricant, showed excellent resistance to irradiation.

The GETR trail cable facility was used for LiH shield material irradiation and for fuel irradiations to develop a fission product inventory. Post-irradiation examination of the LiH capsules has not been completed. Data from the fuel irradiations is being used in the WANL Source Term Program to define hazards related to NERVA flight conditions as was discussed in Project NE 2731 yesterday.

PBRF

Installation and operation of the in-pile W-2 water-cooled loop at the Plum Brook Reactor Facility progressed sufficiently, so that experiments could be conducted during the latter part of Calendar Year 1963. Tests conducted are listed in 1980-12. These components were irradiated for 60 hours; total power generated was 1993 Mw-hr.

In addition, two other tests, the control drum bearing test, and the LVDT's, were delivered to the PBRF site for testing in October.

RADIATION EFFECTS INFORMATION, ANALYSIS AND REPORTS:

The Radiation Effects Information, Analysis and Reports phase of Project 1980 consists of compiling and maintaining radiation effects information, on a current basis, of the various WANL tests conducted.

One of the most important phases is the maintenance of the Radiation Effects Data Book on status of component selection and design. The RED book is a comprehensive record of all tests conducted, containing specific, detailed data as well as background and general information for each of the various experiments which are programmed.

Analytical studies are conducted on the components from the standpoint of radiation effects on performance and durability.

In addition, summaries of the analytical studies are prepared in advance of each test, and evaluation of all test data is prepared in the form of test reports after the completion of the tests.

TEST OPERATIONS SUPPORT:

The Test Operations Support phase of Project 1980 involves the establishment of an operating staff at several reactor sites. At present, a resident engineer is located at GD/FW and a resident engineer and operations staff are located at PBRF.

NE 1981

PBRF MODIFICATION DESIGN AND FOLLOW

PURPOSE: "To provide the conceptual design, detailed design and specifications required for the procurement of in-pile test loops for the PBRF, the engineering follow of the construction, installation, and checkout, and to provide a preliminary and final hazards report and other required documents. "

DESCRIPTION OF PBRF:

A majority of the experiments will be conducted in the Plum Brook Reactor Facility, since the high radiation levels of this reactor more closely simulate those required in the NRX reactor system. The Plum Brook Reactor is capable of a maximum power generation of 60 megawatts and neutron fluxes in HT-1 as follows:

1.	Fast neutron flux for flooded HT-1
at core centerline	$2.3 \times 10^{13} \text{ n/cm}^2 \text{ sec}$
2.	Peak fast neutron flux, predicted	$1 \times 10^{14} \text{ n/cm}^2 \text{ sec}$
3.	Peak thermal neutron flux	$8.4 \times 10^{14} \text{ n/cm}^2 \text{ sec}$

During Calendar Year 1963, conceptual designs and specifications were completed for two major loop installations at PBRF. These are the W-1 Cryogenic Test Loop, and the W-2, Water-Cooled Test Loop, which was installed and operated during the latter part of Calendar Year 1963.

PROPOSED W-1 CRYOGENIC TEST LOOP:

1981-1 is a flow diagram of the proposed W-1 Cryogenic Test Loop which is to be installed in PBRF late in Calendar Year 1964. This loop will provide a high radiation-low temperature environment for the performance testing of a variety of basic components and materials. Use of a cryogenic helium atmosphere, capable of cooling helium in the range of 50° to 150°R , and having a heat removal capability of 20 kw, is proposed.

In operation, helium will be pressurized to about 1000 psi, water-cooled, filtered and further cooled and purified in a cold box. Cooling to 50°R will be accomplished by expansion of the helium in the expansion engine. After passing through the experiment, the helium flows to the cold box, heat is interchanged, and the warm gas returns to the compressor. The flow is shown separately.

A conceptual design has been completed, and a preliminary Design and Hazards Manual has been submitted to PBRF for comments. Discussions with PBRF personnel indicate general agreement with the design and there are no foreseeable problems on the associated hazards analyses.

Drawings and specifications suitable for contractor selection have been completed. The bid package was submitted to SNPO-C for comments and approval, and will be ready for release this month.

W-2 WATER-COOLED LOOP INSTALLATION AND OPERATION:

During CY 1963, a dynamic in-pile capsule and water-cooled loop were fabricated, installed and put into operation in the HT-1 facility of the Plum Brook Reactor. Three series of tests were completed before the end of CY 1963, in addition to dosimetry studies.

Accomplishment of this phase of the work required completion of the following major work details:

- a. Design, procurement and operation of the water-cooled loop.
- b. Preparation of a Design and Hazards Manual which was approved by the AEC. Preparation of an Operations Manual for PBRF approval.
- c. Establishment of a technical staff at PBRF, and subsequent indoctrination of the experiment operators in PBRF procedures.
- d. Instruction of personnel in reactor operations and certification of loop operators. We have at present located at the site, a WANL Supervisor, two certified engineers, three certified technicians, and several in training.

The W-2 loop and capsule system illustrated in 1981-2 is designed to allow remote operation of valves for insertion and withdrawal of an experiment during reactor operation. Cooling for the system is provided by reactor primary coolant which is diverted through an end entry, re-entrant capsule. An emergency, demineralized water supply is provided which is also used for purging during experiment withdrawal. Instrumentation for flow, temperature and pressure measurement, is part of the system. A helium gas system for actuating experiment components during irradiation is an integral part of the design.

The W-2 capsule, detailed in 1981-3, is situated in a horizontal through-hole of the reactor to provide a facility near the core centerline which yields high fluxes. 1981-4 shows the location of the system in Quadrant C of the PBRF. This capsule is 8 1/2 inches in diameter by about 13 ft. long and can accommodate specimens up to 6 inches O.D. by 6.5 inches long. Under present PBRF safeguards approval, maximum

weight of an experiment is 9 lbs. The capsule is designed for an external pressure of 160 psi and to operate at a maximum temperature of 300°F.

The in-core section of the capsule was fabricated from 8.5 inch O.D. aluminum seamless pipe and the out-of-core end of stainless steel. The aluminum-to-stainless steel transition joint described earlier, which connects both sections is shown here (refer to 1981-3).

Extending the length of the capsule interior is the sample holder (point to 1981-3), which is constructed of 2-in. Alcoa Unitrace. The Unitrace holder centers and supports the sample fixture within the capsule and provides a compact supply for coolant and gas lines.

A photo of the sample holder is shown in 1981-5.

W-2-1 WATER-COOLED LOOP:

Preliminary design of a third loop was initiated and major equipment is being requisitioned. This is the W-2-1, water-cooled loop, or trail cable facility shown in 1981-6. This facility will allow more rapid insertion and withdrawal of experiment in HT-1 for short-term irradiations, and thereby provide a facility for the many scheduled tests in CY 1964. The coolant system is independent of the reactor and can therefore be used for fueled experiments.

MOCK-UP LOOP:

Design and installation of a mock-up capsule and loop design was begun. This system is being installed at the WANL-Large Site and will simulate the temperature, flow, and pressure conditions in HT-1. This will permit evaluation of test performance in a non-radiation environment for use in pre-irradiation test checks.

SUMMARY:

At the end of Contract Year 1963, all of the required objectives and milestones established for the Radiation Effects Program were met.

Test Number 4

Test Date May, 1963

Test Item	Test No.	Location	Duration
Transition Joint	8/W001	Pool	1000 Mw-hr
Neutron Detectors	21/W002	Pool	Several Mw-hr
Neutron Detector Preamps & Cables	21/W006	Outside	300 Mw-hr
Accelerometers	23/W002	N.Dewar	300 Mw-hr
Resistance Thermometers	23/W102	E.Dewar	300 Mw-hr
Strain Gauges	23/W402	E.Dewar	300 Mw-hr
Pressure Transducers	23/W502	E.Dewar	300 Mw-hr
Electronic Components	31/W011	Outside	300 Mw-hr
Control Drum Bearing	40/W001/2	N.Dewar	300 Mw-hr
		E.Dewar	300 Mw-hr

GTR Test 4

Transition Piece and Pressurization System

Test Number 7a

Test Date August 26, 1963

<u>Test Item</u>	<u>Test No.</u>	<u>Location</u>	<u>Duration</u>
1. Pressure Transducers	23/W503	North	300 Mw-hr
2. Accelerometers	23/W004	North	300 Mw-hr
3. Resistance Thermometers	23/W104	West	300 Mw-hr
4. Strain Gauges	23/W404	West	300 Mw-hr
5. Linear Displacement Transducers	23/W203	West	300 Mw-hr
6. Thermocouples & Connectors	23/W302	West	300 Mw-hr
7. Cables	31/W012	Pool	300 Mw-hr
8. Control Drum Actuator Components	14/W001	East	300 Mw-hr

GTR Test 7a

Actuator Component Fixture Prior to Inserting in Dewar

Accelerometer Fixture Prior to
Inserting in Dewar

Transducer Assembly

Test Number 8 and 9

Test Date September 17-28, 1963

<u>Test Item</u>	<u>Test No.</u>	<u>Location</u>	<u>Duration</u>
1. Tie Rod Components	15/W201	North	450 Mw-hr
2. Tie Rod Materials	37/W001	East	450 Mw-hr
3. Actuator Components	14/W002	West	450 Mw-hr

GTR Test 8 and 9

Tie Rod Component

Tie Rod Material Pallet

Actuator Pallet

WELDING: HELI - ARC

Astronuclear

PBRF TESTS DURING CY63

<u>Test No.</u>	<u>Description</u>
23/W103	Resistance Thermometers
23/W301	Cu-Constantan Thermocouples
23/W403	Strain Gauges

VPI ENGINEERING COMPANY A DIVISION OF VPI GROUP, INC. (NASDAQ) NEW YORK, N. Y.	
<h1>CRYOGENIC IN-PILE TEST LOOP</h1> <h2>PROCESS FLOW DIAGRAM</h2>	
DESIGNED BY <u>A. MANDELL</u> APPROVED BY <u>[Signature]</u> APPROVED BY <u>RF D'AMICO 3163</u>	DATE <u>6-6-03</u> SCALE <u>NONE</u> DRAWING NO. <u>1462-M-1</u>

W-2 Water Cooled In-Pile Capsule,
General Arrangement

Experiment Sample Holder

NE-4200

FISCAL CONTROL

PURPOSE: "To provide for budget management, planning, analysis and control of the Nuclear Subsystem Development program."

FISCAL CONTROL OPERATION:

The function of the Fiscal Control Group is to provide for the budgeting, control and analysis of all direct costs in the program.

This function is required by Contract Statement of Work which in part requires that Fiscal Control shall prepare financial reports in the form of budgets and revisions for all tasks and sub-tasks and a report of actual costs for the same tasks and subtasks as the budget. The Cost Report is required to be submitted to the customer on a regularly scheduled basis.

Budget Planning and Analysis efforts took place during the CY '63 which resulted in several budget revisions.

Cost analyses and forecasts were prepared on a continuing basis as part of the Cost Control program.

An additional contract requirement in the cost area was met by the Fiscal Control Group. The Reliability Group at WANL must provide Aerojet-General a ratio of period and year-to-date reliability costs to the total costs for assembly, inspection, installation and test. So that the ratio may be obtained, the cost elements must be identified. This information appears as a supplement to the Cost Report submitted to Aerojet-General.

COMPUTER PROGRAM DEVELOPMENT:

Contract Year 1963 required considerably more NERVA tasks and subtasks than the previous Contract Year. Substantial reprogramming of the budget and cost collection codes was required as well as debugging effort both before and after October 1, 1962.

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WANL-TME-573

As a result of the KIWI tests in December, 1962, a major program redirection took place necessitating a completely revised Work Program and, in turn, a revised Budget. This effort took place during the second quarter of CY '63 and included a redefined reactor test program as well as accelerated and expanded component and fluid flow test programs.

During the third quarter of CY '63, a revised project numbering system evolved and is now applicable to the CY '64 budget and cost reporting program. It is significant to note that a considerable amount of computer reprogramming took place during the fourth quarter of CY '63 in preparing to implement this program in CY '64.

COST COLLECTION AND REPORTING:

The subtasks of the Work Program provide a means for defining details of the over-all program and also a means to collect top cost figures. Purchase Orders are written against specific subtask numbers and invoices collected and reported against these subtasks. All purchase orders are reviewed by Fiscal Control for consistency with the current Work Program and approved for costs prior to placement. A daily totaling of all purchase orders to date is maintained to provide instantaneous information of program commitments by subtask.

Semi-monthly inputs for a Cost Report are prepared. These consist of the Purchase Report and the invoices for the period.

The Purchase Report lists all Purchase Orders which have been placed during the period and the Accounting Report lists all invoices posted in the period.

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WANL-TME-573

After being appropriately and adequately checked for accuracy by Fiscal Control, the data are fed into the Computer Program as Current Period inputs. A Period Cost Report is then produced on the IBM 7094 computer.

All Cost Reports are thoroughly checked before release.

SUMMARY:

- (1) As a level of effort project, Fiscal Control during Contract Year 1963 has provided for budget management, planning, analysis and control of the Nuclear Subsystem Development Program. This activity is directly related to the individual projects making up the Nuclear Subsystem programs,
- (2) In addition, a revised operating budget was developed, defined and published during Contract Year 1963 reflecting the major Program Redirection which took place during the second quarter.

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- (3) The budget and cost requirements of CY '64 were analyzed and developed during the third and fourth quarters of the Contract Year.
- (4) A major effort at the end of the Contract Year was required to identify the closeout costs and commitments and to provide a smooth transition in the cost area from CY '63 to CY '64. This effort covered the CY '64 program and the CY '63 carry-overs. Close liaison was maintained with the Purchasing and Accounting Departments to ensure completeness of costs for CY '63.

NE-1360

DEVELOPMENT PARTS, FABRICATION AND PROCUREMENT

PURPOSE: "To provide the delivered component and subassembly to the configuration and schedule requirement for the reactor and component test programs."

CHRONOLOGY OF MAJOR PROCUREMENT EVENTS:

The manpower charging this project are responsible for the releases and changes associated with this hardware. These personnel must monitor and be fully cognizant of all phases of procurement, fabrication, assembly and inspection of this hardware. Over-all program potential problems are analyzed from a schedular and budget standpoint and possible solutions are investigated with appropriate activities. Detail hardware control schedules are maintained and procurement is redirected as required to meet schedular

requirements. All proposed design changes are analyzed for proper definition and effect on schedule and costs prior to approval.

Heavy design releases were made in the month of October, 1962 for the NRX-A1 and A2 reactors and the NRX-A1 lithium hydride shield. Outer reflector beryllium segments, beryllium control drums and beryllium shim rods were released. In the core area, the aluminum top support plate, graphite lateral support blocks, graphite tile pins, graphite liner tubes, tie rods, pyrographite insulating tile, washers, sleeves and cups were released. Lithium hydride shield components including tubes, powder, springs, discs, boronated steel, dome and nozzle end formings, inserts, and drum drive raw materials were released.

In November, 1962 initial releases for 95% of all NRX-A1 and A2 components had been completed.

The procurement was well underway for the NRX-A1 lithium hydride shield and the first lithium hydride material was received in December, 1962.

In January, 1963 long lead components for NRX-A3 were released. These components were similar to those of NRX-A1 and A2 released in October, 1963.

In February, 1963, the program was subjected to a major redirection. Immediately, additional raw material was released to cover the additional Phase I and Phase II Vibration Test and component testing requirements.

Heavy procurement activity continued following redirection of the program and in March, 1963 raw material was being received. Core support ring forgings were released, support plate machining was started and tie rod materials were ordered for the NRX-A1 and A2. Inner reflector graphite barrel machining started and filler strip raw material was ordered. The NRX-A3 core support plate forging was ordered and the outer reflector support ring forging, beryllium segment material and control drum material were ordered.

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WANL-TME-573

With the NRX-A1 and A2 raw material being received in March, machining orders were placed in April, 1963 for core support ring, cluster plates and inner reflector filler strips.

In April, 1963 the redesigned lateral support system design was released and changes to existing orders were made. The lateral support seal segment material was ordered at this time.

In April, 1963 detail releases for specification grade hardware for the component program were completed and orders placed for parts not already covered in previous NRX-A1 and A2 releases.

In May, 1963 all NRX-A1 raw material (except filler strips) had been received. At this same time a complete design release of instrumentation for NRX-A1 was made and procurement action initiated. By June of 1963, 90% of the instrumentation purchase orders were placed and in July, 1963 all instrumentation for NRX-A1 was on order.

Heavy deliveries of Phase I Vibration Test reactor hardware were made in July, 1963. Included in these deliveries were inner reflector cylinder, seal segments, seal springs and seal brackets. All outer reflector components were delivered at that time.

The NRX-A1 deliveries in July included cluster plate material, control drum cylinders and miscellaneous hardware and the shield completely modified for instrumentation.

In July design changes were released to the basic reactor hardware to incorporate final instrumentation requirements. Included were modification to the core support ring and the core support plate.

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WANL-TME-573

In August and September of 1963 the outstanding NRX-A1 hardware was delivered. Included in these deliveries were graphite barrel, interface seals, tie bolts, dome end support ring, beryllium sectors, core support plate, jet orifices, shim rods, aluminum barrel, seal segments, spring bearing ring, spring support ring, coated bottom support blocks, core support ring and core support plate.

In September, 1963 a major effort was initiated to satisfy the requirements for gold plated instrumentation leads. This change was made necessary by difficulties with the previously planned technique. This was necessary to provide a proper brazing bond between the lead and the seal block in the final routing at final assembly. This program was in operation at the end of the contract year.

The final filler strip assemblies for Phase I Vibration reactor were received and installed in early September.

As part of the NRX-A1 hardware deliveries, second sets of parts were completed and delivered in September. Included in these were coated bottom support blocks for component testing, core support plate for plugged channel test, orifices for plugged channel test and cluster plates for plugged channel test and PAX.

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NE-4100

NUCLEAR SUBSYSTEM PROGRAM MANAGEMENT AND CONTROL

PURPOSE: "To provide for the Management, Monitoring, and Control of the Nuclear Subsystem Development Program utilizing the PERT system for program control to ensure proper program coordination. In addition, to provide specification data on the Nuclear Subsystem and provide revisions to the NERVA Engine Model Specification and those for qualification, Engine PFRT and Qualification Tests."

Effort under Project 4100 consisted of three basic functions:

- a) PERT Operations
- b) Customer Reporting and Documentation
- c) Computer Program Development

PERT OPERATIONS

The PERT Operations Group provided planning and reporting for several major areas of the NERVA program. The more significant of these areas were the NRX-A1 and NRX-A2 Reactors, the Component Test Program, and the Safety Program. Major PERT planning networks were drafted and data processed through the computer for internal use on both the Phase I Vibration Reactor and the Plugged Core Reactor.

In its typical operation, the PERT Group drafted the original layouts of all networks. Conferences then were held with project sponsors to review the logic of the networks and to obtain initial time estimates. Final drafting was accomplished next, together with initial computer runs. Final reviews were held with project sponsors at which time responsibilities were assigned to particular individuals to provide updating information routinely to the PERT Group. Periodic reviews were held with Project Sponsors to correct the logic of the networks and to insure that the established responsibilities were being fulfilled.

A total of twenty-three (23) bi-weekly PERT reports were issued during the contract year. Three bi-weekly reports were omitted with the consent of the customer.

A PERT report consisted of distinct subsections which treated the NRX-A1, NRX-A2 Reactors, the Component Test Program and the Safety Program individually. Constraints however from one subsection to another were identified and fully integrated. Constraints such as Aerojet General Interface Gauges or Flight Safety Data Requirements also were identified as well as GFE constraints.

Each subsection of the PERT Report was divided further into eight (8) subsections. Four of these subsections were those commonly found in PERT Reports, that is:

- a) Sequenced by Critical Path
- b) Sequenced by Expected Date
- c) Sequenced by Event Number
- d) Sequenced by Organization

In addition to these, a listing of all punched cards which formed the input for a computer was furnished. Three other subsections furnished information on all activities completed, all time estimate changes and all logic changes respectively.

A PERT report in the early part of the contract year consisted of approximately 250 activities. PERT reporting in the final quarter of the contract year covered approximately 2000 activities. Information in the reports was current as of the day prior to issuance of the report.

CUSTOMER REPORTING AND DOCUMENTATION

Two other methods of status and progress reporting were used during contract year 1963 in addition to the bi-weekly PERT Report.

A weekly significant items report was published covering the overall program status and the Component Test Program. Late in the contract year, fuel development progress was highlighted in the weekly report.

Originally established as a TWX, the report evolved into a letter in order to safeguard classified information contained therein.

The contents of this report were reviewed for consistency with the PERT report and other schedule data and any inconsistencies were resolved.

Fifty-two (52) TWX reports were issued.

The NERVA Technical Progress Report was a narrative document published monthly. Each project reported on the technical progress which had been accomplished or major problems yet unresolved. The contents of this report also were reviewed for consistency with the PERT Report and the weekly significant items report and discrepancies were resolved.

The transmittal letter for the NERVA Technical Progress Report contained milestones status information. Twelve (12) Progress Reports were reviewed and edited.

The redirection in effort during the second quarter of the contract year required the publication of a new Work Program. WANL Document WANL-WP(J)-016 was issued March 1, 1963. This Work Program was compiled from drafts submitted by project sponsors. Each draft was reviewed thoroughly and discrepancies and omissions corrected. A second Work Program, WANL-PE(J)-017 dated May 17 was issued following negotiation of the March Work Program. Again each draft was reviewed with project sponsors before final publication.

COMPUTER PROGRAM DEVELOPMENT

A total of six (6) computer programs have been developed and debugged. Three of these programs provide schedule data which support the PERT Report.

The Provisioning List was the first attempt to utilize a computer to record detailed schedule data. A program was written which indicated demand dates and schedule performance measured against these demand dates. An advanced version of the Provisioning List program was developed for the Component Test Program. The principal differences from the Provisioning List program were its efficiency in sorting and categorizing information and its capacity to handle a large volume of detail. A single edition, which covers some 4000 lines of schedule information, can be updated within a 24 hour period. Updating takes place once per week.

The third bar chart computer program has been developed for the Irradiation

Test Program. Figure 1 shows the format of the output of this program. The top area of chart contains a time scale which is read vertically, the test identification and number, date of report, cognizant engineer, etc. The left side of the chart contains descriptions of events or activities required to complete the experiment. The right hand side of the chart contains line sequence numbers and test number. At least three other pages of schedules supplement this chart. Specimen hardware, test rig details, and documentation schedules are further broken-down as required by the cognizant engineer on these supplementary pages.

One of the significant improvements in this program is the capability of the computer to reschedule items according to pre-programmed instructions. If the test date is changed, all schedules are changed by the computer. Actual performance recorded, however, is not affected by such a change.

The line sequence numbers on the bar chart are identical with PERT event numbers which permits a simple correlation of the two reports.

The PERT network seen in Figure 2 is a simple diagram which matches the bar chart shown in Figure 1. The PERT had the advantage of showing interrelationship not only of the items scheduled for the test but also on a test-to-test basis.

The bar charts are reviewed by the cognizant engineers, changed to up-date information and forwarded to the Irradiation Test Group for review. After review by this group, the bar charts are forwarded to the PERT Group where the data is converted to PERT input and a PERT computer run is made. If the PERT run indicates that slippages will occur for subsequent tests, this information will be reviewed in a planning meeting. If new test dates are required, and agreed upon, the bar charts will be updated accordingly.

Three other computer programs were developed during the contract year. One program records milestones and their status. It is intended to use the output of this program as part of the monthly NERVA Technical Progress Report rather than discuss the status of the milestones in the transmittal letter.

IRRADIATION TEST PROGRAM PERT NET

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