~~CONFIDENTIAL~~

Westinghouse
Astronuclear
Laboratory

MASTER

SNPO-C

RECORD COPY

WANL-TME-2751
FEBRUARY 1971

DATA ITEM S-110

RADIATION EFFECTS ANALYSIS
REPORT
NUCLEAR SUBSYSTEM (U)

NOTICE
PORTIONS OF THIS REPORT ARE ILLEGIBLE. It
has been reproduced from the best available
copy to permit the broadest possible avail-
ability.

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

~~CONFIDENTIAL~~
~~RESTRICTED DATA~~
~~Atomic Energy Act - 1954~~

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

~~CONFIDENTIAL~~
~~RESTRICTED DATA~~

Westinghouse Electric Corporation

Industry & Defense Products

Astronuclear Laboratory

Box 10864
Pittsburgh Pennsylvania 15236
(412) 892 5600

DM:RAJ:0859

APR 20 1971

Mr. J. L. Dooling
Manager, Subcontracts
NERVA Rocket Operations
Aerojet Nuclear Systems Company
Post Office Box 13070
Sacramento, California 95813

Attention: Mr. P. P. Ventura

Subject: Transmittal of Data Item S-110, WANL-TME-2751, "Radiation Effects Analysis Report Nuclear Subsystem" (U)

Gentlemen:

Enclosed are six (6) copies of the subject data item.

Respectfully,

ORIGINAL SIGNED BY

R. F. Dickson, Manager
Program Management
NERVA Nuclear Subsystem

RAJ/cas

Enclosures: Six (6) copies of TME-2751

- cc: Mr. R. W. Schroeder, SNSO-C, w/1 enclosure
Mr. R. C. Wilke, SNSO-C, w/1 enclosure
Mr. C. L. Archer, SNSO-C, w/3 enclosures
Mail and Records, SNSO-C, w/1 enclosure ←
Mr. E. H. Schneider, SNSO, w/1 enclosure
Mr. W. R. Slivka, SNSO, w/2 enclosures
Mr. H. H. Hoffman, SNSO-C Resident, WANL, w/1 enclosure

WHEN SEPARATED FROM ENCLOSURES, LABEL THIS
DOCUMENT AS UNCLASSIFIED
(INSERT PROPER CLASSIFICATION)

~~CONFIDENTIAL~~
~~RESTRICTED DATA~~

252

~~CONFIDENTIAL~~

MASTER

Westinghouse Astronuclear Laboratory

NOTICE
This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Energy Research and Development Administration, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

Subcontract NP-1
CODE IDENT NO. 14683

WANL-TME-2751
FEBRUARY 1971

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

DATA ITEM S-110

RADIATION EFFECTS ANALYSIS REPORT NUCLEAR SUBSYSTEM (U)

NOTICE
PORTIONS OF THIS REPORT ARE ILLEGIBLE
has been introduced by the best available
copy to permit the broadest possible avail-
ability.

~~DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED~~

Approved by:
F.D. Retallick
F. D. Retallick, Manager
NERVA Engineering

SPECIAL REVIEW
DETERMINATION
Class: U

Reviewer DBW Class. U Date 4-16-82

EXCLUDED FROM AUTOMATIC DOWNGRADING
AND DECLASSIFICATION

INFORMATION CATEGORY
SECRET

[Signature] 30.7
AUTHORIZED CLASSIFIER DATE

DISTRIBUTION OF THIS DOCUMENT UNLIMITED

~~CONFIDENTIAL~~
~~RESTRICTED DATA~~
~~ADMINISTRATIVE USE ONLY~~

BLANK

TABLE OF CONTENTS

<u>Section</u>		<u>Page</u>
1.0	INTRODUCTION	1-1
1.1	SCOPE AND PURPOSE OF DATA ITEM	1-1
1.2	RELATIONSHIP OF DATA ITEM TO OVERALL RADIATION EFFECTS PROGRAM	1-2
2.0	NUCLEAR SUBSYSTEM	2-1
2.1	INTRODUCTION	2-1
3.0	FUEL ELEMENTS (Chapter not required for NSS PDR)	
4.0	CLUSTER HARDWARE (Chapter not required for NSS PDR)	
5.0	CORE PERIPHERY (Chapter not required for NSS PDR)	
6.0	SUPPORT PLATE AND PLENA	6-1
6.1	SUMMARY AND CONCLUSIONS	6-1
6.2	INTRODUCTION	6-1
6.2.1	Item Description	6-1
6.2.2	Location	6-1
6.2.3	Description of Sub-Components	6-1

TABLE OF CONTENTS (CONTINUED)

<u>Section</u>		<u>Page</u>
6.3	ENVIRONMENTAL AND OPERATIONAL REQUIREMENTS	6-2
6.3.1	Nuclear Environment	6-2
6.3.2	Thermal Environment	6-2
6.4	ANALYSIS	6-3
6.4.1	Aluminum 6061-T6	6-3
6.4.2	Aluminum 6061-T61	6-5
6.4.3	Aluminum 6061-T651	6-6
6.4.4	Aluminum Alloy 5086-H34	6-6
6.4.5	Aluminum (Special)	6-7
6.4.6	304 Stainless Steel	6-8
6.4.7	302 Stainless Steel	6-9
6.4.8	A-286 Stainless Steel	6-10
6.4.9	321 Stainless Steel	6-11
6.4.10	Component Systems Analysis	6-12
	List of References	6-30
	Radiation Effects Information Requests (REIRs)	6-32
	Reports (T119s) Responding to REIRs	6-36
7.0	INTERNAL SHIELD	7-1
7.1	SUMMARY AND CONCLUSIONS	7-1

TABLE OF CONTENTS (CONTINUED)

<u>Section</u>	<u>Page</u>
7.2 INTRODUCTION	7-1
7.2.1 Item Description	7-1
7.2.2 Location	7-1
7.2.3 Description of Sub-Components	7-2
7.3 ENVIRONMENTAL AND OPERATIONAL REQUIREMENTS	7-3
7.3.1 Nuclear	7-3
7.3.2 Thermal	7-3
7.4 ANALYSIS	7-3
7.4.1 BATH	7-3
7.4.2 Lead	7-5
7.4.3 Aluminum 6061-T6	7-6
7.4.4 Aluminum 6061-T61	7-6
7.4.5 Aluminum 6061-T651	7-7
7.4.6 304 Stainless Steel	7-8
7.4.7 A-286 Stainless Steel	7-8
7.4.8 Stainless Steel 18-8	7-9
7.4.9 Component Systems Analysis	7-10
List of References	7-18
Radiation Effects Information Requests (REIRs)	7-19
Reports (T-119s) Responding to REIRs	7-23

TABLE OF CONTENTS (CONTINUED)

<u>Section</u>		<u>Page</u>
8.0	REFLECTOR ASSEMBLY	8-1
8.1	SUMMARY AND CONCLUSIONS	8-1
8.2	INTRODUCTION	8-2
8.2.1	Item Description	8-2
8.2.2	Location	8-2
8.2.3	Sub-Component Description	8-2
8.3	ENVIRONMENTAL AND OPERATIONAL REQUIREMENTS	8-4
8.3.1	Nuclear Environment	8-4
8.3.2	Thermal Environment	8-4
8.4	ANALYSIS	8-5
8.4.1	Aluminum 6061-T6	8-5
8.4.2	Aluminum 6061-T61	8-6
8.4.3	Aluminum 6061-T652	8-7
8.4.4	Aluminum 6061-T4	8-8
8.4.5	Beryllium	8-8
8.4.6	Copper-Boron	8-13
8.4.7	Copper	8-14
8.4.8	304 Stainless Steel	8-14
8.4.9	440C Stainless Steel	8-15
8.4.10	A-286 Stainless Steel	8-16

TABLE OF CONTENTS (CONTINUED)

<u>Section</u>		<u>Page</u>
	8.4.11 321 Stainless Steel	8-17
	8.4.12 Graphite and Grafoil	8-17
	8.4.13 Component Systems Analysis	8-17
	List of References	8-33
	Radiation Effects Information Requests (REIRs)	8-38
	Reports (T-119s) Responding to REIRs	8-47
9.0	CONTROL DRUM DRIVE ASSEMBLY	9- 1
9.1	SUMMARY AND CONCLUSIONS	9-1
9.2	INTRODUCTION	9-1
9.3	ENVIRONMENTAL AND OPERATIONAL REQUIREMENTS	9-3
	9.3.1 Nuclear	9-3
	9.3.2 Thermal	9-4
9.4	ANALYSIS	9-4
	9.4.1 302 Stainless Steel	9-4
	9.4.2 440C Stainless Steel	9-5
	9.4.3 A-286 Stainless Steel	9-6
	9.4.4 Polyimide	9-7
	9.4.5 PSC Durock	9-8

TABLE OF CONTENTS (CONTINUED)

<u>Section</u>		<u>Page</u>
	9.4.6 Copper Wire	9-8
	9.4.7 Silicon Iron	9-9
	9.4.8 Alnico V	9-9
	9.4.9 Diallyl Phthalate	9-10
	9.4.10 Relay 5	9-10
	9.4.11 Brass with Gold Plate	9-11
	9.4.12 Shrinkable Tubing	9-11
	9.4.13 System or Component Data and Experience	9-12
	References	9-28
	Radiation Effects Information Requests (REIRs)	9-31
	Reports (T-119s) Responding to REIRs	9-36
10.0	SUPPORT STEM COOLANT VALVE	10-1
10.1	SUMMARY AND CONCLUSIONS	10-1
10.2	INTRODUCTION	10-1
10.3	ENVIRONMENTAL AND OPERATIONAL REQUIREMENTS	10-2
10.4	ANALYSIS	10-3
	Radiation Effects Information Requests (REIRs)	10-7
	Reports (T-119s) Responding to REIRs	10-8

TABLE OF CONTENTS (CONTINUED)

<u>Section</u>		<u>Page</u>
11.0	NSS INSTRUMENTATION	11-1
11.1	RESISTANCE TEMPERATURE THERMOMETER	11-1
11.1.1	Summary and Conclusion	11-1
11.1.2	Introduction	11-1
11.1.3	Environmental and Operational Requirements	11-2
11.1.4	Analysis	11-4
	References	11-13
	Radiation Effects Information Requests (REIRs)	11-16
	Reports (T-119s) Responding to REIRs	11-20
11.2	LOW TEMPERATURE THERMOCOUPLES	11-21
11.2.1	Summary and Conclusions	11-21
11.2.2	Introduction	11-21
11.2.3	Environmental and Operational Requirements	11-23
11.2.4	Analysis	11-24
	References	11-31

TABLE OF CONTENTS (CONTINUED)

<u>Section</u>	<u>Page</u>
11.3 MEDIUM TEMPERATURE THERMOCOUPLES	11-34
11.3.1 Summary and Conclusions	11-34
11.3.2 Introduction	11-34
11.3.3 Environmental and Operational Requirements	11-36
11.3.4 Analysis	11-37
References	11-52
Radiation Effects Information Requests (REIRs)	11-56
Reports (T-119s) Responding to REIRs	11-60
11.4 HIGH TEMPERATURE THERMOCOUPLES	11-61
11.4.1 Summary and Conclusions	11-61
11.4.2 Introduction	11-61
11.4.3 Environmental and Operational Requirements	11-62
11.4.4 Analysis	11-64
References	11-73
Radiation Effects Information Requests (REIRs)	11-75
Reports (T-119s) Responding to REIRs	11-77
12.0 PROPELLANTS	12-1
(Chapter not required for NSS PDR)	

LIST OF ILLUSTRATIONS

<u>Figure</u>		<u>Page</u>
1-1	Data Item and its Relation to Radiation Effects Program	1-3
6-1	AL6061-T6 NERVA Test Experience	6-26
6-2	Effects of Irradiation on Room-Temperature Yield Strength and Ductility of Type 304 Stainless Steel	6-27
6-3	Effect of Test Temperature on the Strength of Irradiated Annealed and 25 Percent Cold-Worked Type 304 Stainless Steel	6-28
6-4	Room-Temperature Tensile Properties of Irradiated Type 304 Stainless Steel as a Function of Irradiation Temperature—Specimens were irradiated to a fast fluence of $7 \times 10^{20} \text{ n/cm}^2$.	6-29
7-1	Thermal Conductivity of Irradiated BATH Material	7-17
8-1	Swelling of Beryllium Irradiated to a Fast Fluence of $1 \times 10^{21} \text{ n/cm}^2$ at a Temperature of 70°C as a Function of Annealing Time and Temperature.	8-28
8-2	Swelling Threshold Temperature of Beryllium as a Function of Gas Content and Fluence	8-28
8-3	Calculated Change in Electrical and Thermal Conductivity of Impure Beryllium Upon Irradiation to a Fast Fluence of 10^{19} n/cm^2	8-29
8-4	Changes in Yield Strength, Ductility, and Hardness of Beryllium Irradiated at Fast Fluences and a Temperature Range of 75 to 100°C . Tested at Room Temperature	8-30
8-5	Thermal Conductivity of Unirradiated Beryllium S-200 in Both Perpendicular and Parallel Directions	8-31
8-6	Annealing Curves for Irradiated Beryllium	8-32
9-1	Diagram of CDDA in Relation to NSS	9-27
11.1-1	Reactor Temperature Measurement Locations	11-10
11.1-2	NRX-A6 EP-III A Reflector Inlet Plenum RTT's Reflector P. V. Annulus 936B363H01	11-11
11.1-3	Platinum Sensing Element RTT Maximum Radiation Test Levels In Terms of Percent of the Expected NERVA Radiation Levels	11-12

LIST OF ILLUSTRATIONS (CONTINUED)

<u>Figure</u>	<u>Page</u>
11.2-1 Reactor Temperature Measurement Locations	11-30
11.3-1 Reactor Temperature Measurement Locations	11-48
11.3-2 Temperature Deviation vs Radiation Exposure: T/C 2 (CrAl, CCC T2951-2, S/N 53243)	11-49
11.3-3 Temperature Deviation vs Radiation Exposure: T/C 3 (CrAl, CCC T2951-2, S/N 53244)	11-50
11.3-4 963B361 Chromel-Alumel Thermocouple NRX-A6 Radiation Levels in Terms of Percent of Expected NERVA Radiation Levels	11-51
11.3-5 978D186 Chromel-Alumel Thermocouple NRX-A6 Radiation Levels in Terms of Percent of Expected NERVA Radiation Levels	11-51
11.4-1 Cluster Exit Gas Candidate Thermocouple Design Profile	11-70
11.4-2 Reactor Temperature Measurement Locations	11-71
11.4-3 Deviation from Initial Calibration in Degrees Rankine of a W/W- 25 Re Thermocouple as a Function of Temperature after 0.5, 1, 3, 6 Months at 10^{14} nv (thermal) and also the Deviation of a W - 1 Os/W - 25 Re Thermocouple after 6 Months at 10^{14} nv (thermal). (from Reference 13)	11-72 11-72

LIST OF TABLES

<u>Table</u>		<u>Page</u>
6-1	List of Materials (940J449)	6-13
6-2	Materials Summary–Support Plate and Plena	6-15
6-3	The 3σ Maximum Radiation Environment for Materials in Support Plate ECC	6-16
6-4	Summary of Support Plate and Plena Component Material Temperatures and Other Key Performance Parameters (Graphite Core)	6-17
6-5	Effect of Irradiation on Mechanical Properties of Aluminum Alloys	6-19
6-6	Comparison of Properties of Aluminum Alloys	6-21
6-7	Effects of Irradiation on 2219 Aluminum Alloy 2219 T6	6-22
6-8	Effects of Irradiation on 6061–T6 Alloy	6-23
6-9	In-Reactor Creep of Annealed Type 304 Stainless Steel	6-24
6-10	Minimum Creep Rate of Type 304 Stainless Steel at 650 C ^(a)	6-24
6-11	Stress–Rupture Ductility of Type–304 Stainless Steel at 650 C	6-25
7-1	List of Materials (939J741)	7-11
7-2	Materials Summary – Internal Shield	7-12
7-3	The 3σ Maximum Radiation Environment for Materials in Internal Shield ECC	7-13
7-4	Summary of Shield Component Material Temperatures and other Key Performance Parameters (Graphite Core)	7-14
7-5	Mechanical Properties of Irradiated Bath Materials	7-16
8-1	List of Materials (939J733)	8-18
8-2	Reflector Assembly Material Summary	8-19
8-3	The 3σ Maximum Radiation Environment for Materials in Reflector ECC	8-20
8-4	Summary of Reflector Assembly Component Material Temperatures and Other Key Performance Parameters (Graphite Core)	8-21

LIST OF TABLES (CONTINUED)

<u>Table</u>		<u>Page</u>
8-5	GTR-18 Beryllium Data Irradiated at 140°R to Approximately $3 \times 10^{17} \text{ n/cm}^2$	8-23
8-6	GTR-18 Beryllium Data - Pressing z	8-24
8-7	Plane Strain Fracture Toughness (K_{IC}) of Irradiated Beryllium	8-25
8-8	Tensile Properties of Irradiated and Unirradiated Copper	8-26
8-9	Results of Irradiation at 125° F on Mechanical Properties of Cu24% B ¹⁰	8-27
9-1	Drum Actuator and AMOD Parts List	9-14
9-2	Material Summary	9-18
9-3	Summary of Radiation Effects on Actuator Materials	9-19
10-1	(Refer to Table 9-1, page 9-14)	10-4
10-2	(Refer to Table 9-2, page 9-18)	10-4
11.1-1	Transducer Test Survey, NERVA Technology Program	11-18
11.2-1	Tranducer Test Survey, NERVA Technology Program	11-28
11.3-1	NERVA Test Survey, NERVA Technology Program	11-45
11.3-2	Reactor Location and Nuclear Environment, 963B361 and 978D186 Chromel-Alumel Thermocouples	11-47
11.4-1	Calculated Compositions of W and W - 25 Re After Exposure to a $10(14) \text{ n/cm}^2 \text{ - sec}$ Flux	11-68
11.4-2	Radiation and Temperature Environment, NRX-A6 NAD XE Reactor Tests and NERVA Reactor	11-69

1.0 INTRODUCTION

1.1 PURPOSE AND SCOPE

The purpose of this data item is to investigate and document the effects of nuclear radiation on the materials and components of the Nuclear Subsystem. The materials used in the NSS Engineering Critical Components (ECC) are evaluated to determine if significant mechanical/physical property changes can be expected to occur when subjected to the anticipated thermal and radiation environments for the specific ECC applications. This evaluation is conducted by rigorously examining data from the NERVA program and other nuclear programs and correlating the findings with the expected NERVA ECC environmental conditions. This correlation then provides a basis for determining whether the material may be employed without radiation-induced performance degradation. There is no attempt to evaluate the effects of other environmental conditions.

In those instances where there is insufficient data available to make the proper correlation, Radiation Effects Information Requests (REIRs) are developed and define what data is needed to support the design and analysis effort.

This data item contains separate chapters for the Nuclear Subsystem and each of its Engineering Critical Components. In addition, there is a separate chapter, covering propellants. The NSS analysis applies to the NSS configuration described in WANL data item E-105. Each of the ECC chapters contains the following type of information:

1. a brief description of the components, their location within the Nuclear Subsystem and an identification of each of the materials it contains.
2. a definition of the environmental and operational requirements for each material.
3. an analysis of irradiated materials property data for each ECC material.
4. conclusions regarding the acceptability of each material for the intended applications or the need for additional testing.

5. a compilation of supporting data for each ECC (tables, figures, references).
6. Radiation Effects Information Requests for specific materials properties data needed for the ECC.

1.2 RELATION TO OVERALL RADIATION EFFECTS PROGRAM

The Radiation Effects Analysis Report provides a basic input to the overall Radiation Effects Program. It represents an integration of existing knowledge on radiation effects on ECC components and materials which have been considered in the design selection process. Where there is insufficient data to permit making an assessment for each material or component application, the REIRs describe specific data needed for design and analysis purposes.

Figure 1-1 depicts the S-110 Data Item and its relation to the overall Radiation Effects Program. Presently, design data which assesses the effects of radiation on materials or components is provided by DRMs. These DRMs are being provided as an appendix to Data Item E-105. DRMs are estimates which are based on experimental data and engineering judgment. The Radiation Effects Program is designed to update this data or otherwise verify it to meet the requirements of SNPO-C-1 or TD 69-28.

The S-110 analysis integrates the documentation and rationale which has been provided by Materials, Design Engineering and Radiation Effects people in a central document. Either the design is substantiated or REIRs are issued. These REIRs form the basis for a Radiation Effects testing/analysis program which feeds data, through the Materials Department, back into the design process. As the data item matures, it will provide justification for each material and component in the NSS to meet the projected radiation environment.

Figure 1-1. Data Item and its Relation to Radiation Effects Program

2.0 NUCLEAR SUBSYSTEM

2.1 INTRODUCTION

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

The subsystem includes engineering critical components as defined in the NERVA Engine Specification Tree, SNPO-C-ST-1, and the NSS Specification, CP-677555. These include the Fuel Elements, Cluster Hardware, Core Periphery, Support Plate and Plena, Internal Shield, Reflector Assembly, Control Drum Drive Assembly, Structural Support Coolant Assembly (Valve and Actuator) and NSS Instrumentation. A separate chapter is provided for each component and an additional chapter is provided for Propellants. The analysis of each ECC is at varying levels of maturity depending upon the relative maturity of the design.

Several ECCs utilize common materials, and many materials are used for several applications in a given ECC. The analysis provided herein has focused on the most critical application of each material and is conservative with respect to other applications.

As used in this analysis, Start of Life (SOL) is defined as the unirradiated condition, although the definition in other data items includes the entire first cycle of operation. Similarly, End of Life (EOL) is defined as the last ten minute cycle of the 60 cycles and 600 minutes of operation. The analysis has attempted to provide sufficient information for SOL and EOL design conditions. In some cases, they are identical; in other cases, they are different. In the case of a ductile material, strength generally increases or remains relatively unchanged with increasing radiation. Thus, SOL design conditions are utilized. However, some screening tests are required to verify that the material remains ductile with irradiation to NSS fluences and that strength is not significantly decreased. For example, screening tests for various aluminum alloys and stainless steels are recommended for this purpose.

The case for brittle materials is different. Mechanical property data required for brittle materials depends upon the material itself, the application and the environment to which it is subjected. Either ultimate tensile strength (including its application in Weibull analysis), fracture toughness or fracture strain may be the required property. If these properties degrade with radiation, EOL design must be utilized. Whether EOL conditions require an evaluation of the one-hour or ten-hour irradiation data will depend upon annealing. If nearly complete annealing occurs during cooldown or storage, one-hour data is applicable. If not, ten-hour analysis is required.

Other design parameters, such as thermophysical properties, must be analyzed in the same manner.

Since the NSS is being designed for a variety of missions, some simplifying assumptions are required regarding firing times and cooldown periods. From the standpoint of radiation induced degradation, it has been assumed that the worst case will either be the ten-hour irradiation or the one-hour irradiation followed by 30 hours of cooldown. Several shorter burns (not exceeding a total of one hour) would cause less radiation damage than the one-hour firing, even though the cooldown periods may be relatively short. Since there is no mission requiring a burn exceeding one hour, and since the propellant supply in the ground or flight configuration will not permit burns exceeding one hour, this assumption is reasonable. Each component would experience temperatures close to the maximums specified in Section 3.2 of each chapter provided in this data item. Thus, the annealing of radiation-induced materials changes, which are associated with this 30 hour temperature profile, is assumed to occur.

The Radiation Effects testing program will be designed on the basis of Radiation Effects Information Requests (REIRs). Since available facilities do not always have fluence and temperature capabilities to match requirements, a Program Plan is constructed annually to reconcile requirements with capabilities.

Some materials will experience a variety of temperature and fluences in various applications. Consequently, WANL has adopted a testing philosophy which takes these

conditions into account. Materials are irradiated over a range of fluences which usually includes the design fluence or the maximum facility capability. Similarly, irradiation and testing is performed over a temperature range which includes the present design temperature range. Annealing tests are performed in some cases to determine if one or ten-hour irradiation fluences are applicable.

Although materials testing is recommended for the structural or structurally important ECCs, component level testing is recommended for actuators and various transducers. For example, most of the CDDA materials can be expected to withstand the environmental conditions projected. However, component level tests of the motor, potentiometer and complete actuator are recommended to verify this contention and provide data which may be applied to the establishment of the reliability of that component. Similarly, the primary difficulty observed in NSS transducers has not been material degradation. Rather, it has generally involved a drift or change in transducer output which can most likely be investigated by testing the entire transducer. Thus, tests must be statistically designed both to determine radiation effects on transducers and to provide data which may be applied to the establishment of the reliability of that transducer.

It should be noted that the REIRs only document requirements. The mechanisms employed to fulfill these requirements require additional test planning or analysis before actual testing can be conducted. The results of this planning are documented in test specifications which inform the test facility of WANL requirements.

Each chapter for each ECC presents a summary and conclusions which interpret the significance of the data presented and specifies where additional testing or analysis is required.

CHAPTER 3

FUEL ELEMENTS

(Chapter not required for NSS PDR)

CHAPTER 4

CLUSTER HARDWARE

(Chapter not required for NSS PDR)

CHAPTER 5

CORE PERIPHERY

(Chapter not required for NSS PDR)

6.0 SUPPORT PLATE AND PLENA

6.1 SUMMARY AND CONCLUSIONS

The applications of aluminum 6061 (T6, T61, T651) aluminum 5086-H-34, SS-304, SS-302, A-286 and SS-321 in the NSS Support Plate and Plena (SPP) and associated hardware has been reviewed. Aluminum alloys are known to exhibit increased strength and decreased ductility at NSS SPP fluences and temperatures. A possible exception is the short transverse forging direction at cryogenic temperatures and tests are planned to further investigate this problem. Thus, Start-of-Life (SOL) or unirradiated design conditions are applicable based on radiation effects analysis with the exception noted above. No additional tests are required.

The stainless steels used in this ECC are also known to undergo a relatively small change in strength and decreased ductility. Thus SOL design conditions will also be applicable in these cases. However, stress rupture and creep above room temperature require investigation due to observations made by non-NERVA sources on this problem. Similarly, SS-302 springs are known to undergo stress relaxation and additional tests are planned to measure this effect. All other material/applications are considered satisfactory in the NSS SPP without additional data.

6.2 INTRODUCTION

6.2.1 Item Description

Functionally the Support Plate and Plena (SPP) support the reactor core, distribute and collect stem and by-pass flow, and provide a mechanism for separating the turbine inlet from the turbine outlet flow within the NSS. Additionally, the SPP must carry the structural loads imposed on it by the Internal Shield.

6.2.2 Location

The aft face of the SPP is located approximately 3.5 inches from the forward end of the core.

6.2.3 Description of Sub-Components

There are four major sub-components of the SPP listed below. A more detailed description of the SPP can be found in E-105. The bill of materials for the support plate and

plena has been extracted from the Reference Drawings (940J449 and 939J725) and presented in Table 6-1. Table 6-2 summarizes the materials used in this component by generic names.

6.2.3.1

The core support plate contains three integral plena. These plena are the SSCA bypass flow plenum, a support stem inlet flow plenum and a plenum which combines the bypass flow with the support stem effluent.

6.2.3.2

The SSCA feed and bypass system includes the inlet torus, inlet tubing, bypass torus and bypass tubing.

6.2.3.3

The piping cylinder keeps the flow to the turbine inlet from the discharge flow, transmits the axial load from the internal shield to the support plate and supports the peripheral shield.

6.2.3.4

The Central Dome separates the turbine inlet from the discharge flow and mates with the forward flange of the piping cylinder.

6.3 ENVIRONMENTAL AND OPERATIONAL REQUIREMENTS

6.3.1 Nuclear Environment

Table 6-3 presents the maximum gamma and neutron radiation (plus 3 sigma) environments experienced by the materials in the SPP. The radiation levels presented in this data item, represent the most conservative application of each material in the SPP since they are the highest experienced by each material. The radiation levels for both the composite and graphite cores are given. The more conservative levels associated with the composite core are used in the analysis.

6.3.2 Thermal Environment

Table 6-4 presents the temperature ranges (including uncertainties) experienced by each material during typical operation modes, cooldown and storage.

6.4 ANALYSIS

Each of the materials listed in Table 6-2 is analyzed in this section for their most critical application(s) in the SPP.

6.4.1 Aluminum 6061-T6

Aluminum 6061-T6 is used in several parts of the Support Plate and Plena as listed in Table 6-1, which are not structurally critical to SPP design.

6.4.1.1 Non-NERVA Data

Aluminum alloys have been used extensively as structural components in water cooled, plutonium producing materials testing and research reactors.⁽¹⁾ The applications have been at temperatures exceeding the NERVA operating temperatures but in the same range as those experienced in the SPP during cool-down (Table 6-4). Neutron irradiation is known to alter the mechanical properties of aluminum, but to a lesser degree than is true for stainless steels or the refractory metals.⁽¹⁾ Those property changes have been demonstrated to be primarily related to fast fluence.⁽²⁾

At the NERVA fluences specified in Table 6-3, very little change has been observed in aluminum properties. For example, an extensive irradiation program conducted at Oak Ridge on aluminum 6061-T6 (age-hardened condition) showed only small changes in tensile properties as a result of neutron irradiation to fast fluences of 1×10^{19} n/cm² ($E > 1$ Mev).⁽³⁾ These irradiations were carried out at temperatures below 570°R and all damage was annealed out after 30 minutes at 850°R.

Test results on several aluminum alloys are summarized in Table 6-5 which has been compiled in REIC Report No. 45.⁽⁴⁾ The results show a consistent trend in the irradiated properties of the aluminum alloys which have been investigated. At all fluences and test temperatures, aluminum alloys show increased strength and decreased ductility following irradiation. In some rare instances, there was no change observed. The magnitude of the changes tend to vary from alloy to alloy and are summarized in Table 6-6.

Other investigations have shown greater changes in mechanical properties at

higher fluences. However, the tendency observed in all data has been for increased strength and decreased ductility following irradiation.

6.4.1.2 NERVA Data

Aluminum 2219 has been extensively investigated for applications in NERVA and the data are summarized in Table 6-7.⁽⁵⁾ Expected increases in strength and decreases in ductility were observed. Table 6-8 summarizes NERVA testing experience for Al-6061-T6 with the same trend in results.⁽⁵⁾ Irradiation and testing for both aluminum alloys (Al 2219 and Al 6061) were conducted from 140°R to 1,000°R. The temperatures and fluences at which these data were gathered for Aluminum 6061-T6 are compared diagrammatically to NERVA design conditions (for one and ten hour conditions) in Figure 6-1. Similar results were seen by Schwanbeck in NERVA studies conducted at Plum Brook⁽²⁵⁾ and were reported in a summary article for NERVA material by Lombardo et al.⁽²⁶⁾

Estimates of the unirradiated and irradiated properties of Aluminum 6061-T6 bar and tube have been made in DRM 51294⁽⁶⁾ for 10 minute and 60 minute irradiations. These data have been plotted for study, although they have not been included in this report. As has been observed in prior investigations of aluminum 6061-T6 and other Aluminum alloys, the estimates depict small increases in strength and decreases in ductility following irradiation. There is no attempt to estimate ten hour properties because the SPP reaches a maximum temperature of 750°R during cooldown following a one-hour firing and nearly complete annealing occurs in aluminum at that temperature. WANL has classified this material as ductile under all conditions of use in the NSS SPP.⁽⁷⁾

6.4.1.3 Conclusions

Aluminum 6061-T6 has been demonstrated to remain ductile over all temperatures and fluences experienced in the NSS SPP. Thus, start of life (SOL) design conditions are applicable. The slight increases in strength following irradiation are not considered for design purposes. In addition, there are only minimal structural loads applied to this material as utilized in the SPP. Thus, no additional Radiation Effects testing is required and no REIRs have been issued.

6.4.2 Aluminum 6061-T61

This alloy is used in the SPP as a 6-inch thick aluminum plate forging, 44.5 inches in diameter, which is internally machined to form three separate plena. The forward plenum supplies flow to each core support stem. The central and aft plena distribute flow. The support plate transmits the core loads to the reflector barrel and thence to the engine load path. Other applications of aluminum 6061-T61 plate forging include the piping cylinder which supports the internal shield and several other less crucial ones. In addition, aluminum 6061-T61 ring forging material is used for the plenum closure ring.

6.4.2.1 Non-NERVA Data

The discussion of Section 6.4.1.1 on aluminum 6061-T6 is applicable. There is no irradiated non-NERVA data for this alloy. However, at NERVA fluences, all evidence suggests the same invariable increased strength and decreased ductility observed in aluminum alloys.

6.4.2.2 NERVA Data

There have been no experimental investigations of aluminum 6061-T61 in the NERVA program. However, as discussed in 6.4.1.2, similar aluminum alloys exhibit increased strength and decreased ductility at NERVA fluences.

Estimates of the unirradiated and irradiated properties of Aluminum 6061-T61 have been made in DRM 51276 (ring forging)⁽⁸⁾ and DRM 53161 (plate forging)⁽⁹⁾ for 10 and 60 minute irradiations. These data have been plotted for study, although they have not been included in this report. As is true for all aluminum alloys, these estimates depict small increases in strength and decreases in ductility following irradiation. There is no attempt to estimate ten hour properties because the SPP reaches a maximum temperature of 750°R during cooldown following a one hour firing and nearly complete annealing is expected to occur in aluminum at that temperature. WANL has classified this material as ductile under all conditions of use in the NSS except in the case where the support plate may be brittle in the short transverse direction.⁽⁷⁾

6.4.2.3 Conclusions

Aluminum 6061-T61 will likely remain ductile (by SNPO-C-1 definition), over all temperatures and fluences experienced in the NSS, based on an evaluation of the other aluminum alloys. Thus, Start of Life (SOL) design conditions are applicable and the slight increases in strength following irradiation need not be considered for design purposes. A possible exception is the potential embrittlement cited above for the transverse direction. Thus, screening tests are planned on aluminum 6061-T651 plate planned for the Internal Shield to verify this (REIR No. 9) contention. In addition, a similar test is planned for the barrel forging (T-61) planned for the reflector assembly (REIR No. 11). The results of these tests will be applicable to the aluminum 6061-T61 in the SPP.

No additional consideration of radiation effects in Al 6061-T61 is required for SPP design and no REIRs have been issued except those noted above.

6.4.3 Aluminum 6061-T651

This material is used for non-structural applications in the SPP such as rings and sleeves.

6.4.3.1 Non-NERVA Data

The discussion of 6.4.1.1 on aluminum 6061-T6 is applicable. There is no additional non-NERVA data.

6.4.3.2 NERVA Data

There has been no experimental investigation of this material form in the NERVA program. However, its properties have been estimated⁽¹⁰⁾ and the predicted increased strength and decreased ductility is in agreement with all other aluminum alloys.

6.4.3.3 Conclusions

Both because of the relatively minor applications of aluminum 6061-T651 in the NSS SPP and because SOL design conditions are applicable, its irradiated properties need no further investigation.

6.4.4 Aluminum Alloy 5086-H34

This material will be utilized for piping in the SPP which requires welded fabrication.

6.4.4.1 Non-NERVA Data

The discussion of Section 6.4.1.1 on Aluminum 6061-T6 is applicable. There is no irradiated non-NERVA data available for this alloy. However, at NERVA fluences, all evidence suggests the same invariable increased strength and decreased ductility observed in aluminum alloys. It has generally been observed that the properties of welds, in aluminum and other alloys, may not be identical to the base metal in both the irradiated and non-irradiated condition.

6.4.4.2 NERVA Data

There has been no experimental investigation of this alloy in the NERVA program, although the invariable increased strength and decreased ductility following irradiation is suggested from related NERVA data. Similarly, there are no current estimates of its properties available. However, it has been suggested that aluminum pipe welds will be brittle or become brittle following irradiation at cryogenic temperatures.⁽⁷⁾ Although this is not anticipated for 5086, some experimental verification is desirable.

6.4.4.3 Conclusion

Increased strength and decreased ductility are expected following irradiation of aluminum 5086-H34. However, an REIR, No. 6 has been issued to measure the effects of radiation in the weldments of pipe made from this material. This REIR will only require screening this material because its use as pipe will not involve the transmission of loads. Similarly, the primary requirement is for an estimate of its irradiated ductility to allow the use of SOL design.

6.4.5 Aluminum (Special)

Additionally, aluminum is planned for some special applications at elbows and expansion and reduction junctures. Both because of the application and because of the increased strength observed in aluminum following irradiation, no additional analysis or testing is required.

6.4.6 304 Stainless Steel (annealed)

This material is employed in the annealed condition in the SPP for bolts, caps, locking device, etc. These applications are important in maintaining the integrity of this component and degradation due to irradiation would be unacceptable.

6.4.6.1 Non-NERVA Technology

The austentic stainless steels have found wide application in the nuclear industry because of their relative resistance to radiation-induced degradation. Of all the stainless steels, AISI 304 has probably been the most extensively investigated.

Figure 6-2 illustrates the effects of irradiation on strength and ductility at room temperature at 10^{20} n/cm² ($E > 1$ Mev) and above.⁽¹¹⁾ Figure 6-3 shows the increase in strength in irradiated SS 304, in both the annealed and cold-worked form, at room temperature and above.⁽¹²⁾ At fluences closer to the NERVA application in the SPP, the yield stress of irradiated and unirradiated SS 304, as well as elongation are shown at room temperature and above in Figure 6-4.⁽²⁴⁾ At all temperatures investigated, the irradiation effects on mechanical properties are characterized by minor increases in strength and reduction in ductility. Although the radiation-induced reduction in ductility tends to be severe,^(12, 13) it does not appear to cause SS 304 to become brittle at NSS SPP fluences and temperatures.

The results of creep and stress rupture tests at Battelle⁽¹⁴⁾ and Oak Ridge^(15, 16) are summarized in Tables 6-9, 6-10, and 6-11. Although these tests are at higher temperatures than the NERVA application, it is interesting to note the higher creep rates following irradiation. Similarly, fatigue life of SS 304 appears to be reduced at elevated temperatures.⁽¹⁷⁾ Due to the lower NSS SPP fluences and temperatures, stress rupture and creep will probably not represent a design constraint.

6.4.6.2 NERVA Data

There is no directly applicable NERVA data for SS 304. Estimates of the properties of unirradiated 304 stainless steel have been made for NSS PDR.⁽⁸⁾ These data show an increased strength and decreased ductility over the NERVA range of temperatures

and fluences. Similarly, SS 304 is shown to remain ductile in the WANL material classification.⁽⁷⁾

SS 304 has been used for a variety of applications in the KIWI, NRX, Phoebus and XE reactor test series without any apparent failure traced to this material.

6.4.6.3 Conclusions

Non-NERVA experimental data show slight increases in strength and significant decreases in ductility of SS 304 following irradiation. There are no NERVA experimental data although PDR estimates substantiate the ductility of SS 304 at NERVA conditions. Thus, SOL design is applicable and the radiation-induced increases in strength are not considered in design.

Since there is some evidence of radiation-induced changes in creep and stress rupture properties of SS 304 at elevated temperatures, an REIR, No. 7, is issued for these properties at NERVA conditions.

6.4.7 302 Stainless Steel (1/4 hard and 1/2 hard)

This material is used for springs in the 1/4 hard condition as well as in some miscellaneous applications in the NSS SPP. The application for springs is the more important of the two although both are reviewed in this section.

6.4.7.1 Non-NERVA Technology

There has been very little radiation effects work conducted for SS 302. However, it is nearly identical to SS 304, differing only in carbon content. The properties of SS 304 were reviewed in 6.4.6 and are applicable to the uses of SS 302 in the SPP excepting where this material is used in spring applications.

Work on stress relaxation measurements in 304 stainless steel has been performed by J. W. Joseph⁽¹⁹⁾ and R. E. Schrieber⁽²⁰⁾ and is applicable to the use of SS 302 in springs. Joseph found that stainless steels stressed at 20,000 to 39,000 psi at a temperature less than 650°R underwent as much as 50% stress relaxation during exposure to 9×10^{20} n/cm² ($E > 1$ Mev). The 50% reduction is also observed for specimens stressed to

20,000 psi after exposure to $4 \times 10^{20} \text{ n/cm}^2$ ($E > 1 \text{ Mev}$). Schreiber concluded, as did Joseph, that marked relaxation occurs at low temperatures.

6.4.7.2 NERVA Data

No irradiation experiments have been performed on SS 304 in the NERVA program. However, a limited number of Inconel - 718 springs were irradiated to $3.5 \times 10^{17} \text{ n/cm}^2$ ($E > 1 \text{ Mev}$) in GTR 20C⁽²¹⁾. Significant stress relaxation occurred and both cryogenic exposure and radiation are seen to enhance this phenomenon. Annealing for one hour at 1000°R caused approximately 50% recovery. In addition, SS 302 springs were irradiated in GTR-16 and measurable stress relaxation was observed⁽²⁷⁾.

A variety of springs have been utilized in past NERVA reactors without apparent deleterious effects. In addition, an comprehensive discussion of spring materials for NERVA Reactor Applications has been performed.⁽²²⁾ It concludes that the austenitic stainless steel, and specifically SS 302, are particularly suitable for SPP springs.

6.4.7.3 Conclusions

Non-spring applications of SS 302 in the SPP will not be affected by irradiation. As is true for SS 304, the material will remain ductile and SOL design criteria are applicable.

When used in springs, stress relaxation will undoubtedly occur. The magnitude of this effect under NSS design conditions is unknown and REIR No. 8 has been issued for this purpose. It should be noted that pre-setting springs can compensate for the radiation induced stress relaxation when the magnitude of the effect is known.

6.4.8 A-286

This material is used in the SPP for various bolts, screws and studs.

6.4.8.1 Non-NERVA Data

An extensive compilation of non-NERVA data for A-286 will be presented in Chapter 4, Cluster Hardware, which represents its most important application in the NSS. In summary, A-286 remains ductile following irradiation and slight changes (increases and decreases) in strength have been observed.

6.4.8.2 NERVA Data

An extensive Radiation Effects program for A-286 is currently underway and the results will be presented in Chapter 4. In addition, WANL has estimated A-286 property data for NSS PDR.⁽²³⁾ The slight changes in strength and decreases in ductility which characterizes A-286 following irradiation are observed both in DRM estimates and data available from the literature. In the annealed conditions which characterize the application of this material in the SPP, it is classified as ductile.⁽⁷⁾

6.4.8.3 Conclusion

It can be safely concluded that the application of A-286 in the NSS SPP will not result in degradation due to irradiation.

6.4.9 321 Stainless Steel

This material is utilized in the NSS SPP solely for O-rings.

6.4.9.1 Non-NERVA Data

This stainless steel is equivalent to SS 304 with titanium added in an amount equal to five times the carbon content. Irradiation data are not available. However, it can certainly be deduced that its irradiated properties are similar to SS 304.

6.4.9.2 NERVA Data

There is no NERVA data available, however, data on SS 304, Section 6.4.5, is applicable.

6.4.9.3 Conclusions

Both because of the application of this material (O-rings) and the data available for SS 304, it can be concluded that it can be employed for this application without degradation. No additional studies are required.

6.4.10 Component Systems Analysis

6.4.10.1 Non-NERVA Data

There is no available information.

6.4.10.2 NERVA Data

Support Plate and Plena design for past NERVA reactors have differed significantly from the present design. However, they have primarily been aluminum structures which have functioned without apparent difficulties.

6.4.10.3 Conclusions

It may be concluded that the present NSS SPP design will not be degraded by the radiation and thermal environments specified in Tables 6-3 and 6-4. For the materials utilized in this component, radiation at these temperatures, invariably causes decreased ductility and generally increases strength. Some minor decreases in strength have been observed, but this will not affect SPP design. Thus, SOL or unirradiated design criteria will be applicable.

TABLE 6-1
LIST OF MATERIALS (940J449)

78	▶	14	O'RING .937 O.D .063 TUBE DIA .010 THK WALL	U-2210-00957-SEA	CRES	TYPE 321 A6 COATED		
72		13	CAP SCREW, 50C HD.312-24 UNF-2A		CRES	AMS 5737 (A-286)		
36		12	LOCKING CUP		CRES	QQ-5-763 CL 304 COND A		
36		11	BOLT .375-24 UNF-2A		CRES	AMS 5737 (A-286)		
1		10	RING, INSTRUMENTATION		ALUMINUM	QQ-A-250/11-T61 (6061)		
6		9	SEGMENT, DAFFLE		↓	↓		
1		8	RING, PLENUM CLOSURE		ALUMINUM	PDS 30258-1 (6061-T61)		
1	▶	7	O' RING 38.500 I.D .125 TUBE DIA .010 WALL	U-2410-38750-50A	CRES	TYPE 321 A6 COATED		
1	▶	6	O RING 41.000 O.D.	U-2410-41000-50A				
1	▶	5	O' RING 41.250 I.D	U-2410-41500-50A				
1	▶	4	O' RING 42.625 O.D	U-2410-42625-50A				
36		3	LOCKING CUP		CRES	QQ-5-763 CL 304 COND A		
36		2	BOLT .375-24 UNF-2A		CRES	AMS 5737 (A-286)		
1		1	SUPPORT PLATE AND PLENA		ALUMINUM	PDS30258-2 (6061-T61)		
	QTY. REQD	QTY. REQD	NOMENCLATURE OR DESCRIPTION	CODE IDENT NO.	PART OR IDENTIFYING NO.	MATERIAL	SPECIFICATION	ZONE
LIST OF MATERIALS								

TABLE 6-1

LIST OF MATERIALS (939J725) (CONT'D)

1	1	48	ELBOW (SPECIAL)			AL	(6061)		
1	1	47	EXPANDING "Y" (SPECIAL)			AL	(6061)		
1	1	46	REDUCING "Y" (SPECIAL)			AL	(6061)		
1	1	45	BYPASS INLET PIPE			AL	QQ-A-200/5-H34-(5086)		
1	1	44	SSCA INLET PIPE			AL	QQ-A-200/5-H34-(5086)		
1	1	43	ORING-.125 DIA.012 WALL 9.750 O.D.			AL	WW-T-700/6-T6-(6061)		
1	-	42	RING			AL	QQ-A-250/11-T651-(6061)		
36	-	41	STUD-.375-24 UNF-2A			CRES	AMS-5735 (A286)		
		40							
36	-	39	TUBING .875 O.D.X.058 WALL			AL	WW-T-700/5-H34-(5086)		
1	1	38	TUBING .875 O.D.X.058 WALL			AL	WW-T-700/5-H34-(5086)		
12	12	37	SCREW-.250-28 UNF 2A			CRES	AMS-5735 (A286)		
12	12	36	LOCKING DEVICE			CRES	QQ-S-763-TY304-CONDA		
1	1	35	SEAL, INTERFACE RING			AL	PDS 3025B-2 (6061-T61)		
1	1	34	SPIROLOX 1.562 O.D.,.068 THK	RST-156		CRES	QQ-S-766-TY302-1/4HARD		
1	1	33	RING, RETAINER			CRES	QQ-S-763-TY304-CONDA		
2	2	32	WAVE SPRING			CRES	QQ-S-766-TY302-1/4HARD		
2	2	31	PISTON RING SEAL			CRES	QQ-S-766-TY302-1/2HARD		
1	1	30	SCREEN #20 MESH (.016 DIA WIRE)			CRES	QQ-W-423-TY304-CONDA		
1	1	29	INTERFACE SEAL SLEEVE			AL	PDS 3025B-2 (6061-T61)		
1	1	28	SPIROLOX 3.875 O.D.,.120 THK	RST-387		CRES	QQ-S-766-TY302-1/4HARD		
1	1	27	RING, RETAINER			CRES	QQ-S-763-TY304-CONDA		
1	1	26	SCREEN #20 MESH (.016 DIA WIRE)			CRES	QQ-W-423-TY304-CONDA		
2	2	25	PISTON RING SEAL			CRES	QQ-S-766-TY302-1/2HARD		
2	2	24	WAVE SPRING			CRES	QQ-S-766-TY302-1/4HARD		
1	1	23	INTERFACE SEAL SLEEVE			AL	PDS 3025B-2 (6061-T61)		
1	1	22	BY PASS TORUS			AL	WW-T-700/5-H34-(5086)		
1	1	21	INLET TORUS			AL	WW-T-700/5-H34-(5086)		
6	6	20	TUBING .875 O.D.X.058 WALL			AL	WW-T-700/5-H34-(5086)		
6	6	19	ELBOW UNION (SPECIAL)			AL	(6061)		
29	29	18	TUBING .875 O.D.X.058 WALL			AL	WW-T-700/5-H34-(5086)		
30	30	17	ELBOW UNION (SPECIAL)			AL	(6061)		
1	1	16	CENTRAL DOME			AL	PDS 3025B-2 (6061-T61)		
8	8	15	BOLT-.375-24 UNF-2A			CRES	AMS 5735 (A286)		
8	8	14	LOCKING DEVICE			CRES	QQ-S-763-TY304-CONDA		
1	1	13	FLOW DIFFUSER			AL	PDS 3025B-2 (6061-T61)		
8	8	12	SLEEVE			AL	QQ-A-225/8-T651-(6061)		
1	1	11	PISTON RING HOUSING			AL	PDS 3025B-2 (6061-T61)		
2	2	10	PISTON RING SEAL			CRES	QQ-S-766-TY302-1/2HARD		
2	2	9	WAVE SPRING			CRES	QQ-S-766-TY302-1/4HARD		
36	36	8	LOCKING DEVICE			CRES	QQ-S-763-TY304-CONDA		
36	36	7	NUT-.375-24 UNF-2B			CRES	AMS-5735 (A286)		
-	36	6	STUD-.375-24 UNF-2A			CRES	AMS-5735 (A286)		
12	12	5	LOCKING DEVICE			CRES	QQ-S-763-TY304-CONDA		
12	12	4	BOLT-.375-24 UNF-2A			CRES	AMS-5735 (A286)		
-	2	3	ORING-.125 DIA.012 WALL-9.750 O.D.			AL	WW-T-700/6-T6 (6061)		
-	36	2	TUBING-.875 O.D.X.058 WALL			AL	WW-T-700/5-H34-(5086)		
-	1	1	PIPING CYLINDER			AL	PDS 3025B-2 (6061-T61)		
602	601	QTY.	ITEM NO.	NOMENCLATURE OR DESCRIPTION	CORE IDENT NO.	PART OR IDENTIFYING NO.	MATERIAL	SPECIFICATION	2008
LIST OF MATERIALS									

TABLE 6-2
MATERIALS SUMMARY-SUPPORT PLATE AND PLENA

Aluminum	6061	T6
Aluminum	6061	T61 (Ring and Plate Forging)
Aluminum	6061	T651
Aluminum	5086	H34 (Piping)
Aluminum		(Special)
SS	304	(Annealed)
SS	302	(1/4 hard and 1/2 hard)
SS		A-286
SS		321 (Silver coated)

TABLE 6-3

THE 3 σ MAXIMUM RADIATION ENVIRONMENT FOR MATERIALS IN SUPPORT PLATE ECC

Item	MATERIAL	POSITION		GAMMA ENVIRONMENT			NEUTRON ENVIRONMENT								
		RADIUS	STATION	ENVIRONMENT			FAST ($E > 1.0$ MeV)			INTERMEDIATE ($0.4 \text{ eV} \leq E \leq 1.0$ MeV)			THERMAL ($E < 0.4 \text{ eV}$)		
		(IN)	(IN)	Dose Rate	1-Hour Dose	60-Ten Min. Cycle Dose	1-Hour Fluence	60-Ten Min. Cycle Fluence	Flux	1-Hour Fluence	60-Ten Min. Cycle Fluence	Flux	1-Hour Fluence	60-Ten Min. Cycle Fluence	Flux
				RAD-C/Hr	RAD-C	RAD-C	N/CM ² -Sec	N/CM ²	N/CM ²	N/CM ² -Sec	N/CM ²	N/CM ²	N/CM ² -Sec	N/CM ²	N/CM ²
1	Al 6061-T6	19.476	-10.83C	7.6(8)	7.6(8)	7.6(9)	3.2(14)	1.1(18)	1.1(19)	2.5(15)	9.2(18)	9.2(19)	2.9(14)	1.0(18)	1.0(19)
			G	7.6(8)	7.6(8)	7.6(9)	2.5(14)	9.1(17)	9.0(18)	1.8(15)	6.7(18)	6.7(19)	2.2(14)	8.0(17)	8.0(19)
Al(2)	Al 6061-T61	0.0	-3.5 C*	5.9(9)	5.9(9)	5.9(10)	3.2(15)	1.2(19)	1.2(20)	2.2(16)	7.8(19)	7.8(20)	8.4(14)	3.4(18)	3.4(19)
			G	6.5(9)	6.5(9)	6.5(10)	2.4(15)	8.7(18)	8.7(19)	1.6(16)	5.6(19)	5.6(20)	7.4(14)	2.7(18)	2.7(19)
3	Al 6061-T651														
4	Al 5086 - H34			SAME AS ABOVE AT R = 19.476; Z = -10.83											
5	Al 6061 - SPEC														
	CRES QQ-S-763 (6) 304 COND A (SS-304)	21.115	-3.5	SAME AS BELOW AT R = 21.115; Z = -3.5											
7	SS-302	9.476	-10.83C S	SAME AS BELOW AT R = 19.476; Z = -10.83											
CRES (8)	AMS-5737 (A-286)	.115	-3.5C G	1.8(9) 2.0(9)	1.8(9) 2.0(9)	1.8(10) 2.0(9)	9.5(14) 7.2(14)	3.4(18) 2.6(18)	3.4(19) 2.6(19)	7.1(15) 5.2(15)	2.6(19) 1.9(19)	2.6(20) 1.9(20)	8.8(14) 6.8(14)	3.2(18) 2.4(18)	3.2(19) 2.4(19)
CRES (9)	AlSi-321 (SS-321)	.115	-3.5	SAME AS ABOVE AT R = 21.115; Z = -3.5											

* C - Composite Core

G - Graphite Core

Numbers in () refer to powers of ten, i. e. 5.9 (9) = 5.9×10^9

TABLE 6-4
SUMMARY OF SUPPORT PLATE AND PLENA COMPONENT MATERIAL TEMPERATURES AND
OTHER KEY PERFORMANCE PARAMETERS (GRAPHITE CORE)

COMPONENT	LOCATION	PARAMETER	NORMAL MODE OPERATION						COOLDOWN OPERATION RANGE	SPACE SOAK RANGE	
			RATED PERFORMANCE 99/95 TOLERANCE LIMIT			THROTTLE HOLD 99/95 TOLERANCE LIMIT				Solar Flux of 300 Btu/HR/Ft ²	Solar Flux of 3000 Btu/HR/Ft ²
			NOMINAL	UPPER	LOWER	NOMINAL	UPPER	LOWER		(3)	(4)
Support Plate		SOL T max (°R)	507	558	456	432	470	394	(250-530°R) (1) 800°R (2)	350-550	550-800
		T min (°R)	180	158	202	151	131	171			
		T avg (°R)	342	373	311	337	367	307			
		T _{bypass} -T _{feed} (°R)	60	78	42	72	90	54			
		EOL T max (°R)	437	475	399	390	421	359			
		T min (°R)	149	169	129	129	117	141			
		T avg (°R)	324	354	294	319	343	295			
		T _{bypass} -T _{feed} (°R)	73	91	55	84	105	63			
	Feed Plenum Radial Flow Channel	Radial Fluid Temperature Rise at SOL (°R)	48	58	38	30	36	24			
		Radial Fluid Temperature Rise at EOL (°R)	30	36	24	20	24	16			
		Radial ΔP at EOL (Psi)	2	2.4	1.6						
	Bypass Plenum Radial Flow Channel	Radial Fluid Temperature rise at SOL (°R)	120	150	90	137	171	103			
		Radial Fluid Temperature rise at EOL (°R)	118	148	88	139	173	105			
	Axial Flow Channel	Fluid ΔT	3	3.5	2.5	-1.5	-2	-1			
		Fluid ΔP	11	13	9	3	7	9			
Feed Flow Supply System	SOL T max (°R)	285	301	269				(200-550°R) (1) 800°R (2)	350-550	550-800	
		T min (°R)	155	175	135						
	EOL T max (°R)	282	298	266	302	323	281				
		T min (°R)	138	156	120	119	133	105			
	SOL Maximum Fluid Temperature Rise (°R)	37	42	30							
	EOL Maximum Fluid Temperature Rise (°R)	27	33	21	26	31	21				
	ΔP at EOL	3.2	4.2	2.2	5.2	6.8	3.6				

TABLE 6-4

**SUMMARY OF SUPPORT PLATE AND PLENA COMPONENT MATERIAL TEMPERATURES AND
OTHER KEY PERFORMANCE PARAMETERS (GRAPHITE CORE) (CONT' D)**

COMPONENT	LOCATION	PARAMETER	NORMAL MODE OPERATION						COOLDOWN OPERATION RANGE	SPACE SOAK RANGE	
			RATED PERFORMANCE 99/95 TOLERANCE LIMIT			THROTTLE HOLD 99/95 TOLERANCE LIMIT				Solar Flux ⁽³⁾ of 300 Btu/HR/Ft ²	Solar Flux ⁽⁴⁾ of 3000 Btu/HR/Ft ²
			NOMINAL	UPPER	LOWER	NOMINAL	UPPER	LOWER			
Bypass Flow Supply System		SOL T max (°R)	274	290	258						
		T min (°R)	125	140	110						
		EOL T max (°R)	278	294	262	308	330	286			
		T min (°R)	128	143	113	131	147	115			
		SOL-Maximum Fluid Temperature Rise (°R)	24	29	19				(275-600°R) (1) 800°R (2)		
	EOL-Maximum Fluid Temperature Rise (°R)	24	29	19	30	36	24				
Piping Cylinder		Max. Temperature	325	342	308	338	359	317	(275-600°R) (1) 800°R (2)	350-550	550 900
		Min. Temperature	285	300	270	305	325	385			
Central Dome		Max. Temperature	298	314	282	340	361	319	(275-600°R) (1) 800°R (2)	350-550	550 900
		Min. Temperature	288	304	272	335	356	314			

(1) This is the temperature range within 3.65 hours after a ten minute burn and throttle hold, as reported in WANL-TME 2678 "Systems Analysis Summary Report, Nuclear Subsystem", Section 5.2.6.3

(3) This is the minimum flux shown in WANL-TME-2678 "Systems Analysis Summary Report, Nuclear Subsystem" Section 5.2.6.3. Solar Flux in earth orbit is 444 Btu/Hr/Ft² as reported in WANL-TME-2678

(2) This is the maximum allowable temperature between cooling pulses.

(4) This is the extreme solar flux in the vicinity of mercury as reported in WANL-TME-2678.

TABLE 6-5 EFFECT OF IRRADIATION ON MECHANICAL PROPERTIES OF ALUMINUM ALLOYS

Material	Condition	Fast Fluence	Irr. Temp. °R	Test Temp. °R	Yield Strength		Ultimate Tensile Strength		Elongation, percent		
		n/cm ² (E > 1 MeV)			1000 psi Unirr.	1000 psi Irr.	1000 psi Unirr.	1000 psi Irr.	Total		Ref.
									Unirr.	Irr.	
Al (99.99)	Annealed	8.0 × 10 ²⁰ (a)	670	RT	2.1	2.1	6.6	12.2	46	30	5
1100	Ditto	8.7 × 10 ²¹	580	RT	4.8	48.3	9.7	55.9	33	9.7	6
1100	"	8.7 × 10 ²¹	580	RT	4.8	43.2	9.7	53.0	33	6.6	6
1100	"	1.1 × 10 ²²	580	RT	4.8	50.6	9.7	55.4	33	3.0	
1100	"	1.1 × 10 ²²	580	RT	4.8	46.0	9.7	53.7	33	3.2	6
1100	H-12	5.0 × 10 ¹⁹	670	RT	14.0	16.0	16.0	19.0			7
1100	H-12	6.4 × 10 ¹⁹	670	RT	15.5	16.6	17.4	19.8			7
1100	H-12	8.0 × 10 ^{20(a)}		RT	15.1	18.7	17.6	22.0			5
2024		2 × 10 ¹⁹	580	RT	45.3	48.7	71.6	71.1	26	25	8
2024		1.2 × 10 ²⁰	580	RT	45.3	48.2	71.6	74.3	26	26	8
2024		5.6 × 10 ²⁰	580	RT	45.3	56.1	71.6	79.4	26	25	8
2024		9.8 × 10 ²⁰	580	RT	45.3	66.2	71.6	84.9	26	24	8
6061		5.2 × 10 ²⁰	580	RT	40	43.7	47.2	50.2	21	22	8
6061		1.2 × 10 ²¹	580	RT	40	42.8	47.2	51.9	21	21	8
6063	Annealed	1.3 × 10 ²⁰	670	RT	8.0	24.0	19.0	34.0	33.0	20.0	5
6063	Annealed	8.0 × 10 ^{20(a)}	670	RT	8.2	33.9	19.4	24.2	24	13	3
6063	T-4	1.0 × 10 ¹⁸	560	RT	12.0		24.5		30.4	31.2	5
6063	T-4	6.4 × 10 ¹⁹	670	RT	13.8	28.6	28.5	40.7			3
6063	T-4	8.0 × 10 ^{20(a)}	670	RT	14.1	39.4	28.5	48.2			3
6063	T-6	1.0 × 10 ¹⁸	560	RT	28.3	28.1	33.2	34.1	23.4	20.3	5
6063	T-6	1.0 × 10 ¹⁸	560	RT	33.4	33.8	34.6	35.8	16.4	14.1	5
6063	T-6	5.0 × 10 ¹⁹	670	RT	34.0	36.0	39.0	40.0			5
X-2219		1.3 × 10 ²⁰	670	RT	25.0	28.0	38.0	42.0	17	16	7
X-2219		3.0 × 10 ^{20(a)}		RT	28.5	32.8	38.1	47.0	12	12	5
X-2219		1.3 × 10 ²⁰	670	RT	38.0	50.0	58.0	68.0	17	11	7
X-2219		8.0 × 10 ^{20(a)}		RT	38.3	47.2	57.8	63.8	12	11	5

TABLE 6-5 EFFECT OF IRRADIATION ON MECHANICAL PROPERTIES OF ALUMINUM ALLOYS (CONT'D)

Material	Condition	Fast Fluence	Irr. Temp. °R	Test Temp. °R	Yield Strength		Tensile Strength		Elongation, percent		Ref.
		n/cm ² (E>1 MeV)			Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
M-257	Extruded	5.4 × 10 ¹⁹	560	RT	28.6	34.7	36.2	38.6	8.0	6.5	1
M-257	Ditto	1.0 × 10 ²⁰	560	RT	28.6	35.8	36.2	40.2	8.0	6.3	1
M-257	"	1.2 × 10 ²⁰	580	RT	30.0	31.4	37.4	39.7	16.8	11.8	9
M-257	"	1.2 × 10 ²⁰	580	RT	19.0	20.0	21.2	21.5	11.5	11.6	9
M-257	"	1.5 × 10 ²⁰	580	RT	30.0	32.0	37.4	41.0	16.8	11.5	9
M-257	"	1.2 × 10 ²⁰	580	RT	19.0	20.1	21.2	22.0	11.5	10.5	9
M-257	"	3.1 × 10 ¹⁹	980	RT	31.9	28.6	43.3	37.4	12.3	17.2	9
M-257	"	3.1 × 10 ¹⁹	980	RT	21.0	18.2	28.0	21.0	11.5	11.0	9
A-356		2.0 × 10 ¹⁹	580	RT	26	29.1	33.1	36.7	4	6	8
A-356		1.2 × 10 ²⁰	580	RT	26	33.5	33.1	42.0	4	6	8
A-356		5.6 × 10 ²⁰	580	RT	26	42.4	33.1	45.9	4	6	8
A-356		9.8 × 10 ²⁰	580	RT	26	52.1	33.1	54.4	4	6	8
M-486	Extruded	7.7 × 10 ¹⁹	580	RT	31.9	30.7	40.8	44.1	17.0	15.5	9
M-486	Extruded	7.7 × 10 ¹⁹	580	RT	19.8	20.4	21.0	21.9	16.8	16.6	9

(a)

energy greater than 0.1 MeV

TABLE 6-6 COMPARISON OF PROPERTIES OF ALUMINUM ALLOYS

<u>Alloy</u>	<u>Fast Fluence n/ cm² (E > 1.0 Mev)</u>	<u>Type</u>	<u>Results</u> ^(a)		<u>Reference</u>
			<u>Ultimate Tensile Strength</u>	<u>0.2% Yield Strength</u>	
6061 T6 DRM 51294 estimated RT	1 hr. NERVA	Ring Forging Tang & Axial	0%	0%	
1110	0.4×10^{19}	H-12	16% inc ^(b)	12% inc	7
2024	1.2×10^{20}	--	4% inc	6% inc	8
6061	5.2×10^{20}		6% inc	10% inc	8
6063	6.4×10^{19}	T-4	30% inc	50% inc	7
6063	5.0×10^{19}	T-6	2% inc	3% inc	7
X-2219	1.3×10^{20}		9% inc	5% inc	7
X-8001	5.0×10^{19}	Annealed	6% inc	20% inc	7
X-8001	1.3×10^{20}	H-14	4% inc	5% inc	7

(a) all room temperature tests

(b) inc = increase

TABLE 6-7
EFFECTS OF IRRADIATION ON 2219 ALUMINUM ALLOY ⁽⁵⁾
2219-T6

FORM	SAMPLE	NEUTRON FLUENCE	TEMPERATURE °R			ULTIMATE KSI		YIELD STRENGTH, KSI		ELONGATION %	
			IRRAD	ANNL	TEST	CONTROL	IRR	CONTROL	IRR	CONTROL	IRR
Forging	4	7.9×10^{17}	140	0	140	72	80	52	78	9.3	3.9
Forging		6.1×10^{17}	140	540	140	***	72	***	54	***	8.3
Forging		6.4×10^{17}	140	0	540	54	61	46	47	9.2	9.1
Sheet		8.5×10^{17}	140	0	140	73	75	49	63	9.0	7.8
Sheet		8.5×10^{17}	140	540	140	***	74	***	51	***	10.0
Sheet		8.0×10^{17}	140	1000	140	68	67	40	39	12.8	11.3
Sheet		8.5×10^{17}	140	1000	1000	22	22	20	20	7.3	7.5
Sheet F		7.5×10^{17}	140	0	540	59	58	42	41	6.8	5.6

2219-T852

FORM	SAMPLE	NEUTRON FLUENCE	TEMPERATURE, °R			ULTIMATE, KSI		YIELD STRENGTH, KSI		ELONGATION %	
			IRRAD	ANNL	TEST	CONTROL	IRR	CONTROL	IRR	CONTROL	IRR
Forging	5	9.0×10^{17}	140	0	140	76	80	56	71	9.0	7.1
Forging		9.0×10^{17}	140	240	140	***	79	***	70	***	3.0
Forging		7.3×10^{17}	140	340	140	***	77	***	65	***	7.1
Forging		7.8×10^{17}	140	440	140	***	73	***	56	***	9.0
Forging		7.1×10^{17}	140	540	140	***	75	***	56	***	7.6
Forging		6.3×10^{17}	140	0	240	68	69	57	58	9.6	7.8
Forging		7.4×10^{17}	140	0	340	65	65	49	54	12.1	9.3
Forging		6.3×10^{17}	140	0	440	63	63	49	49	7.6	5.5
Forging		1.1×10^{17}	140	540	540	61	61	47	47	9.2	5.5
Plate		8.0×10^{17}	140	0	140	81	84	64	78	11.2	9.6
Plate		8.0×10^{16}	140	540	140	***	81	***	65	***	8.7
Plate		3.1×10^{17}	140	540	540	67	67	54	54	10.3	9.7
Plate		8.0×10^{17}	140	0	140	59	63	45	58	3.5	2.5
Plate		8.0×10^{17}	140	540	140	***	58	***	43	***	3.3
Plate		3.1×10^{17}	140	540	540	43	43	36	35	2.5	3.0

TABLE 6-8 EFFECTS OF IRRADIATION ON 6061-T6 ALLOY (5)

FORM	NEUTRON FLUENCE	TEMPERATURE, °R			ULTIMATE, CONTROL	KSI IRRD	YIELD STRENGTH, KSI		ELONGATION %	
		IRRAD	ANNL	TEST			CONTROL	IRRD	CONTROL	IRRD
Sheet	6.0×10^{16}	30	0	40	63	69	45	55	23.7	24.1
Sheet	6.0×10^{16}	30	0	40	68	71	50	56	7.5	9.3
Sheet	6.0×10^{16}	30	0	40	57	44	31	32	9.2	6.3
Forging	1.1×10^{18}	140	0	140	62	66	53	64	6.6	8.2
Forging	1.1×10^{18}	140	540	140	***	63	***	59	***	8.2
Forging	1.1×10^{18}	140	0	140	64	65	51	62	17.7	14.2
Forging	1.1×10^{18}	140	0	140	58	61	48	56	17.2	8.3
Sheet	3.1×10^{18}	40	0	40	73	76	52	72	26.4	18.1
Sheet	2.2×10^{18}	40	140	40	***	71	***	63	***	22.4
Sheet	2.0×10^{18}	140	0	140	61	61	51	58	20.7	16.5

TABLE 6-9
IN-REACTOR CREEP OF ANNEALED TYPE 304 STAINLESS STEEL ⁽¹⁴⁾

Temp, C	Stress, 1000 psi	Percent Creep Strain at Failure		Failure Time, hours	
		In-Reactor	Ex-Reactor	In-Reactor	Ex-Reactor
650	20	8 - 0	25 - 26	150	570
550	30	3 - 4	8	770	4500

TABLE 6-10
MINIMUM CREEP RATE OF TYPE 304 STAINLESS STEEL AT 650 C (a) ⁽¹⁵⁾

Stress, 1000 psi	Minimum Creep Rate ^(b) , percent/hr		
	Unirradiated	In-Reactor	Postirradiation
35	2.26 (7.7)		
30	0.92 (9.0)		0.50 (12.0)
27	0.33 (44.1)		
25	0.043 (122.8)	0.138 (17.1)	0.139 (24.4)
22.25	0.042 (157.7)		0.014(123.4)
20	0.0073 (366.6)	0.016 (98.0)	
17		0.0035 (423.0)	0.006(314.4)
15	0.00032	0.0014	0.0018(956.9)

(a) The instantaneous fluence for in-pile specimens was 6×10^{13} n/cm² thermal and 5×10^{20} n/cm² fast (2.9 MeV). The fluence of postirradiation specimens was 2×10^{20} n/cm² thermal and 2.9×10^{19} n/cm² fast.

(b) Numbers in parentheses are rupture life in hours.

TABLE 6-11
STRESS-RUPTURE DUCTILITY OF TYPE-304 STAINLESS STEEL AT 650 C (16)

Stress, 1000 psi	Elongation, percent			Reduction in Area, percent		
	Unirradiated	In-Reactor	Postirradiation	Inirradiated	In-Reactor	Postirradiation
35	37.2			35.1		
30	17.2		8.4	25.7		9.1
27	26.6			28.0		
25	17.2	6.7	4.9	22.3	11.1	4.2
22.25	18.4		2.9	19.9		4.0
20	17.5	2.6		13.9	3.0	
17			3.7			
15			2.8			0.2

- (a) The instantaneous fluence for in-pile specimens was $6 \times 10^{13} \text{ n/cm}^2$ thermal and $5 \times 10^{12} \text{ n/cm}^2$ fast ($>2.9\text{MeV}$).
The fluence of postirradiation specimens was $2 \times 10^{20} \text{ n/cm}^2$ thermal and $2.9 \times 10^{19} \text{ n/cm}^2$ fast.

Figure 6-1. AL6061-T6 NERVA Test Experience

Figure 6-2 Effects of Irradiation on Room-Temperature Yield Strength and Ductility of Type 304 Stainless Steel

Figure 6-3 Effect of Test Temperature on the Strength of Irradiated Annealed and 25 Percent Cold-Worked Type 304 Stainless Steel⁽¹²⁾

Irradiation temperature 60 C; fast fluence $1.0 \times 10^{21} \text{ n/cm}^2$; all transverse specimens.

Figure 6-4 Room-Temperature Tensile Properties of Irradiated Type 304⁽²⁴⁾
Stainless Steel as a Function of Irradiation Temperature
Specimens were irradiated to a fast fluence of 7×10^{20} n/cm².

LIST OF REFERENCES
(Chapter 6 - Support Plate and Plena)

- (1) Bush, S. H., "Irradiation Effects in Cladding and Structural Materials, " ASM (1965).
- (2) Gronbeck, H. D., "ETR Radiation Damage Surveillance Programs Progress Report No. 2, " IN-1036 (February 1967).
- (3) McCoy, H. E., Jr., and Weir, J. R., Jr., "Influence of Irradiation on the Tensile Properties of Aluminum Alloy 6061, " Nuclear Science Engineering, 25, 319-327 (1966).
- (4) Kangilowski, M., "The Effects of Neutron Irradiation on Structural Materials," REIC Report No. 45 (June 1967).
- (5) Carlson, E., and French, W. E., "NERVA Radiation Effects Summary Report, " S-110-SRO1-W181, (May 1970).
- (6) Schollenberger, R. S., DRM 51294, Westinghouse Astronuclear Laboratory (August 1970).
- (7) Holden, A. N., DRM 51281, Westinghouse Astronuclear Laboratory (June 1970).
- (8) Schollenberger, R. S., DRM 51276, Westinghouse Astronuclear Laboratory (August 1970).
- (9) Schollenberger, R. S., DRM-53161, Westinghouse Astronuclear Laboratory (August 1970).
- (10) Schollenberger, R. S., DRM 51201, Westinghouse Astronuclear Laboratory. (Aug. 1970)
- (11) Irvin, J. E., Bement, A. L., and Hoagland, R. G., "The Combined Effects of Temperature and Irradiation on the Mechanical Properties of Austenitic Stainless Steels, " BNWL-1 (January 1965).
- (12) Irvin, J. E., and Bement, A. L., "The Nature and Engineering Significance of Radiation Damage to Various Stainless Steel Alloys, " BNWL-SA-376 (April 1, 1966).
- (13) Martin, W. R., and Weir, J. R., "Influence of Grain Size on the Irradiation Embrittlement of Stainless Steel at Elevated Temperatures, " ORNL-TM-1043 (March 1965).

- (14) Williams, J. A., and Carter, J. W., "The Creep of Annealed 304 Stainless Steel During Irradiation and Its Engineering Significance," Symposium on Effects of Radiation on Structural Metals, ASTM-STP-426 (1967).
- (15) Bloom, E. E., and Weir, J. R., "In-Reactor and Postirradiation Creep-Rupture Properties of Type 304 Stainless Steel," Trans. Am. Nucl. Soc., 10, 131-132 (June 1967).
- (16) Reynolds, M. G., "Strain-Cycle Phenomena in Thin-Wall Tubing," Flow and Fracture of Metals and Alloys in Nuclear Environments, ASTM-STP-380, 323-336 (1965).
- (17) Joseph, J. W., "Stress Relaxation in Stainless Steel During Irradiation," USAEC Report DP-369 (June 1959).
- (18) Schollenberger, R. S., DRM 51232, Westinghouse Astronuclear Laboratory (August 1970).
- (19) Schweiker, R. E., "Relaxation of Torsional Stresses in Stainless Steel During Irradiation," USAEC Report DP-669 (April 1962).
- (20) Smith, K., "Reactor Handbook, Volume I, Materials," Tipton, C. R., Editor Interscience Publishers, Inc., New York, (1960).
- (21) Allen, J. W., "Personal Communication", (November 1970).
- (22) McAndrew, J. B., DRM 53174, Westinghouse Astronuclear Laboratory (October 1970).
- (23) Schollenberger, R. S., DRM 51250, Westinghouse Astronuclear Laboratory (October 1970).
- (24) Bloom, E. E., Stiegler, J. O., Martin, W. R., and Weir, J. R., "Comparison of Displacement Damage and Strength for Stainless Steel Irradiated at Intermediate Temperature," ORNL-TM-1535 (June 1966).
- (25) Schwanbeck, C. A., "Effect of Nuclear Radiation on Materials at Cryogenic Temperatures, Final Report," NASA-CR-54881 (N66-24695) (January 1965).
- (26) Lombardo, J. J. Dixon, C. E., and Begley, J. A., "Cryogenic Radiation Effects on NERVA Structural Materials," Paper presented at the 69th Annual Meeting of ASTM in Atlantic City, New Jersey, June 27 - July 1, 1966.

RADIATION EFFECTS INFORMATION REQUESTS (REIRS)

(Chapter 6 – Support Plate & Plena)

<u>No.</u>	<u>Description</u>
6	Aluminum 5086 – H34 Pipe (Welded) Tensile and Fracture Strength
7	SS 304 – Stress Rupture and Creep
8	SS 302 Springs – Stress Relaxation

IDENTIFICATION NO. 6

RADIATION EFFECTS INFORMATION REQUEST

DATE November 10, 1970

SUBJECT Aluminum 5086 - H34 Pipe FUNCTIONAL CATEGORY Test and Evaluation

REF. ITEM SUPPORT EC 677561 REF. DATA ITEMS S-110

INFORMATION DESIRED Effect of irradiation on tensile properties and fracture strength of welded pipe.
(PURPOSE OF STUDY OR EXPERIMENT)

DATE REQUIRED CY 71

RELATION OF ITEM TO ENGINE SSCA Feed and Bypass System
(LOCATION)

PER DRAWINGS, DATA ITEMS Drawing 940J449 and 939J725

ENVIRONMENT (4): NEUTRON FLUX _____ NEUTRON FLUENCE 1.1×10^{18} and 1.1×10^{19} ($E > 1$ Mev)
(Ref. data items) GAMMA DOSE RATE _____ GAMMA DOSE _____

TEMPERATURE $140^{\circ} - 750^{\circ}R$ NUCLEAR HEATING RATE _____

PRESSURE _____ OTHER _____

FLUID _____

CYCLE REQUIREMENTS TBD

RELIABILITY ESTIMATE Screening* REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL Screening* APPROVED _____ DATE _____

APPROVED _____ DATE _____

SPECIAL CONSIDERATIONS _____ APPROVED _____ DATE _____

PERTINENT REFERENCES _____

* If pipe is brittle, cat. "A" reliability is required (99% reliability at 95% C.L.).

IDENTIFICATION NO. 7

RADIATION EFFECTS INFORMATION REQUEST

DATE November 10, 1970

SUBJECT SS 304, cond. A (annealed) FUNCTIONAL CATEGORY Test and Evaluation

REF. ITEM SUPPORT EC 677561 REF. DATA ITEMS S-110

INFORMATION DESIRED Effect of irradiation on stress rupture and creep properties of SS 304 BAR.
(PURPOSE OF STUDY OR EXPERIMENT)

DATE REQUIRED CY 71-72

RELATION OF ITEM TO ENGINE Hardware in support plate and plena
(LOCATION)

PER DRAWINGS, DATA ITEMS Drawing 940J449 and 939J725

ENVIRONMENT (4): NEUTRON FLUX _____ NEUTRON FLUENCE 3.4×10^{18} and 3.4×10^{19} (E>1 Mev)
(Ref. data items)

GAMMA DOSE RATE _____ GAMMA DOSE _____

TEMPERATURE 540 - 1600°R NUCLEAR HEATING RATE _____

PRESSURE _____ OTHER _____

FLUID TBD

CYCLE REQUIREMENTS _____

RELIABILITY ESTIMATE 99% REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL 95% APPROVED _____ DATE _____

APPROVED _____ DATE _____

SPECIAL CONSIDERATIONS _____ APPROVED _____ DATE _____

PERTINENT REFERENCES _____

IDENTIFICATION NO. 8

RADIATION EFFECTS INFORMATION REQUEST

DATE November 10, 1970

SUBJECT SS 302 Springs FUNCTIONAL CATEGORY Test and Evaluation

REF. ITEM SUPPORT EC 677561 REF. DATA ITEMS S-110

INFORMATION DESIRED Effect of irradiation on stress relaxation properties of SS 302 (1/4 hard) springs
(PURPOSE OF STUDY OR EXPERIMENT)

DATE REQUIRED CY 71-72

RELATION OF ITEM TO ENGINE SSCA Feed and Bypass System
(LOCATION)

PER DRAWINGS, DATA ITEMS Drawing 940J449 and 9395725

ENVIRONMENT (4): NEUTRON FLUX _____ NEUTRON FLUENCE 1×10^{18} and 1×10^{19} (E 1 Mev).
(Ref. data items)

GAMMA DOSE RATE _____ GAMMA DOSE _____

TEMPERATURE 140° - 750°R NUCLEAR HEATING RATE _____

PRESSURE _____ OTHER _____

FLUID _____

CYCLE REQUIREMENTS TBD

RELIABILITY ESTIMATE Screening* REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL Screening* APPROVED _____ DATE _____

APPROVED _____ DATE _____

SPECIAL CONSIDERATIONS _____ APPROVED _____ DATE _____

PERTINENT REFERENCES _____

* Requirement for reliability will be determined after review of screening test results.

REPORTS (T-119s) RESPONDING TO REIRS

(None prepared at this issuance)

7.0 INTERNAL SHIELD

7.1 SUMMARY AND CONCLUSIONS

The applications of BATH, lead, aluminum (T6, T61 and T651), SS 304 Cond. A (annealed), SS 321 and A-286 in the Internal Shield (IS) have been reviewed. Bath studies have shown that its characteristic low ductility is further reduced by radiation and low temperature exposure. Thus, additional tests are required to give design K_{IC} data on this material. In addition, its thermal conductivity properties following irradiation are not understood and screening tests are suggested for that purpose. Lead is not structurally employed in the Internal Shield and further tests are not considered necessary.

Aluminum (T6, T61 and T651) is known to remain ductile under NSS IS usage conditions. Thus, SOL design will apply and further tests are not required. However, aluminum 6061-T651 may become brittle in the short transverse direction as employed in the NSS IS and screening tests are requested to further investigate this potential problem.

The Stainless Steels (304, Cond. A, 321 and A-286) as employed in this component will not undergo degradation and no additional tests are considered necessary.

7.2 INTRODUCTION

7.2.1 Item Description

Functionally, the Internal Shield (IS) is designed to attenuate radiation below specified levels to protect engine components from radiation-induced degradation of reliability, performance and/or endurance.

7.2.2 Location

The Internal Shield (IS) is located approximately eleven inches (11") from the forward end of the core. The central and peripheral shields are located forward of the support plate.

7.2.3 Description of Sub-Components

The Internal Shield (IS) consists of a central and peripheral shield as well as the supporting structure described below. A more detailed description of the IS can be found in Data Item E-105. The bill of materials for the IS has been extracted from the reference drawing (939J741) and presented in Table 7-1. Table 7-2 summarizes the materials used in this component by generic names.

7.2.3.1 The central shield is made up of hexagonal cross section BATH modules. These modules are approximately two inches (2") across the flat dimension and 11.1 inches (11.1") in length. In the peripheral region of the central shield, irregular modules required to form the circumference are cut from regular modules. Coolant channels are provided by grooving the faces of each modules and drilling six (6) axial holes through each module. Modules are arranged in arrays of seven (7) and the central one is located by a stud through the cover plate, lead cluster, BATH modules and into the support plate. The six (6) surrounding modules are pinned to the support plate.

The central shield also contains lead clusters, 1.3 inches (1.3") thick, shaped to cover each BATH module array. The lead cluster is retained by a stud which passes through the cover plate, lead cluster, central BATH module of the array and into the support plate. Flow channels in the lead align with those in the BATH modules and are sized to compensate for the difference in CTE between lead and BATH materials.

7.2.3.2 The peripheral shield BATH modules are rectangular sections of rings. Four (4) circumferential rows of modules make up the peripheral shield. Modules have coolant channels at all module faces. Special shaped modules surround the propellant flow and bypass feed pipes. Eighteen (18) lead sectors, 1.3 inches (1.3") thick, cover the BATH modules circumferentially between the control drum drive shaft penetrations and radially from the feed and bypass pipes to the support ring. Lead filler blocks cover the outer

row of BATH modules between the feed and bypass pipes and piping cylinder.

7.2.3.3 The supporting structure is a 0.805 inch (0.805") thick aluminum plate. The plate is attached at its outer periphery to a radial flange on the piping cylinder. Load transmission is from the shield support plate to the piping cylinder plenum closure ring and thence to the reactor core load path. The peripheral shield is supported by the forward radial flange of the piping cylinder.

7.3 ENVIRONMENTAL AND OPERATIONAL REQUIREMENTS

7.3.1 Nuclear Environment

Table 7-3 presents the maximum gamma and neutron radiation environments (plus 3 sigma) experienced by sub-components of the Internal Shield (IS). Thus, the radiation levels presented in this table represent the most severe levels experienced by each material in the IS.

7.3.2 Thermal Environment

Table 7-4 presents the temperature ranges (including uncertainties) experienced by each material during typical operational modes, cooldown and storage.

7.4 ANALYSIS

Each of the materials listed in Table 7-2 is analyzed for its most critical application(s) in this section.

7.4.1 BATH - Aluminum - 24 v/o Ti H₂ - 6 v/o B₄C

This material is used for the modules (peripheral and central) in the Internal Shield (IS). Mechanical loads are negligible but thermal loads are important.

7.4.1.1 Non-NERVA Data

Since BATH was developed by WANL and has been used primarily in the NERVA program, there is no non-NERVA data available.

7.4.1.2 NERVA Data

There is only limited data available for BATH specimens irradiated at approximately 600°R to a fluence of $9 \times 10^{18} \text{ n/cm}^2$ ($E > 1 \text{ Mev}$). Mechanical property data for these specimens are presented in Table 7-5⁽¹⁾. Small increases in yield strength, both in tension and compression, were observed at testing temperatures as high as 800°R. However, the ultimate tensile strength was not affected above room temperature. The tensile elongation was significantly reduced by fast neutron irradiation.

This material is classified as brittle (by SNPO-C-1 definition) in both the irradiated and unirradiated state. Additional tensile tests on irradiated BATH specimens in GTR 20C further substantiate this observation⁽²⁾. However, the final data is not available at the publication of this report. Further, fracture toughness tests are planned in GTR 22 to determine BATH K_{IC} values and the extent of post-irradiation annealing as a function of temperature. At the present time, there is little experimental information available on these two parameters. However, WANL has estimated BATH K_{IC} values for NSS PDR⁽³⁾.

Thermal conductivity test on BATH were conducted and the data are presented in Figure 7-1.⁽¹⁾ There appears to be no significant effects due to irradiation from 570° - 800°R. Similarly, WANL has estimated that BATH thermophysical properties do not change with irradiation⁽³⁾. However these data show higher values (irradiated and unirradiated) than those cited from Reference 1.

7.4.1.3 Conclusions

It can definitely be concluded that BATH is a brittle material and becomes more brittle following irradiation and/or low temperature exposure. The magnitude of the change is not well-known. Similarly, the extent of annealing is not known and the use of one or ten hour data can only be conjectured. This will be investigated in GTR 22 and REIR No. 10 specifies the nature of the experimental data required.

Although the IS design utilizing BATH must be to brittle criteria; the magnitude of

the changes in K_{IC} values for BATH is not known. It has been estimated that BATH fracture toughness decreases with irradiation and that no significant annealing occurs⁽³⁾. Thus, EOL properties were used in the structural analysis. BATH fracture toughness is being investigated in GTR 22 and REIR No. 10 specifies the experimental data required. Since there is only limited data on BATH thermal conductivity, an additional REIR, No. 11, has been issued for this property, to verify the estimates cited above.

7.4.2 Lead

This material is used in 1.3 inches (1.3") thick lead clusters and sectors which are located above the BATH shield and attached to it. Mechanical loads are negligible, but thermal loads are important.

7.4.2.1 Non-NEVA Data

There is no non-NEVA data on the effects of radiation on the properties of lead applicable to the NEVA program. Lead is commonly used in the nuclear industry as shield material. However, it has not been used as a structural member because of its low melting point and low strength. Thus, no known studies on the effects of radiation on the mechanical properties of lead are available.

7.4.2.2 NEVA Data

There is no NEVA data available on the effects of irradiation on lead properties.

7.4.2.3 Conclusion

The application of lead in NSS design does not produce significant structural loads. Thus, there is no reason to be concerned with the effect of irradiation on mechanical properties of lead in the current NSS design. Thermal loads are primary and radiation does not appear to affect the physical properties of lead. No studies on its irradiated properties are considered necessary.

7.4.3 Al 6061-T6

This Aluminum is used for bolts, studs and locating pins in the Internal Shield.

7.4.3.1 Non-NERVA Data

There is a considerable amount of non-NERVA data cited in section 6.4.1.1 which establishes that this material is ductile at fluences and temperatures experienced in this application. Irradiation tends to decrease ductility and increase strength as is true for other aluminum alloys.

7.4.3.2 NERVA Data

NERVA data cited, in section 6.4.1.2, re-inforces the non-NERVA observation summarized above.

7.4.3.3 Conclusions

Aluminum 6061-T6 is ductile under NSS IS conditions. As has been concluded in section 6.4.1.3, SOL design criteria (unirradiated) are applicable and no additional testing is necessary.

7.4.4 Aluminum 6061-T61

This material is used in the Internal Shield (IS) for the flow control cylinder (as the ring forging) which experiences only slight structural (pressure) loads.

7.4.4.1 Non-NERVA Data

As discussed in section 6.4.2, there is no directly applicable non-NERVA data available. However, at NERVA Internal Shield fluences and temperatures, all evidence suggests that this material remains ductile with an increase in strength and decrease in ductility.

7.4.4.2 NERVA Data

The discussion of section 6.4.1.2 on Aluminum 6061-T6 is applicable because of the similarity in alloying composition and is used in the absence of directly applicable NERVA data. At the NSS Internal Shield fluences and temperatures, the same increased strength and decreased ductility is expected.

7.4.4.3 Conclusions

Aluminum 6061-T61 will certainly remain ductile over the NSS Internal Shield temperature and fluence range. Thus, to the extent required for design analysis, SOL criteria will apply. No additional testing is required.

7.4.5 Aluminum 6061-T651

This material will be used in the Internal Shield (IS) for the support ring sleeve, locking devices, cover plate and support plate. The latter is its most important application and the analysis will be tied to that application.

7.4.5.1 Non-NERVA Data

The discussion of Aluminum 6061-T6 is applicable since there is no directly applicable non-NERVA data for this material. At NSS Internal Shield fluences and temperatures, the invariable increased strength and decreased ductility which characterizes all aluminum alloys is expected.

7.4.5.2 NERVA Data

The discussion of section 6.4.1.2 on Aluminum 6061-T6 is applicable because there is no directly applicable NERVA data. At the NSS Internal Shield fluences and temperatures, the same invariable increased strength and decreased ductility is likely. However, there is some limited evidence, mostly based on engineering judgment, which suggests that this material, as used in the Internal Shield support plate, may exhibit significantly lower ductility.

7.4.5.3 Conclusions

It seems likely that Aluminum 6061-T651 will remain ductile in its application in the NSS Internal Shield and that SOL design criteria should apply. However, the possible embrittlement in the short transverse direction at cryogenic temperatures should be investigated and REIR No.9 is issued for that purpose.

7.4.6 Stainless Steel 304, Cond. A (Annealed)

This material's sole application is for several nuts in the Internal Shield (IS).

7.4.6.1 Non-NERVA Data

This material is reviewed in section 6.4.4.1 and has been extensively investigated. Slight increases in strength and substantial decreases in ductility have been observed following irradiation.

7.4.6.2 NERVA Data

No directly applicable NERVA data is available. However, section 6.4.4.2 compares the data from section 6.4.4.1 with WANL estimated property data prepared for NSS PDR. Slight increases in strength and decreases in ductility are observed following irradiation.

7.4.6.3 Conclusion

Due to the data cited above, as well as the application of Stainless Steel 304 Cond. A in the Internal Shield, radiation-induced changes will not affect NSS design. SOL design is pertinent and no additional testing is required for this application.

7.4.7 A-286 Stainless Steel

This material will be used for an impedance ring in the Internal Shield (IS). A-286 Stainless Steel will be thoroughly analyzed for its most important application in the cluster hardware support stems, Chapter 4.

7.4.7.1 Non-NEVA Data

Rather extensive non-NEVA data will be presented in Chapter 4. In summary, A-286 Stainless Steel remains ductile following irradiation and slight changes (both increases and decreases) in strength have been observed.

7.4.7.2 NEVA Data

An extensive NEVA Radiation Effects program for A-286 Stainless Steel is currently underway and will be presented in Chapter 4. In addition, WANL DRMs⁽⁴⁾ have estimated A-286 Stainless Steel property data for NSS PDR. The increases in strength and slight decreases in ductility which characterize A-286 Stainless Steel following irradiation, are observed in both these DRM estimates and in the data available from the literature.

7.4.7.3 Conclusions

The application of A-286 Stainless Steel as an impedance ring in the Internal Shield (IS) is relatively minor. NEVA and non-NEVA data suggest that SOL properties are applicable for design.

7.4.8 Stainless Steel 18-8

This material is used for heli-coil inserts in the Internal Shield (IS).

7.4.8.1 Non-NEVA Data

The irradiated properties of SS 18-8 are similar to SS-302, SS-304 and SS-321 as discussed in Chapter 6.4. There is no directly applicable non-NEVA data.

7.4.8.2 NEVA Data

There is no NEVA data available. However, data in section 6.4 on SS-304 and SS-321 is applicable.

7.4.8.3 Conclusions

Both because of its application (heli-coil inserts) and the data available for similar alloys, it can be concluded that SS-18-8 can be employed for this application without degradation. No additional studies are planned.

7.4.9 Component Systems Analysis

7.4.9.1 Non-NERVA Data

There is no applicable information.

7.4.9.2 NERVA Data

The IS in past NERVA reactors has included materials which differ significantly from the NSS design. Thus, there is no applicable NERVA data.

7.4.9.3 Conclusions

It must be concluded that the present NSS IS design will not be degraded by the radiation and thermal environments specified in Tables 7-3 and 7-4, based on the assessment of individual materials made above. However, verification testing (both screening and data which has 99.9% reliability at the 95% confidence level) is recommended for BATH and the aluminum support plate.

TABLE 7-2

MATERIALS SUMMARY - INTERNAL SHIELD

BATH

LEAD

AL-6061-T6

AL-6061-T61 (RING FORGING)

AL-6061-T651 (VARIOUS FORMS)

STAINLESS STEEL 304 - COND. A
(ANNEALED)

A-286 STAINLESS STEEL

STAINLESS STEEL 18-8

TABLE 7-3

THE 3 σ MAXIMUM RADIATION ENVIRONMENT FOR MATERIALS IN INTERNAL SHIELD ECC

Item	MATERIAL	POSITION		GAMMA ENVIRONMENT			NEUTRON ENVIRONMENT								
		RADIUS (IN)	STATION (IN)	Dose Rate RAD- C/Hr	1-Hour Dose RAD-C	60-Ten Min Cycle Dose RAD-C	FAST ($E > 1.0$ MeV)			INTERMEDIATE ($0.4 \text{ e.v.} < E < 1.0$ MeV)			THERMAL ($E < 0.4 \text{ e.v.}$)		
							Flux	1-Hour Fluence	60-Ten Min Cycle Fluence	Flux	1-Hour Fluence	60-Ten Min Cycle Fluence	Flux	1-Hour Fluence	60-Ten Min Cycle Fluence
							N/CM^2 Sec	N/CM^2	N/CM^2	N/CM^2 Sec	N/CM^2	N/CM^2	N/CM^2 Sec	N/CM^2	N/CM^2
BATH	PDS-30259 (BATH)	0.0	-11.83	1.5(9) 1.5(9)	1.5(9) 1.5(9)	1.5(10) 1.5(10)	6.0(14) 4.5(14)	2.2(18) 1.6(18)	2.2(19) 1.6(19)	5.0(15) 3.6(15)	1.8(19) 1.3(19)	1.8(20) 1.3(20)	2.6(14) 2.0(14)	9.4(17) 7.0(17)	9.4(18) 7.0(18)
LEAD	LEAD	0.0	-22.93	6.1(7) 6.0(7)	6.1(7) 6.0(7)	6.1(8) 6.0(8)	1.7(13) 1.3(13)	6.0(16) 4.7(16)	6.0(17) 4.7(17)	6.5(13) 4.9(13)	2.3(17) 1.8(17)	2.3(18) 1.8(18)	1.3(12) 9.5(11)	4.7(15) 3.4(15)	4.7(16) 3.4(16)
Al	QQ-A-225/8 (6061-T6)	0.0	-10.83	1.8(9) 1.8(9)	1.8(9) 1.8(9)	1.8(10) 1.8(10)	7.8(14) 6.0(14)	2.8(18) 2.1(18)	2.8(19) 2.1(19)	6.3(15) 4.5(15)	2.3(19) 1.6(19)	2.3(20) 1.6(20)	6.2(14) 4.7(14)	2.2(18) 1.7(18)	2.2(19) 1.7(19)
Al	PDS-30258-1 (6061-T6)	25.375	-11.04	4.9(8) 5.1(8)	4.9(8) 5.1(8)	4.9(9) 5.1(9)	1.7(14) 1.3(14)	6.3(17) 4.8(17)	6.3(18) 4.8(18)	1.2(15) 8.5(14)	4.2(18) 3.1(18)	4.2(19) 3.1(19)	1.3(14) 9.7(13)	4.7(17) 3.5(17)	4.7(18) 3.5(18)
Al	QQ-A-250/11 (6061-T651) MIL-B2995	0.0	-10.83	SAME AS ABOVE R = 0.0; Z = 10.83											
CRES	QQ-S-763 304 Cond. A (SS-304)	0.0	-24.23	1.1(7) 1.1(7)	1.1(7) 1.1(7)	1.1(8) 1.1(8)	1.1(13) 8.5(12)	4.0(16) 3.1(16)	4.0(17) 3.1(17)	4.4(13) 3.3(13)	1.6(17) 1.2(17)	1.6(18) 1.2(18)	4.9(12) 3.6(12)	1.8(16) 1.3(16)	1.8(17) 1.3(17)
CRES	AMS-5525 (A-286)	27.00	-21.94	5.7(6) 5.7(6)	5.7(6) 5.7(6)	5.7(7) 5.7(7)	3.9(12) 3.0(12)	1.4(16) 1.1(16)	1.4(17) 1.1(17)	1.1(13) 8.5(12)	4.1(16) 3.1(16)	4.1(17) 3.1(17)	6.6(11) 4.8(11)	2.4(15) 1.7(15)	2.4(16) 1.7(16)
CRES	AMS-7254 (SS-18-B)	0.0	-10.83	SAME AS ABOVE R = 0.0; Z = 10.83											

C = Composite Core
G = Graphite Core

Numbers in () refer to Powers of Ten; i.e., 1.5 (9) = 1.5×10^9

TABLE 7-4
SUMMARY OF SHIELD COMPONENT MATERIAL TEMPERATURES
AND OTHER KEY PERFORMANCE PARAMETERS (GRAPHITE CORE)

COMPONENT	LOCATION	PARAMETER	NORMAL MODE OPERATION						COOLDOWN* OPERATION RANGE °R	SPACE SOAK RANGE	
			RATED PERFORMANCE 99/95 TOLERANCE LIMIT			THRUSTLE HOLD 99/95 TOLERANCE LIMIT				Solar Flux ⁽³⁾ of 300 Btu Hr/Ft ²	Solar Flux ⁽⁴⁾ of 3000 Btu Hr/Ft ²
			NOMINAL	UPPER	LOWER	NOMINAL	UPPER	LOWER			
Peripheral Shield BATH Cluster	Aft End	T max (°R)	446	470	422	448	476	420	(300 - 600) ⁽¹⁾ with 800 max ⁽²⁾	350 - 550	550 - ≈ 900
		T avg (°R)	372	392	352	398	423	363			
		T min (°R)	305	321	289	345	366	324			
		(T avg - T min) (°R)	67	89	46	53	70	36			
Peripheral Shield Lead Sector	Forward End	T max (°R)	310	326	294	345	366	324	(300 - 600) ⁽¹⁾ with 800 max ⁽²⁾	350 - 550	550 - ≈ 900
		T avg (°R)	304	320	288	343	364	322			
		T min (°R)	300	315	285	340	361	319			
		(T avg - T min) (°R)	4	5.3	207	3	4	2			
Peripheral Shield		Axial Pressure difference ΔP (psid)	11	12.5	8.5	8	9.2	6.8			
Peripheral Shield		Fluid Temperature Rise (°R)	2	2.5	1.5	2	2.5	1.5			
Central Shield BATH Cluster	Aft End	T max (°R)	410	434	386	419	445	393	(280 - 600) ⁽¹⁾ with 800 max ⁽²⁾	350 - 550	550 - ≈ 900
		T avg (°R)	393	416	370	408	434	382			
		T min (°R)	357	377	337	383	417	359			
		(T avg - T min) (°R)	36	48	24	25	33	17			
Central Shield Lead Cluster	Forward End	T max (°R)	293	308	278	333	353	313	(280 - 600) ⁽¹⁾ with 800 max ⁽²⁾	350 - 550	550 - ≈ 900
		T avg (°R)	292	297	277	332	352	312			
		T min (°R)	290	295	275	330	350	310			
		(T avg - T min) (°R)	2	2.6	1.4	2	2.5	1.5			

TABLE 7-4 (CONT' D)
SUMMARY OF SHIELD COMPONENT MATERIAL TEMPERATURES
AND OTHER KEY PERFORMANCE PARAMETERS (GRAPHITE CORE)

COMPONENT	LOCATION	PARAMETER	NORMAL MODE OPERATION						COOLDOWN OPERATION RANGE °R	SPACE SOAK RANGE ¹	
			RATED PERFORMANCE 99.95 TOLERANCE LIMIT			THROTTLE HOLD 99.95 TOLERANCE LIMIT				Solar Flux ⁽³⁾ of 300 Btu Hr/Ft ²	Solar Flux ⁽⁴⁾ of 3000 Btu Hr/Ft ²
			NOMINAL	UPPER	LOWER	NOMINAL	UPPER	LOWER			
Aft Central Shield Support Plate	Aft Face	T max (°R)	400	423	377				(280 - 600) ⁽¹⁾ with 800 max ⁽²⁾	350 - 550	550 - 900
		T avg (°R)	385	408	362						
		T min (°R)	360	380	340						
		T avg - T min (°R)	15	21	9						
Central Shield		Axial Pressure Difference, (psid)	11	12.1	8.9	8	8.9	7.1	(280 - 600) ⁽¹⁾		
Central Shield		Fluid Temperature Rise	4	5.2	2.8	3	4.1	1.9	800 ⁽²⁾		

1. This is the temperature range within 3.65 hours after a ten minute burn and throttle hold, as reported in WANL-TME-2678, "Systems Analysis Summary Report, Nuclear Subsystem," Section 5.2.6.3.

2. This is the maximum allowable temperature between cooling pulses.

3. This is the minimum flux shown in WANL-TME-2678, "Systems Analysis Summary Report, Nuclear Subsystem," Section 5.2.6.3. Solar flux in earth orbit is 444 BTU/HR-FT² as reported in WANL-TME-2678.

4. This is the extreme solar flux, in the vicinity of Mercury, as reported in WANL-TME-2678.

1. This is the temperature range within 3.65 hours after a ten minute burn and throttle hold, as reported in WANL-TME-2678, "Systems Analysis Summary Report, Nuclear Subsystem," Section 5.2.6.3.

2. This is the maximum allowable temperature between cooling pulses.

3. This is the minimum flux shown in WANL-TME-2678, "Systems Analysis Summary Report, Nuclear Subsystem," Section 5.2.6.3. Solar flux in earth orbit is 444 BTU/HR-FT² as reported in WANL-TME-2678.

4. This is the extreme solar flux, in the vicinity of Mercury, as reported in WANL-TME-2678.

TABLE 7-5
MECHANICAL PROPERTIES OF IRRADIATED BATH MATERIALS⁽¹⁾

Test Temp °R	Fluence n/cm ²	0.2% Offset Yield Strength KSI	Ultimate Tensile Strength KSI	Elongation %	Compressive Yield Strength KSI
140	0	16.9	23.0	2.6	18.5
		11.1	13.8	.6	18.5
		15.0	22.1	4.0	-
	9 x 10 ¹⁸	22.2	24.8	0.7	22.0
		18.0	21.1	0.7	24.1
		13.8	19.0	0.8	24.1
	0	11.2	19.2	2.5	12.0
		11.9	19.3	2.5	12.2
		11.3	18.2	4.1	11.5
540	9 x 10 ¹⁸	14.5	18.4	1.0	14.8
		14.9	20.1	1.6	16.2
		14.4	20.4	1.5	14.8
	0	11.0	14.2	3.2	10.0
		9.4	12.4	7.1	9.5
		8.5	12.4	4.5	-
	9 x 10 ¹⁸	11.3	15.5	2.5	9.7
		9.7	13.7	4.0	9.2
		9.7	13.6	3.0	-
800	0	11.0	14.2	3.2	10.0
		9.4	12.4	7.1	9.5
		8.5	12.4	4.5	-
	9 x 10 ¹⁸	11.3	15.5	2.5	9.7
		9.7	13.7	4.0	9.2
		9.7	13.6	3.0	-
	0	11.0	14.2	3.2	10.0
		9.4	12.4	7.1	9.5
		8.5	12.4	4.5	-

Figure 7-1. Thermal Conductivity of Irradiated BATH Material ⁽¹⁾

LIST OF REFERENCES
(CHAPTER 7 - INTERNAL SHIELD)

1. Shogan, R. P., "Irradiation of BATH Shield Material at Ambient Temperatures," WANL-TME-1898, Westinghouse Astronuclear Laboratory (March 1969).
2. Allen, J. W., General Dynamics, Fort Worth, Texas, Personal Communication. (November 1970).
3. Schollenberger, R. S., DRM 51229, Westinghouse Astronuclear Laboratory (October 1970).
4. Schollenberger, R. S., DRM 51250, Westinghouse Astronuclear Laboratory (August 1970).

RADIATION EFFECTS INFORMATION REQUESTS (REIRs)

(CHAPTER 7 - INTERNAL SHIELD)

<u>Number</u>	<u>Description</u>
9	Aluminum 6061-T651 Plate - Tensile and Fracture Toughness (Screening)
10	BATH - Fracture Toughness
11	BATH - Thermal Conductivity

IDENTIFICATION NO. 9

RADIATION EFFECTS INFORMATION REQUEST

DATE November 10, 1970

SUBJECT ALUMINUM 6061-T651 PLATE FUNCTIONAL CATEGORY Test & Evaluation

REF. ITEM SUPPORT EC 677562 REF. DATA ITEMS S-110

INFORMATION DESIRED Aluminum 6061-T651 Plate
(PURPOSE OF STUDY OR EXPERIMENT) Effects of irradiation on strength and ductility as well as fracture toughness (if
material is brittle in any direction)

DATE REQUIRED CY 71

RELATION OF ITEM TO ENGINE Plate which supports Internal Shield between Reactor and External Shield
(LOCATION)

PER DRAWINGS, DATA ITEMS Drawing 939J741

ENVIRONMENT (4):
(Ref. data items) NEUTRON FLUX _____ NEUTRON FLUENCE 3×10^{18} and 3×10^{19} n/cm² (E>Mev)
GAMMA DOSE RATE _____ GAMMA DOSE _____
TEMPERATURE 140°R - 750°R NUCLEAR HEATING RATE _____
PRESSURE _____ OTHER _____
FLUID _____
CYCLE REQUIREMENTS TBD

RELIABILITY ESTIMATE Screening* REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL Screening* APPROVED _____ DATE _____

APPROVED _____ DATE _____

SPECIAL CONSIDERATIONS _____ APPROVED _____ DATE _____

PERTINENT REFERENCES _____

*If Material is brittle in this application, Cat. "A" fracture roughness data will be required (99% Reliability at 95% C.L.)

IDENTIFICATION NO. 10

RADIATION EFFECTS INFORMATION REQUEST

DATE November 10, 1970

SUBJECT BATH FUNCTIONAL CATEGORY Test & Evaluation

REF. ITEM SUPPORT EC 677562 REF. DATA ITEMS S-110

INFORMATION DESIRED BATH - Fracture Toughness and Annealing Characteristics following Irradiation
(PURPOSE OF STUDY OR EXPERIMENT)

DATE REQUIRED CY 71

RELATION OF ITEM TO ENGINE Internal Shield between Reactor and External Shield
(LOCATION)

PER DRAWINGS, DATA ITEMS Drawing 939J741

ENVIRONMENT (4): NEUTRON FLUX _____ NEUTRON FLUENCE 2.1×10^{18} and 2.1×10^{19} n/cm² (E > Mev)
(Ref. data items) GAMMA DOSE RATE _____ GAMMA DOSE _____

TEMPERATURE 140°R - 740°R NUCLEAR HEATING RATE _____

PRESSURE _____ OTHER _____

FLUID _____

CYCLE REQUIREMENTS TBD

RELIABILITY ESTIMATE 99% REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL 95% APPROVED _____ DATE _____

APPROVED _____ DATE _____

SPECIAL CONSIDERATIONS _____ APPROVED _____ DATE _____

PERTINENT REFERENCES _____

IDENTIFICATION NO. 11

RADIATION EFFECTS INFORMATION REQUEST

DATE November 10, 1970

SUBJECT BATH FUNCTIONAL CATEGORY Test & Evaluation

REF. ITEM SUPPORT EC 677562 REF. DATA ITEMS S-170

INFORMATION DESIRED BATH - Thermal Conductivity following Irradiation and Annealing Characteristics
(PURPOSE OF STUDY OR EXPERIMENT)

DATE REQUIRED CY 71 - 72

RELATION OF ITEM TO ENGINE Internal Shield between Reactor and External Shield
(LOCATION)

PER DRAWINGS, DATA ITEMS Drawing 939 J741

ENVIRONMENT (4): NEUTRON FLUX _____ NEUTRON FLUENCE 2.1×10^{18} and 2.1×10^{19} n/cm² (E>Mev)
(Ref. data items) GAMMA DOSE RATE _____ GAMMA DOSE _____

TEMPERATURE 140°R - 740°R NUCLEAR HEATING RATE _____

PRESSURE _____ OTHER _____

FLUID _____

CYCLE REQUIREMENTS TBD

RELIABILITY ESTIMATE Screening REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL Screening APPROVED _____ DATE _____

APPROVED _____ DATE _____

SPECIAL CONSIDERATIONS _____ APPROVED _____ DATE _____

PERTINENT REFERENCES _____

REPORTS (T-11~~9~~) RESPONDING TO REIRS

(None prepared at this time)

8.0 REFLECTOR ASSEMBLY

8.1 SUMMARY AND CONCLUSIONS

The applications of Aluminum 6061 (T6, T61, T652, T4), Beryllium, Copper-Boron, SS 304 Cond. A (annealed), SS 440C, SS 321, A-286 and copper rivets in the Reflector Assembly (RA) of the NSS have been reviewed. Aluminum alloys (all forms) are known to exhibit increased strength and decreased ductility following irradiation. A possible exception is the short transverse direction of the T61 barrel forging which may become brittle due to irradiation at cryogenic temperatures. A REIR for a screening test in this material has been issued. A second possible exception involves the low cycle fatigue properties of this material and another REIR has been issued for that purpose. In all other cases SOL design or unirradiated properties are applicable.

Beryllium swelling was shown to be negligible at NSS RA conditions. Beryllium mechanical properties do undergo some degradation. However, the beryllium is attached to an aluminum forging and is not loaded during NSS operation. On the basis of engineering judgement, additional information on Dynamic Modulus and Cyclic Crack Growth has been requested. Also, additional study of irradiation effects on beryllium thermal conductivity is required because the drastic changes in this property at low temperatures must be quantitated for design purposes.

Copper Boron is shown to be a brittle material and a quantification of the changes in fracture toughness is required for design. In addition, additional tests to measure changes in its thermal conductivity under NSS RA conditions will be required for design.

The stainless steels used in this ECC are known to undergo increases in strength and decreases in ductility at NSS RA fluences and temperatures. Since they remain ductile, SOL design conditions will apply and no additional data is required for most applications. However, additional tests on A-286 spring stress relaxation is required to allow adequate pre-setting of springs for design purposes.

The graphite materials in the RA will be analyzed in Chapter 5, Core Periphery.

8.2 INTRODUCTION

8.2.1 Item Description

The primary function of the Reflector Assembly (RA) is to regulate the reactivity level of the core by controlling the return of neutrons to the core. It must also be structurally adequate to transmit loads imposed upon it by other components within the pressure vessel as well as the fluid pressure differentials to which it is exposed.

8.2.2 Location

The Reflector Assembly is located within the pressure vessel and surrounds the NSS core and core periphery. It is located axially between the nozzle flange and the core support plate.

8.2.3 Description of Sub-Components

The Reflector Assembly (RA) consists of an aluminum structural cylinder, its forward extension and aft support rings, reflector segments, control drums with standoffs and pads, and the additional subcomponents described below.

A more detailed description of the RA may be found in Data Item E-105. The bill of materials for the RA has been extracted from Reference Drawing 939J733 and presented in Table 8-1. Table 8-2 summarizes the materials used in this component by generic names.

8.2.3.1 Structure

The primary support structure is the Aluminum (6061 - T61 barrel forging) structural cylinder which is utilized to transmit the NSS axial and radial loads to the interfacing nozzle flange and ultimately to the aft pressure vessel flange. Additionally, the structure must withstand the radial fluid pressure drop and support the reflector beryllium segments and control drums.

8.2.3.2 Forward Extension Ring and Aft Support Ring

The forward extension ring (aluminum 6061 - T 61 forging) is an extension of the aluminum structural cylinder which mates with the core support plate at its forward surface. The radial flange of the ring supports the axial and radial load developed by the 18 control drums. The outermost axial cylindrical portion of the forward support ring serves the function of a coolant flow restrictor for the pressure vessel coolant flow passage.

The aft support ring (aluminum 6061 - T6) located at the aft end of the aluminum structure positions the aft end of the control drums and the reflector baffle can.

8.2.3.3 Standoff, Pad and Clamp Ring

A set of 18 A-286 standoffs are located at the aft end of the RA to provide an axial loading capability to force the aluminum structure-nozzle interface together. The load is transmitted from the standoffs to the aluminum 6061 - T6 pads which are bolted to the pressure vessel interior. An additional aluminum 6061 - T6 clamp ring is added to assure closure of the aluminum structure nozzle interface.

8.2.3.4 Locating Cylinder

The locating cylinder (aluminum 6061 - T652) is sandwiched between the standoff pads and the forward face of the aft pressure vessel flange. This cylinder centers the NSS within the pressure vessel prior to nozzle insertion and supports the NSS during assembly.

8.2.3.5 Reflector Segments

Beryllium segments, which are not part of the RA support system, are located within the aluminum structure to provide neutron reflection.

8.2.3.6 Control Drums

Eighteen control drums are located within the beryllium portion of the RA. The cylindrical beryllium control drums have Copper-Boron vanes mounted at the outer diameter.

The control drum is encased within a thin walled Aluminum 6061 T-6 structure which is welded to aluminum bearing shafts located at each end of the control drum.

8.2.3.7 Reflector Baffle Can and Control Drum Bore Liner

The entire RA is surrounded by a thin aluminum 6061 T6 cylinder which provides a circumferential barrier between the pressure vessel and the RA. Tubular control drum bore liners (6061-T6) are located between the forward and aft bearing support rings to direct coolant flow.

8.2.3.8 Seal Segments and Insulation Segments

They will be discussed in Chapter 5, Core Periphery which will also present a detailed review of the effects of radiation on graphites.

8.2.3.9 Lateral Support Spring, Plunger, and Capsule Assembly

They will be discussed in Chapter 5, Core Periphery.

8.3 ENVIRONMENTAL AND OPERATIONAL REQUIREMENTS

8.3.1 Nuclear Environment

Table 8-3 presents the maximum gamma and neutron radiation levels (plus 3 sigma) experienced by sub-components of the Reflector Assembly. Thus, the radiation levels presented in this table represent the most severe levels experienced by each material in the RA. The radiation levels for both the composite and graphite cores are given. However, the more conservative levels associated with the composite core are used in this analysis.

8.3.2 Thermal Environment

Table 8-4 represents the temperature ranges (including uncertainties) experienced by each material during typical operation modes, cooldown, and storage.

8.4 ANALYSIS

Each of the materials listed in Table 8-2 is analyzed in this section for the most critical application(s) in the RA.

8.4.1 Aluminum 6061 - T6 (Bar and Forging)

This material is utilized for a variety of structurally loaded and non-loaded applications in the NSS RA. The most important applications are the aft support ring, standoff pads, ring clamp, and control drums. Several other applications are less critical, but important nevertheless. These include the reflector can, drum shafts, drum supports and housing and various caps, pins, O-rings, nuts, etc.

8.4.1.1 Non-NERVA Data

As discussed in Section 6.4.1.1, aluminum alloys have been extensively utilized in the nuclear industry and their properties following irradiation are well known. As depicted in Table 8-5, an invariable increase in strength and decrease in ductility is found over a wide range of temperatures and fluences⁽¹⁾. These data have been confirmed by several investigations as discussed in Section 6.4.1.1. Aluminum 6061 - T6 appears to behave as do all other aluminum alloys in exhibiting these trends.

8.4.1.2 NERVA Data

Aluminum alloys investigated in the NERVA Program show the same invariable decreased ductility and increased strength observed for other aluminum alloys.⁽⁴⁾ These data have been confirmed by Schwanbeck⁽²⁾ in NERVA studies conducted at Plum Brook and in a summary paper by Lombardo et al.⁽³⁾ As discussed in Section 6.4.1.2, estimates of the properties of Aluminum 6061 - T6 were made for NSS PDR.⁽⁵⁾ These estimates also depict the same predictable increased strength and decreased ductility following irradiation as has been seen in NERVA and non-NERVA investigations of aluminum alloys. There has been no attempt to estimate ten hour properties because the RA reaches a maximum temperature of 710 - 760°R during cooldown following a one-hour firing and nearly complete annealing has been shown to occur in aluminum at that temperature. WANL has classified this material as ductile under all conditions of use in the NSS RA.⁽⁶⁾

8.4.1.3 Conclusions

Aluminum 6061 - T6 has been demonstrated to remain ductile over all temperatures and fluences experienced in the NSS RA. Thus, start of life (SOL) design conditions are applicable. The slight increases in strength and decreases in ductility which occur following irradiation are not considered for design purposes. Thus, no additional Radiation Effects testing is required and no REIRs are issued.

8.4.2 Aluminum 6061 - T61 (Barrel Forging)

This material will be utilized in the RA in the structural barrel and forward support ring, both of which are important axially loaded NSS structures.

8.4.2.1 Non-NERVA Data

The discussion of Section 6.4.1.1 is generally applicable because of the general observation that all aluminum alloys remain ductile with slight increases in strength following irradiation. There is no directly applicable non-NERVA data for this material.

8.4.2.2 NERVA Data

There has been no experimental investigation of aluminum 6061 - T61 in the NERVA Program. However, as discussed in Section 6.4.1.2, similar aluminum alloys show an invariable increase in strength and decrease in ductility at NERVA fluences and temperatures. Estimates of the unirradiated and irradiated properties of aluminum 6061 - T61 forgings (ring and plate) have been made for ten and sixty minute equivalent NSS irradiations.^(7,8) Although these data have been plotted for study, they have not been included in this report. As is true for other aluminum alloys, these estimates depict small increases in strength and decreases in ductility following irradiation. There is no attempt to estimate ten hour irradiated properties because the RA reaches a maximum temperature of 710-760°R during cooldown following a one hour firing and nearly complete annealing has been shown to occur in aluminum at that temperature. WANL has classified

that material as ductile under all use conditions except in the short transverse direction at cryogenic temperatures where it is tentatively classified as brittle⁽⁶⁾ on the basis of engineering judgement.

There is also some question regarding the irradiated fatigue properties of this material for NERVA applications. This is primarily based upon engineering judgement, extrapolating from other aluminum data.

8.4.2.3 Conclusions

Aluminum 6061 - T61 barrel forging is likely to remain ductile over most conditions. However, because of the extreme importance of the structural cylinder as well as the question of its ductility at cryogenic temperatures in the short transverse direction, REIR No. 12 has been issued to specifically investigate its tensile and fracture toughness properties. If it remains ductile under all conditions, SOL or unirradiated design conditions will be applicable. If not, additional fracture toughness tests will be required to determine EOL conditions.

In addition, the fatigue properties of this material are also of concern for design but are not presently quantitated. Thus, REIR No. 13 has been issued for this purpose.

8.4.3 Aluminum 6061 - T652 (Forged Ring)

This material is used for the locating cylinder and is not structurally loaded after complete engine assembly. However, it does experience pressure differential loads.

8.4.3.1 Non-NERVA Data

There is no non-NERVA data available on this material. However, as pointed out in Section 6.4.1.1, all aluminum alloys are likely to exhibit increased strength and decreased ductility following irradiation.

8.4.3.2 NERVA Data

There has been no experimental investigation of this material in the

NERVA Program. However, it will certainly exhibit the same increased strength and decreased ductility following irradiation which characterized the NERVA studies of other aluminum alloys.

8.4.3.3 Conclusions

Both because of the nature of the loads applied to this material in the RA following assembly, and because of its expected properties following irradiation, no REIRs are issued

8.4.4 Aluminum 6061 - T4

This material (plate/sheet) is used for non-loaded filler segments and retainers. No additional analysis is required because this material does not represent a design problem. SOL properties can be utilized where required since the properties of irradiated aluminum tend to be predictable as cited elsewhere in this report.

8.4.5 Beryllium

This material is used in non-structural (encased) segments of the RA and is attached to the aluminum barrel forging.

8.4.5.1 Non-NERVA Data

Beryllium is an excellent reflector for neutrons and has the lowest neutron-absorption cross section of all the metals because of its low atomic weight and high atomic density. Beryllium reflectors have been and are being used in a number of testing reactors such as the Materials Testing Reactor, Engineering Test Reactor, Experimental Breeder Reactor, Advanced Engineering Test Reactor, Oak Ridge Research Reactor, the High Flux Isotope Reactor, Pegasus and Osiris in France and the Japanese Materials Testing Reactor.

Beryllium has a high modulus of elasticity, low density, high melting point, high specific heat and low thermal expansion. However, its limited ductility is further degraded by cryogenic irradiations or by high fluence and dimensional changes also occur under the latter condition at high temperature. These effects impose limitations on the nuclear applications of beryllium.

Investigations of the effects of neutron irradiation on beryllium have been conducted since the late fifties, and reports have been made on the swelling of beryllium reflector parts in the MTR⁽⁹⁾ and in the ORR⁽¹⁰⁾ after varying lengths of operating time. Swelling occurs because of $(n, 2n)$ and (n, α) capture reactions with fast neutrons and several reactions with thermal neutrons produce helium and hydrogen gas. Several investigations at temperatures exceeding 1000°R .⁽¹¹⁻¹⁴⁾ have measured significant quantities of gas at fluences of 10^{20} to 10^{22} n/cm^2 ($E > 1$ Mev) or otherwise confirmed significant beryllium transmutation. However, at similar fast fluences (10^{21} - 10^{22} n/cm^2) and temperatures (600°R - 2000°R), other investigators found no measureable swelling in Beryllium.^(15,16) Barnes⁽¹⁷⁾ used electron microscopy to conclude that helium gas remains in solid solution at low temperatures and diffuses into larger bubbles at higher temperatures.⁽¹⁷⁾ Where beryllium reflector specimens from the MTR were irradiated near room temperature (620°R) to a fluence of 10^{21} n/cm^2 ($E > 1$ Mev), no appreciable swelling was found until temperatures of 1500°R were attained.⁽¹⁸⁻²⁰⁾ This is shown diagrammatically in Figure 8-1.

Thus, it appears that the threshold temperature for swelling in beryllium at fluences above 10^{20} n/cm^2 ($E > 1$ Mev) is 1500°R - 1800°R . Tromp has determined the swelling threshold temperatures for ETR and MTR beryllium specimens and this data is shown in Figure 8-2.⁽²¹⁾ Although at much higher fluences than the NSS RA, these data demonstrate that measureable swelling does not occur below 1500°R .

Beryllium thermophysical properties have also been shown to change due to irradiation.⁽²²⁾ At a fast fluence of 10^{19} n/cm^2 ($E > 1$ Mev), the measured decrease in thermal conductivity is 0.031 $\text{w/cm}^{\circ}\text{K}$ at 40°R and 0.047 $\text{w/cm}^{\circ}\text{K}$ at 60°R . As shown in Figure 8-3, the beryllium thermal conductivity, over a wide temperature range, decreases with radiation. Other investigators have shown similar results.

Irradiation of beryllium by fast neutrons produces marked effects on mechanical properties which vary with irradiation temperature, test temperature, fluence, and material characteristics as determined by fabrication techniques.

At $5 \times 10^{19} \text{ n/cm}^2$ ($E > 1 \text{ Mev}$), and irradiation temperature of 630°R to 670°R , little changes in tensile properties were observed at test temperatures between 540°R and 1750°R .^(13,23) However, these same investigators⁽¹³⁾ irradiated a different form of beryllium (extrusions from electrolytic powder at 2380°R), 630°R - 670°R and found a linear increase in yield strength and decrease in ductility as a function of fluence where testing was performed at room temperature. (Figure 8-4)

Other investigators have shown increased strength and hardening by fast neutron irradiation to higher fluences⁽²⁴⁻²⁸⁾ which anneals out at higher test temperatures. The mechanism of hardening by fast neutron irradiation is believed to be caused by the interstitial and vacancy defects produced by the transmuted helium atoms. At low testing temperatures, the helium atoms stay in the beryllium matrix in the atomic form producing solid solution hardening. Both the helium atoms and the defects impede the movement of dislocation, which increases the strength and decreases the ductility of beryllium. At higher temperatures, annealing occurs and the irradiation-induced changes in the tensile properties of beryllium are reduced. The temperature range over which these changes occur are closely related to material fabrication techniques. Some limited work performed on extruded beryllium specimens prepared transverseto the extrusion direction⁽²⁵⁾ at $1100 - 1570^\circ\text{R}$ and fluence of 10^{20} n/cm^2 ($E > 1 \text{ Mev}$) showed no change in the already brittle transverse specimens.

8.4.5.2 NERVA Data

The data generated from the NERVA experimentation program for beryllium are indicated in Tables 8-5, 8-6, and 8-7.⁽²⁹⁾ The irradiation to $2.4 \times 10^{17} \text{ n/cm}^2$ ($E > 1 \text{ Mev}$) and $7.6 \times 10^{17} \text{ n/cm}^2$ ($E > 1 \text{ Mev}$) produced significant changes in the mechanical properties of S-200 beryllium, particularly in ultimate tensile strength, plastic strain in tension and notched tensile strength, at low temperatures. At room temperature, no significant changes were observed following irradiation. The ultimate tensile strength of specimens irradiated at 140°R increased to the control values after five minute anneals at $340 - 540^\circ\text{R}$.

Complete annealing was observed to occur at increasingly higher temperatures for higher fluences. Plastic strain changes were found to anneal out over longer time intervals, although complete recovery was not observed even after 1000 minutes at 540°R following irradiation to $7.6 \times 10^{17} \text{ n/cm}^2$ ($E > 1 \text{ Mev}$) at 140°R. The same general results have been observed from an extensive irradiation program conducted for beryllium in GTR 20-C.⁽³⁰⁾ However, the test agency data report has not been received and no analysis was possible prior to release of this Data Item.

Some limited thermal conductivity studies were conducted in GTR 21 and demonstrated that the thermophysical properties of beryllium changed drastically at NSS RA fluences and temperatures.⁽³¹⁾ Because of these results, a program has been undertaken at Oak Ridge National Laboratory. Unirradiated data are shown in Figure 8-5 and irradiated data in Figure 8-6.⁽³²⁾ Thermal conductivity was seen to decrease by greater than a factor of two at $2 \times 10^{18} \text{ n/cm}^2$ ($E > 1 \text{ Mev}$) and has recently been observed to decrease by a factor of four at $4-5 \times 10^{18} \text{ n/cm}^2$ ($E > 1 \text{ Mev}$)⁽³³⁾ where irradiations were conducted at 180°R - 200°R. Annealing results in Figure 8-6 show complete recovery by room temperature.

Estimates for beryllium irradiated properties have been prepared for NSS PDR.⁽³⁴⁾ Beryllium is classified as brittle,⁽⁶⁾ and all experimental data, NERVA and non-NERVA alike, substantiate that fact. These estimates⁽³⁴⁾ have been plotted, although the data is not enclosed herein for brevity. They show a decreased fracture toughness K_{IC} value for beryllium at all temperatures due to irradiation, at $2 \times 10^{18} \text{ n/cm}^2$ ($E > 1 \text{ Mev}$) although the magnitude of the difference between unirradiated and irradiated values decreases with increased temperature as would be expected. These trends agree with those cited in Section 8.4.5.1 above. This difference is further increased, according to these estimates, at fluences of $1 \times 10^{19} \text{ n/cm}^2$ ($E > 1 \text{ Mev}$) and in a hydrogen environment.

Beryllium irradiated thermal conductivity decreases are also estimated⁽³⁴⁾ for 10 and 60 minute irradiations. There is insufficient data from NERVA tests that are under way and from the data released from the test agency, to conclusively determine the validity of

these estimates. However, these estimates do show the same drastic low temperature loss in thermal conductivity which has been observed in the Oak Ridge tests cited above. These observations are further confirmed by the cracking observed in the NRX A-6 reflector which was at least partially attributed to thermal stress introduced in the beryllium due to decreased thermal conductivity toward the end of irradiation (NRX A-6 power operation).

8.4.5.3 Conclusion

Regarding swelling, there is no available NERVA data. However, the non-NERVA data cited above indicates that swelling will not occur at the NERVA operating temperatures. In addition, swelling is unlikely at NSS RA cooldown temperatures, especially at the relatively low fluences experienced. No further tests are required and the problem of swelling apparently need not be considered for design purposes.

Regarding beryllium mechanical properties, NERVA and non-NERVA data show that beryllium is brittle and that it undergoes nearly complete annealing following irradiation to NSS RA one hour fluences at room temperature. Thus, one hour irradiated K_{IC} values are applicable. GTR 20C test results, as yet not received from the test agency, will be "A" quality data which will replace the estimates made in the DRM cited above. However, there is no reason to doubt that they will agree closely with those estimates which are currently being used. Of more importance, the beryllium segments are not structurally loaded, as has been the case in the past, and even significant changes in irradiated K_{IC} values will not substantially affect present design.

On the basis of engineering judgment, the WANL Materials Department has also suggested that irradiation tests be conducted to determine the Dynamic Modulus and Cyclic Crack Growth properties of beryllium. REIRs No. 14 and 15 reflect these requirements.

The most important parameter following irradiation, for the present NSS design is thermal conductivity. Accurate tables of beryllium thermal conductivity changes, as a function of both fluence (one hour NSS equivalent) and temperature are required for design purposes and REIR No. 16 documents this requirement.

8.4.6 Copper-Boron¹⁰ (18%)

This material is used exclusively for control vanes mounted within a recess at the outer diameter of the beryllium control drums. There are minimal structural loads imposed on these vanes. However, crumbling, or general deterioration of these structures would be unacceptable.

8.4.6.1 Non-NERVA Data

There is no non-NERVA data on Copper-Boron composites. However, similar materials, such as Cu B₄C have been utilized in the past without apparent difficulty.

There is some information available for the effects of irradiation on copper. Typical increases in strength were observed over the range from $5 \times 10^{17} - 5 \times 10^{20}$ n/cm² (energy not specified)⁽³⁵⁾ and these results are shown in Table 8-8. Greenfield and Wilsdorf examined the mechanism of radiation hardening in copper⁽³⁶⁾ and concluded from electron microscopy that it is caused by profuse kinking of the glide dislocations and the blocking of free glide planes. Others have verified this radiation-induced strengthening.⁽³⁷⁻⁴⁰⁾ Coltman et al⁽⁴¹⁾ have studied the radiation-induced changes in copper thermal conductivity and demonstrated that substantial changes do occur although annealing is also seen to occur in stages at relatively low temperatures.⁽⁴²⁾

8.4.6.2 NERVA Data

A limited amount of data is available for the Cu 24v/o B alloy irradiated at ambient temperatures at Plum Brook.⁽⁴³⁾ The results are shown in Table 8-9. They demonstrate that the low ductility of this alloy is reduced even farther by irradiation and that its strength remains unchanged. Additional CuB specimens have been irradiated in GTR 20C. The results are not available at the publication of this report. In addition, Aluminum-Boron composites have been used in prior NERVA reactors without apparent difficulty.

Estimates of Copper 18 v/o Boron (natural and enriched) composites K_{IC} have been made.⁽⁴⁴⁾ Fracture strength is seen to decrease with irradiation at low temperatures as

could be expected for a brittle material such as Copper-Boron. Experimental observations are being made in GTR 22 and GTR 23 to generate higher quality data to supplement these estimates. Since the structural loads imposed on this material are minimal, there is no reason to suspect performance degradation at NSS RA fluences and temperatures.

It has been estimated that this material's thermal conductivity is reduced by radiation⁽⁴⁴⁾ and that the magnitude of the difference decreases with increasing temperature.

8.4.6.3 Conclusions

Because brittle design criteria are applicable, EOL or one hour design data may be required. Although it is expected that the Cu 18 v/o B composite will behave as predicted in the DRM cited above, REIR No. 17 has been issued to verify these data and fracture toughness tests are planned for GTR 22 and GTR 23. In addition, because of the importance of this material to NSS design, its thermal conductivity properties will require some experimental data to supplement the estimates cited above. It is probable that screening data will suffice and REIR No. 18 is issued for that purpose.

8.4.7 Copper

This material is used for control plate rivets. Information available on the irradiated properties of copper are cited in Section 8.4.6.1. Both because of the application of this material in the NSS RA and because its strength has been seen to increase with radiation, no additional analysis is required.

8.4.8 Stainless Steel 304 (Annealed)

This material is used for miscellaneous screws, nuts, and locking devices in the NSS RA. These applications are important in maintaining the integrity of the RA.

8.4.8.1 Non-NERVA Data

As discussed in Section 6.4.6.1, the austenitic stainless steels, especially AISI 304 have been extensively investigated and extensively utilized in radiation environ-

ments. They exhibit increased strength and decreased ductility following irradiation to NSS RA fluences and temperatures, but remains ductile over all conditions of use.

8.4.8.2 NERVA Data

Although there is no NERVA experimental data, estimates of its irradiated properties were made for NSS PDR.⁽⁴⁵⁾ As discussed in Section 6.4.6.2, SS 304 remains ductile, but shows increased strength and decreased ductility following irradiation to NSS RA fluences and temperatures.

8.4.8.3 Conclusions

As discussed in Section 6.4.6.3, SOL or unirradiated design conditions are applicable for SS 304. No additional testing is required because of its use in the NSS RA.

8.4.9 SS 440C

This material is planned for use in bearings as is the case elsewhere in the NSS.

8.4.9.1 Non-NERVA Data

There is very little data available for this material which has been especially selected for use in bearings in the NSS. REIC Report No. 45⁽¹⁾ points out that its mechanical properties improve with radiation to 7×10^{17} n/cm² ($E > 1$ Mev) where both irradiation and testing is performed at room temperature. As is the case with similar stainless steel alloys, its mechanical properties will undoubtedly increase at NSS RA fluences and temperatures.

8.4.9.2 NERVA Data

There is no directly applicable experimental or estimated NERVA data. However, this material was used in bearing races and balls in NRX and XE Prime drum bearings without apparent difficulty.

8.4.9.3 Conclusions

Although there is little directly applicable data, SOL (unirradiated) design will undoubtedly apply. No additional tests are required.

8.4.10 Stainless Steel A-286

This material has important application in the Reflector Assembly for standoffs and springs as well as miscellaneous nuts, screws, bolts, etc.

8.4.10.1 Non-NERVA Data

An extensive compilation of non-NERVA data for A-286 will be presented in Chapter 4, Cluster Hardware, which represents its most important application in the NSS. In summary, A-286 has been shown to remain ductile following irradiation and slight changes in strength (increases and decreases) have been observed. As is the case for all springs, changes in stress relaxation are likely where A-286 is used in springs.

8.4.10.2 NERVA Data

An extensive Radiation Effects program for A-286 is currently underway and the results will be presented in Chapter 4. In addition, WANL has estimated A-286 properties for NSS PDR.⁽⁴⁶⁾ The slight changes in strength and decreases in ductility which characterize A-286 following irradiation are observed both in DRM estimates and data available from the literature. In the usage condition which characterizes this application, it is classified as ductile.⁽⁶⁾ A discussion of spring materials for NERVA applications concludes that this material is particularly suited for RA springs.⁽⁴⁷⁾

8.4.10.3 Conclusions

It can be concluded that SOL or unirradiated design conditions will apply and that the NSS RA will not undergo degradation due to the use of A-286. Where it is used in springs, stress relaxation will undoubtedly occur. The magnitude of this effect under NSS RA design conditions is unknown and REIR No. 19 has

been issued for that purpose. When the magnitude of the effect is known, compensation can be accomplished by pre-setting the springs.

8.4.11 Stainless Steel 321

This material is used exclusively for O-rings in the NSS RA. As discussed in Section 6.4.9, SS 321 is identical to SS 304 except that the former has titanium added to five times its carbon content. Both because of its application in the NSS RA and because of the data from related stainless steels, it can be concluded that SS 321 can be used for this application without degradation. No additional studies are required.

8.4.12 Graphite (ZTA and PO-3) and Grafoil

These materials will be analyzed in Chapter 5, Core Periphery, which will present a detailed review of the effects of radiation on graphites.

8.4.13 Component System Analysis

8.4.13.1 Non-NERVA Data

There is no available information.

8.4.13.2 NERVA Data

Although similar materials have been used in past NERVA reactors, the present NSS design is significantly different, especially in the application of structural loads to beryllium. Thus, there is no directly applicable NERVA data for this entire component.

8.4.13.3 Conclusions

On the basis of the assessment of individual materials and the environments specified in Tables 8-3 and 8-4, the materials planned for the NSS RA can be expected to function without significant degradation. Some additional testing is required, primarily to verify estimates.

TABLE 8-1
LIST OF MATERIALS (939J733)

18	71	LOCKING DEVICE			CRES	QQ-S-763 304 CONDA		
18	70	BOLT .375-24UNF-2A			CRES	AMS 5737 (A286)		
18	69	SCREW .250-28UNF-2A			CRES	AMS 5737 (A286)		
8	68	LOCKING DEVICE			CRES	QQ-S-763 304 CONDA		
1512	67	SCREW .250-28UNF-2A			CRES	AMS 5737 (A286)		
189	66	RETAINER SEGMENT			CRES	AMS 5737 (A286)		
18	65	LOCKWASHER	945C493H01-		CRES	QQ-S-766 304 COND A		
18	64	NUT, RETAINER	945C492H01-		AL	QQ-A-225/B (6061-T6)		
72	63	LOCKING DEVICE	946C902H01-		CRES	QQ-S-763 304 COND A		
72	62	BOLT .375-24UNF-2A			CRES	AMS 5737 (A286)		
108	61	PIN, CONTROL PLATE	964B177H01A		AL	QQ-A-225/B (6061-T6)		
593	60	RIVET, CONTROL PLATE			CU	PD13406-AC		
36	59	LOCKING DEVICE			CRES	QQ-S-763 304 COND A		
108	58	KEENEERT .375-24UNF-5A	KNL		CRES	AMS 5737 (A286)		
36	57	LOCKING DEVICE			CRES	QQ-S-763 304 CONDA		
36	56	BOLT .375 24 UNF-2A			CRES	AMS 5737 (A286)		
36	55	LOCKING DEVICE	947C061H01-		CRES	QQ-S-763 304 COND A		
36	54	BOLT .625-18 UNJF-5A	97BD965H01-		CRES	AMS 5735 (A286)		
72	53	LOCKING DEVICE	944C461H01-		CRES	QQ-S-763 304 CONDA		
72	52	BOLT .250-28UNF-2A			CRES	AMS 5737 (A286)		
18	51	LOCKWASHER	944C554H01-		CRES	QQ-S-766 304 COND A		
18	50	NUT, FORWARD BRG RETAINER			AL	QQ-A-225/B-T6 (6061)		
18	49	LOCKWASHER, BRG NUT	945C494H01-		CRES	QQ-S-766 304 CONDA		
18	48	NUT, AFT BRG RETAINER	945C495H01-		AL	QQ-A-225/B (6061-T6)		
36	47	BOLT .375-24UNF-2A			CRES	AMS 5737 (A286)		
18	46	FILLER, SEGMENT			AL	QQ-A-250/11-T4 (6061)		
18	45	HOUSING, SPRING			AL	QQ-A-225/B-T6 (6061)		
18	44	SPRING, DRIVE SHAFT			CRES	AMS 5737 (A286)		
18	43	RETAINER, INSULATION			AL	QQ-A-250/11-T4 (6061)		
18	42	INSULATOR, DRUM CYLINDER			GRAFOIL	PDS 30097-1		
1	41	SEAL, FORWARD IMPEDANCE			CRES	AMS 5525 (A286)		
756	40	O-RING .062 DIA X .012 WALL 1.300 OD			AL	WW-T-700/6-T6 (6061)		
756	39	O-RING .062 DIA X .012 WALL .875 OD			AL	WW-T-700/6-T6 (6061)		
1	38	O-RING .125 DIA X .012 WALL 42.50 OD			CRES	AISI 321 Ag PLATED		
1	37	SEAL, AFT STRUCTURE INNER			CRES	AMS 5525 (A286)		
1	36	SEAL, AFT STRUCTURE OUTER			CRES	AMS 5525 (A286)		
18	35	NUT, ADJUSTMENT			CRES	AMS 5737 (A286)		
756	34	PIN, LOCATING			AL	QQ-A-225/B-T6 (6061)		
1	33	SCREEN			CRES	QQ-W-423 (1304 CONDA		
189	32	SEGMENT, INSULATION			GRAFOIL	PDS 30097-1		
189	31	SEGMENT,			GRAPHITE	PDS 30019-1 (ZTA)		
189	30	SEGMENT,			GRAPHITE	PDS 30019-1 (ZTA)		
756	29	SPRING, LATERAL SUPPORT			CRES	AMS 5737 (A286)		
756	28	PLUNGER			GRAPHITE	PDS 30086-1 (P03)		
756	27	SPRING, SEGMENT RETAINER			CRES	AMS 5525 (A286)		
756	26	SLEEVE,			CRES	AMS 5737 (A286)		
756	25	CAP, CAPSULE			AL	QQ-A-250/11-T6 (6061)		
756	24	CAPSULE			AL	QQ-A-225/B-T6 (6061)		
1	23	RING, CLAMP			AL	QQ-A-250/11-T6 (6061)		
18	22	PAD, STANDOFF			AL	QQ-A-250/11-T6 (6061)		
18	21	STANDOFF			CRES	AMS 5737 (A286)		
36	20	BEARING	945C732H01		CRES	440 C		
18	19	COUPLING			CRES	QQ-S-763 304 CONDA		
18	18	SPRING, ANTI-BACKLASH			CRES	AMS 5525 (A286)		
18	17	STOP			CRES	QQ-S-763 304 CONDA		
54	16	VANE, CONTROL			CU B10	PDS 30127-1		
18	15	SUPPORT, AFT BEARING			AL	QQ-A-225/B-T6 (6061)		
18	14	HOUSING, FORWARD BRG			AL	QQ-A-225/B-T6 (6061)		
18	13	SPRING, DRUM CYLINDER			CRES	AMS 5737 (A286)		
18	12	SHAFT, AFT DRUM			AL	QQ-A-225/B-T6 (6061)		
18	11	SHAFT, FORWARD DRUM			AL	QQ-A-225/B-T6 (6061)		
18	10	CAN, DRUM			AL	WW-T-700/6-T6 (6061)		
18	9	CYLINDER, DRUM			BS	PDS 30018-1		
18	8	LINER, DRUM BORE			AL	WW-T-700/6-T6 (6061)		
1	7	CYLINDER, LOCATING			AL	QQ-A-367 6061-T652		
1	6	CAN, REFLECTOR BAFFLE			AL	QQ-A-250/11-T6 (6061)		
378	5	SEGMENT, OUTER REFLECTOR			BS	PDS 30018-1		
378	4	SEGMENT, INNER REFLECTOR			BS	PDS 30018-1		
1	3	RING, AFT SUPPORT			AL	QQ-A-250/11-T6 (6061)		
1	2	RING, FORWARD EXTENSION			AL	PDS 30258-1 (6061-T6)		
1	1	CYLINDER, STRUCTURAL			AL	PDS 30258-1 (6061-T6)		
QTY REQD	ITEM NO.	NOMENCLATURE OR DESCRIPTION		CODE IDENT NO.	PART OR IDENTIFYING NO.	MATERIAL	SPECIFICATION	ZONE

LIST OF MATERIALS

TABLE 8-2
REFLECTOR ASSEMBLY MATERIAL SUMMARY

Aluminum	-	6061-T6 (Bar & Forging)
Aluminum	-	6061-T61 (Barrel Forging)
Aluminum	-	6061-T652
Aluminum	-	6061-T4
Beryllium	-	(Segments)
Copper-Boron (18%)	-	(Vanes)
Copper	-	(Rivets)
Stainless Steel	-	A1S1 304 (Annealed)
Stainless Steel	-	A1S1 440C
Stainless Steel	-	A1S1 A-286
Stainless Steel	-	A1S1 321
Graphite	-	ZTA
Graphite	-	PO-3
Grafoil	-	(Insulation Segments)

TABLE 8-3

THE 3 σ MAXIMUM RADIATION ENVIRONMENT FOR MATERIALS IN REFLECTOR ECC

Item	MATERIAL	POSITION RADIUS (IN)	STATION (IN)	Dose Rate RAD- C/Hr	GAMMA ENVIRONMENT			NEUTRON ENVIRONMENT							
					1-Hour Dose	60-Ten Min Cycle Dose	FAST (E>1.0 MeV)			INTERMEDIATE (0.4eV<E<1.0 MeV)			THERMAL (E<0.4eV) **		
							Flux	1-Hour Fluence	60-Ten Min Cycle Fluence	Flux	1-Hour Fluence	60-Ten Min Cycle Fluence	Flux	1-Hour Fluence	60-Ten Min Cycle Fluence
	Reference Drawing 935J733				RAD-C	RAD-C	N/CM Sec	N/CM ²	N/CM ²	N/CM ² Sec	N/CM ²	N/CM ²	N/CM ² Sec	N/CM ²	N/CM ²
Al	QQ-A-250/11 (6061-T6)	C G	22.215 26.0	6.0(9) 7.0(9)	6.0(9) 7.0(9)	6.0(10) 7.0(10)	1.1(15) 9.9(14)	4.0(18) 3.5(18)	4.0(19) 3.5(19)	1.2(16) 1.1(16)	4.4(19) 3.8(19)	4.4(20) 3.8(20)	1.3(15) 1.2(15)	4.5(18) 4.2(18)	4.5(19) 4.2(19)
Al	PDS-30258 (6061-T6)	C G	21.115 26.0	8.1(9) 9.5(9)	8.1(9) 9.5(9)	8.1(10) 9.5(10)	1.5(15) 1.3(15)	5.0(18) 4.9(18)	5.0(19) 4.9(19)	1.4(16) 1.2(16)	5.0(20) 4.3(19)	5.0(20) 4.3(20)	1.2(15) 1.1(15)	4.2(18) 4.0(18)	4.2(19) 4.0(19)
Al	QQ-A-367 (6061-T652)	C G	21.115 55.2	1.1(9) 1.3(9)	1.1(9) 1.3(9)	1.1(10) 1.3(10)	2.1(14) 1.8(14)	7.6(17) 6.6(17)	7.6(18) 6.6(18)	1.1(15) 9.6(14)	4.1(18) 3.4(18)	4.1(19) 3.4(19)	8.1(13) 7.3(13)	2.9(17) 2.6(17)	2.9(18) 2.6(18)
Al	QQ-A-250/11 (6061-T4)	C G	22.215 - 3.5	***1.6(9) 1.8(9)	1.6(9) 1.8(9)	1.6(10) 1.8(10)	2.0(14) 1.7(14)	7.3(17) 6.4(17)	7.3(18) 6.4(18)	1.7(15) 1.4(15)	6.1(18) 5.2(18)	6.1(19) 5.2(19)	3.3(14) 2.9(14)	1.2(18) 1.0(16)	1.2(19) 1.0(19)
Be	Be	C G	22.215 26.0		SAME AS ABOVE R = 22.215; Z = 26.0										
CuB	PDS 30127-1 (CuB ¹⁰)	C G	22.215 26.0		SAME AS ABOVE R = 22.215; Z = 26.0										
CRES	QQ-763 304 - Cond. A (SS - 304)	C G	22.215 0.0	2.2(9) 2.5(9)	2.2(9) 2.5(9)	2.2(10) 2.5(10)	3.1(14) 2.8(14)	1.1(18) 1.0(18)	1.1(19) 1.0(19)	3.0(15) 2.6(15)	1.1(19) 9.3(18)	1.1(20) 9.3(19)	4.0(14) 3.6(14)	1.4(18) 1.3(18)	1.4(19) 1.3(19)
CRES	SS - 440C	C G	22.215 0.0		SAME AS ABOVE R = 22.215; Z = 0.0										
CRES	AMS-5737 (A-286)	C G	21.115 26.0		SAME AS ABOVE R = 22.215; Z = 26.0										
	SS - 321	C	22.215 0.0		SAME AS ABOVE R = 22.215; Z = 0.0										
Maximum Thermal for Reflector, ECC		C G											1.5(15) 1.3(15)	5.3(18) 4.5(18)	5.3(19) 4.5(19)

*C = Composite Core G = Graphite Core

**Thermal Flux is at R, Z Coordinates given and is not necessarily the Maximum; the Maximum Thermal Flux for the Reflector ECC is listed separately.

***Maximum Gamma Dose not at this Location; Maximum for Material $\approx 2.2 \times 10^4$ RAD-C/Hr.

Numbers in () Refer to Powers of Ten; i.e., 6.0 (9) = 6.0×10^9 .

TABLE 8-4

**SUMMARY OF REFLECTOR ASSEMBLY COMPONENT MATERIAL TEMPERATURES
AND OTHER KEY PERFORMANCE PARAMETERS (GRAPHITE CORE)**

COMPONENT	LOCATION	PARAMETER	NORMAL MODE OPERATION						COOLDOWN OPERATION RANGE	SPACE SOAK RANGE	
			RATED PERFORMANCE, EOL 99/95 TOLERANCE LIMIT			THROTTLED HOLD, EOL 9/95 TOLERANCE LIMIT				Solar Flux ⁽³⁾ of 300 Btu/Hr/Ft ²	Solar Flux ⁽⁴⁾ of 3000 Btu/Hr/Ft ²
			NOMINAL	UPPER	LOWER	NOMINAL	UPPER	LOWER			
Reflector Assembly	C.S. 0 to 52	Heat Addition to Coolant - MW	21.46	23.88	19.04	14.25	15.77	12.73			
	C.S. 0 to 52	Axial Δp - psi	52.0	55.9	48.1	31.0	34.0	28.0			
	Reflector Exit (including Pressure Vessel Annulus)	Mixed Mean Coolant Temp. deg R	268.	283.	253.	312.	332.5	291.5			
Structure	C.S. 52	Radial Δp - psi	670	689.	651.	386.	398.	374.	(360-710) ⁽¹⁾ 800 ⁽²⁾	350-550	550-900
	C.S. 25	$\bar{T}$ - deg. R	350.	389.	311.	380.	423.	337.			
	C.S. 25	T_{max} - deg. R	485.	519.	451.	493.	523.	463.			
Inner Segments	C.S. 26	$\bar{T}$ - deg. R	276.	292.	260.	—	—	—	(290-760) ⁽¹⁾ 800 max ⁽²⁾ with	350-550	550-900
	C.S. 26	$\bar{T}$ - T_{min} deg. R	31.	41.	21.	—	—	—			
	C.S. 26	T_{max} - $\bar{T}$	65.	85.	45.	—	—	—			
Outer Segments	C.S. 26	$\bar{T}$	255.	270.	240.	—	—	—	(290-760) ⁽¹⁾ 800 max ⁽²⁾ with	350-550	550-900
	C.S. 26	$\bar{T}$ - T_{min}	15.	20.	10.	—	—	—			
	C.S. 26	T_{max} - $\bar{T}$	12.	16.	8.	—	—	—			
Reflector Baffle Can	C.S. 0	T_{max}	269.	284.	254.	295.	313.	277	(290-760) ⁽¹⁾ 800 max ⁽²⁾ with	350-550	550-900
	Control Drum Cylinders	$\bar{T}$	309.	332.	286.	335.	359.	311.			
Control Drum Cylinders	C.S. 26	$\bar{T}$	309.	332.	286.	335.	359.	311.	(290-760) ⁽¹⁾ 800 max ⁽²⁾ with	350-550	550-900
	C.S. 26	$\bar{T}$ - T_{min}	43.	58.	28.	32.	43.	21.			
	C.S. 26	T_{max} - $\bar{T}$	37.	50.	24.	30.	41.	19.			
Drum Cans	C.S. 26	T_{max}	300.	322.	278.	334.	358.	310.	(290-760) ⁽¹⁾ 800 max ⁽²⁾ with	350-550	550-900
Control Vanes	C.S. 26	T_{max}	470.	504.	436.	495.	531.	459.	(290-760) ⁽¹⁾ 800 max ⁽²⁾ with	350-550	550-900
Control Drum Bore Liners	C.S. 0	T_{max}	291.	311.	271.	311.	334.	288.	(290-760) ⁽¹⁾ 800 max ⁽²⁾ with	350-550	550-900
Aft End Hardware	C.S. 52	T Coolant	198.	210.	186.	229.	240.	218.			
Forward Control Drum Hardware	C.S. 0	T Coolant	272.	290.	254.	315.	339.	291.			

8-22

8-22

TABLE 8-4 (CONT' D)

- (1) This is the temperature range within 3.65 hours after a ten minute burn and throttle hold, as reported in WANL-TME-2678, "Systems Analysis Summary Report, Nuclear Subsystem," Section 5.2.6.3.
- (2) This is the maximum allowable temperature between cooling pulses.
- (3) This is the minimum flux shown in WANL-TME-2678, "Systems Analysis Summary Report, Nuclear Subsystem," Section 5.2.6.3. Solar flux in earth orbit is 444 Btu/Hr-ft^2 as reported in WANL-TME-2678.
- (4) This is the extreme solar flux, in the vicinity of Mercury, as reported in WANL-TME-2678.

TABLE 8-5
GTR-18 BERYLLIUM DATA
IRRADIATED AT 140°R TO APPROXIMATELY 3×10^{17} n/cm² (27)

Irradiated (I) Control (C)	Type of Test	Test Temperature	Anneal Temp/Time	No. of Specimens	U.T.S.			0.2% Y.S.			Plastic Strain			No. of Specimens	N.T.S.		
					Average	Max.	Min.	Average	Max.	Min.	Average	Max.	Min.		Average	Max.	Min.
		°R	(°R)/(min.)		(KSI)	(KSI)	(KSI)	(KSI)	(KSI)	(KSI)	(%)	(%)	(%)		(KSI)	(KSI)	(KSI)
Pressing No. 4055																	
C	Tensile	140	---	5	40.4	43.4	38.4	37.3	39.4	35.1	0.31	0.42	0.20	4	37.0	43.7	30.4
I	Tensile	140	---	7	39.0	42.9	30.9	*	---	---	0.00	---	---	4	31.8	34.1	29.0
C	Tensile	240	---	4	43.1	45.1	39.8	34.5	35.6	33.8	0.80	1.09	0.39	4	37.7	39.5	36.1
I	Tensile	240	---	5	41.6	46.2	31.5	*	---	---	0.00	---	---	5	43.7	49.1	38.1
C	Tensile	340	---	4	44.6	46.4	43.3	33.2	33.6	32.5	1.29	1.35	1.18	4	37.0	34.0	35.2
I	Tensile	340	---	4	49.8	53.0	48.3	51.8 (One Specimen)	---	---	0.17	0.27	0.10	3	47.5	50.4	44.8
C	Tensile	440	---	3	43.3	45.4	42.1	32.1	32.5	31.7	1.21	1.31	1.09	4	40.3	41.2	38.9
I	Tensile	440	---	4	48.0	50.2	46.1	37.9	39.1	36.3	1.11	1.23	1.01	4	44.5	47.2	42.8
C	Tensile	540	---	4	40.8	42.3	38.6	31.8	32.9	30.7	1.01	1.08	0.91	4	38.6	49.0	27.1
I	Tensile	540	---	4	41.0	43.7	39.1	31.9	32.8	31.2	1.15	1.41	1.05	3	37.3	44.1	30.0
I	Tensile	140	240/60	4	40.7	41.3	39.7	*	---	---	0.00	---	---	---	---	---	---
I	Tensile	140	340/5	4	42.3	46.3	39.1	*	---	---	0.00	---	---	---	---	---	---
I	Tensile	140	340/60	4	40.8	42.1	38.7	*	---	---	0.00	---	---	---	---	---	---
I	Tensile	140	340/120	4	42.9	43.9	41.5	*	---	---	0.00	---	---	---	---	---	---
I	Tensile	140	440/60	4	46.9	48.1	45.9	*	---	---	0.00	---	---	---	---	---	---
I	Tensile	140	540/5	5	45.7	50.4	36.5	*	---	---	0.05	0.16	0.00	---	---	---	---
I	Tensile	140	540/60	4	46.3	48.1	45.0	38.5	39.4	37.9	0.60	0.69	0.53	4	39.1	42.0	35.6
I	Tensile	140	540/120	4	44.8	46.4	42.8	39.1	41.1	38.1	0.46	0.61	0.38	---	---	---	---
Pressing No. 2099																	
C	Tensile	140	---	4	44.5	45.3	43.3	43.0	43.8	41.7	0.26	0.29	0.23	5	37.4	40.9	33.0
I	Tensile	140	---	4	46.8	48.1	45.4	---	---	---	---	---	---	5	38.7	46.6	25.2
C	Tensile	540	---	4	53.4	55.7	50.7	37.0	38.9	35.9	2.69	3.68	2.27	4	48.2	50.3	46.7
I	Tensile	540	---	4	49.4	51.7	48.6	34.8	35.3	34.3	2.04	2.67	1.75	4	48.1	50.1	46.0
Flexure Strength																	
C	Flexure	140	---	6	66.1	68.3	64.5	---	---	---	---	---	---	---	---	---	---
I	Flexure	140	---	4	55.0	57.5	53.0	---	---	---	---	---	---	---	---	---	---
I	Flexure	140	540/60	4	62.4	66.6	58.0	---	---	---	---	---	---	---	---	---	---
C	Flexure	540	---	6	67.5	69.2	65.9	---	---	---	---	---	---	---	---	---	---
I	Flexure	540	---	4	65.2	68.9	64.6	---	---	---	---	---	---	---	---	---	---

* Br. = broken at 0.2% plastic strain.

TABLE 8-6
GTR-18 BERYLLIUM DATA (27)
Pressing No. 4055

Designated (I) Control (C)	Type of Test	Test Temperature	Anneal Temperature	Anneal Time	No. of Specimens	Ultimate Strength			0.2% Yield Strength			Plastic Strain			No. of Specimens	Notched Ultimate Strength					
						Average	Max.	Min.	Average	Max.	Min.	Average	Max.	Min.		Average	Max.	Min.			
		(°R)	(°R)	(Min)		(KSI)	(KSI)	(KSI)	(KSI)	(KSI)	(KSI)	(%)	(%)	(%)							
C	Tensile	140	---	---	6	45.0	46.3	43.4	36.6	37.6	36.0	0.61	0.83	0.43	4	33.0	34.7	32.0			
I	Tensile	140	---	---	6	39.0	44.7	35.4	*	---	---	0.00	---	---	4	25.7	31.1	21.0			
C	Tensile	240	---	---	4	45.0	46.3	44.2	34.7	35.2	33.7	1.07	1.34	0.85	4	32.9	36.5	29.3			
I	Tensile	240	---	---	5	41.7	44.1	38.0	*	---	---	0.00	---	---	5	33.7	40.7	23.5			
C	Tensile	340	---	---	4	44.0	44.3	43.4	31.7	32.0	31.6	1.12	1.59	1.76	5	32.8	37.1	25.2			
I	Tensile	340	---	---	4	43.4	50.3	43.1	*	---	---	0.01	0.03	0.00	4	30.4	37.2	22.5			
C	Tensile	440	---	---	4	42.7	43.5	41.3	31.2	32.6	29.7	1.30	1.49	1.25	4	30.6	40.7	27.2			
I	Tensile	440	---	---	4	41.4	46.5	45.3	37.2	37.8	36.2	1.07	1.14	1.04	4	33.2	35	28.4			
C	Tensile	540	---	---	4	41.3	41.2	42.5	30.6	31.0	30.1	1.39	1.55	1.27	5	30.3	35.7	25.7			
I	Tensile	540	---	---	4	42.4	41.5	43.5	32.7	33.4	32.2	1.19	1.39	1.00	4	33.1	40.4	25.3			
I	Tensile	140	240	60	4	37.9	39.9	36.5	*	---	---	0.00	---	---	4	27.0	29.9	24.0			
I	Tensile	140	340	5	5	39.5	43.5	36.7	*	---	---	0.00	---	---	---	---	---	---			
I	Tensile	140	340	60	4	39.9	41.5	38.7	*	---	---	0.00	---	---	4	27.2	27.8	26.3			
I	Tensile	140	340	1020	4	41.3	41.3	41.0	*	---	---	0.00	---	---	---	---	---	---			
I	Tensile	140	440	60	5	43.0	50.6	45.2	*	---	---	0.00	---	---	4	32.6	41.0	24.6			
I	Tensile	140	540	5	6	48.6	52.7	45.1	48.1	51.2	44.3	0.21	0.30	0.11	---	---	---	---			
I	Tensile	140	540	60	4	46.6	49.4	44.3	40.2	41.2	39.2	0.48	0.63	0.35	4	33.0	35.0	31.1			
I	Tensile	140	540	1020	4	45.6	47.0	44.0	39.4	40.0	37.6	0.52	0.60	0.39	---	---	---	---			
C	Compression	140	---	---	4	324.0	One Specimen			40.2	42.0	39.0	10.2	One Specimen			---	---			
I	Compression	140	---	---	4	310.0	One Specimen			110.8	113.8	104.0	12.7	One Specimen			---	---			
C	Compression	540	---	---	4	255.0	One Specimen			33.0	32.4	32.3	37.7	One Specimen			---	---			
I	Compression	540	---	---	4	218.0	One Specimen			34.0	34.5	33.6	29.1	One Specimen			---	---			
I	Compression	140	540	60	4	---	---	---	42.9	43.8	42.3	---	---	---	---	---	---	---			
						Flexure Strength			Hardness (R _B)												
C	Flexure	140	---	---	8	68.4	73.1	61.5	102												
I	Flexure	140	---	---	7	54.0	57.0	50.2	103												
C	Flexure	540	---	---	8	67.5	73.0	56.7	83												
I	Flexure	540	---	---	6	69.6	71.9	64.7	83												
I	Flexure	140	540	60	7	64.6	72.4	56.4	97												

Broke before 0.2% offset strain.

TABLE 8-7 (29)
PLANE STRAIN FRACTURE TOUGHNESS (K_{IC}) OF
IRRADIATED BERYLLIUM

Irrad. (I) or* Control (C)	Orientation	Test Temp.	Anneal Temp./Time	K_{IC} KSI $\sqrt{in}$
C	Longitudinal	140°R	-----	13.6
C	Transverse	140°R	-----	16.2
I	Longitudinal	140°R	-----	8.8, 10.1
I	Transverse	140°R	-----	9.7, 8.2
I	Longitudinal	540°R	-----	16.5
I	Transverse	540°R	-----	19.6
I	Longitudinal	140°R	340°R/60 min.	9.8
I	Transverse	140°R	340°R/60 min.	9.2
I	Longitudinal	140°R	540°R/60 min.	14.4
I	Transverse	140°R	540°R/60 min.	14.1

NOTE: Each value represents the results of one test.

* Irradiated to approximately 3.0×10^{17} n/cm² at 140°R.

TABLE 8-8
TENSILE PROPERTIES OF IRRADIATED AND UNIRRADIATED COPPER⁽³⁵⁾

EXPOSURE n/cm^2	TEST TEMPERATURE C	YIELD STRENGTH 10^3 psi	ULTIMATE TENSILE STRENGTH, 10^3 psi
0	-195	8.0	49.0
9.5×10^{17}	-195	10.0	51.0
5.0×10^{18}	-195	16.0	53.0
8.8×10^{18}	-195	--	51.0
4.0×10^{19}	-195	34.0	54.0
6.8×10^{19}	-195	39.0	58.0
7.5×10^{19}	-195	42.0	60.0
1.07×10^{20}	-195	47.0	62.5
0	20	7.0	32.0
9.5×10^{17}	20	8.0	32.0
5.0×10^{18}	20	12.0	32.0
8.8×10^{18}	20	17.0	32.0
4.0×10^{19}	20	23.0	34.0
6.8×10^{19}	20	26.0	37.0
7.5×10^{19}	20	28.0	38.0
1.07×10^{19}	20	31.0	39.0
0	200	6.0	22.0
9.5×10^{17}	200	8.0	23.0
5.0×10^{18}	200	9.0	23.0
8.8×10^{18}	200	--	--
4.0×10^{18}	200	18.0	27.0
6.0×10^{19}	200	19.0	--
7.5×10^{19}	200	20.0	--
1.07×10^{20}	200	23.0	30.0

TABLE 8-9 (43)
RESULTS OF IRRADIATION AT 125°F ON MECHANICAL PROPERTIES OF Cu24% B¹⁰

TENSILE TESTING TEMP. °R	NOMINAL THERMAL NEUTRON FLUENCE	0.2% OFFSET YIELD STRENGTH (psi)	ULTIMATE TENSILE STRENGTH (psi)	FRACTURE STRESS (psi)	TOTAL CHART ELONGATION %
540	--	11,980	16,150	15,840	2.0
540	1 x 10 ¹⁷	12,710	15,040	14,810	1.6
540	1 x 10 ¹⁸	12,980	13,360	(1)	(1)
540	1 x 10 ¹⁹	13,900	14,360	14,090	0.6
940	--	9,460	11,440	11,280	1.3
940	1 x 10 ¹⁷	9,420	11,480	11,280	0.8
940	1 x 10 ¹⁸	10,310	10,820	10,700	0.5
940	1 x 10 ¹⁹	11,910	13,670	12,890	0.9
1340	--	3,540	4,010	3,970	0.5
1340	1 x 10 ¹⁷	4,400	5,100	4,980	1.1
1340	1 x 10 ¹⁸	3,970	4,280	4,200	0.7
1340	1 x 10 ¹⁹	4,200	4,400	4,200	0.7
1740	--	1,370	1,560	(1)	(1)
1740	1 x 10 ¹⁷	1,150	1,260	(1)	(1)
1740	1 x 10 ¹⁸	1,170	1,250	(1)	(1)
1740	1 x 10 ¹⁹	1,090	1,120	(1)	(1)

(1) Because of the shape of the stress-strain curve the fracture load and total elongation could not be determined.

Figure 8-1 Swelling of Beryllium Irradiated to a Fast Fluence of 1×10^{21} n/cm² at a Temperature of 70°C as a Function of Annealing Time and Temperature. (from Ref. 20)

Figure 8-2 Swelling Threshold Temperature of Beryllium as a Function of Gas Content and Fluence. (from Ref. 21)

Figure 8-3 Calculated Change in Electrical and Thermal Conductivity of Impure Beryllium Upon Irradiation to a Fast Fluence of 10^{19} n/cm². (From Ref. 22)

Figure 8-4 Changes in Yield Strength, Ductility, and Hardness of Beryllium Irradiated at Fast Fluences and a Temperature Range of 75 to 100°C. (from Ref. 13)
Tested at Room Temperature

Figure 8-5 Thermal Conductivity of Unirradiated Beryllium S_{7200} in Both Perpendicular and Parallel Directions (32)

(32)

Figure 8- 6 Annealing Curves for Irradiated Beryllium

LIST OF REFERENCES
CHAPTER 8 - REFLECTOR ASSEMBLY

1. Kangilawski, M., "The Effects of Neutron Irradiation on Structural Materials," REIC Report No. 45 (June, 1967)
2. Schwanbeck, C. A., "Effect of Nuclear Radiation on Materials at Cryogenic Temperatures, Final Report," NASA-CR-54881 (N66-24695) (January, 1965)
3. Lombardo, J. J., Dixon, C. E., and Begley, J. A., "Cryogenic Radiation Effects on NERVA Structural Materials," Paper presented at the 69th annual meeting of the ASTM in Atlantic City, New Jersey, June 27 - July 1, 1966
4. Carlson, E., and French, W. E., "NERVA Radiation Effects Summary Report," S-110-SR01-W181 (May, 1970)
5. Schollenberger, R. S., DRM 51294, Westinghouse Astronuclear Laboratory, (August, 1970)
6. Holden, A. N., DRM 51281, Westinghouse Astronuclear Laboratory (June, 1970)
7. Schollenberger, R. S., DRM 51276, Westinghouse Astronuclear Laboratory (August, 1970)
8. Schollenberger, R. S., DRM 53161, Westinghouse Astronuclear Laboratory (August, 1970)
9. Dykes, J. W., and Ford, J. D., "A Preliminary Report of Beryllium Damage Observed in the MTR Reflector," USAEC Report IDO-16899, Idaho Operations Office, (June, 1963)
10. Hurt, S. S., III, Oak Ridge National Laboratory, "Experience with Neutron Damage to Beryllium and Aluminum Lattice Components at the ORR," Paper presented at the ANS Conference on Reactor Experience in Atlantic City, New Jersey, July 23-26, 1967
11. Ellis, C. E., and Perryman, E. C. W., "Effects of Neutron-Induced Gas Formation on Beryllium," Journal of Nuclear Materials, 1: 73-84 (1959)
12. Hickman, B. S., and Stevens, C. T., "The Effect of Neutron Irradiation of Beryllium Metal," Australian Atomic Energy Commission Research Establishment Report AAEC-E-109, Lucas Heights, June 1963

LIST OF REFERENCES (CONT'D)

13. Hickman, B.S., and Chute, J. H., "The Behavior of Helium in Irradiated Beryllium," Journal of the Australian Institute of Metals, 8(3): 289-307 (August, 1963)
14. Weir, J. R., "The Effect of High Temperature Reactor Irradiation on Some Physical and Mechanical Properties of Beryllium," pp 395-409 in the The Metallurgy of Beryllium, Proceedings of International Conference on Beryllium in London, England, October 16-18, 1961, Chapman and Hall, London, 1963
15. J. B. Rich, G. B. Redding, and R. S. Barnes, "The Effects of Heating Irradiated Beryllium," Journal of Nuclear Materials, 1: 96-105 (1959)
16. J. B. Rich, G. P. Walters, and R. S. Barnes, "The Mechanical Properties of Some Highly Irradiated Beryllium," Journal of Nuclear Materials, 4(3): 287-294 (1961)
17. R. S. Barnes, "The Behavior of Neutron Irradiated Beryllium," pp. 372-380 in The Metallurgy of Beryllium, Proceedings of International Conference on Beryllium in London, England, October 16-18, 1961. Chapman and Hall, London, 1963
18. "Materials Testing Reactor, Engineering Test Reactor Technical Branches' Quarterly Report, January 1 -March 31, 1962," USAEC Report IDO-16781, Idaho Operations Office, 1962
19. "Materials Testing Reactor, Engineering Test Reactor Technical Branches' Quarterly Report, April 1 - June 30, 1962," USAEC Report IDO-16805, Idaho Operations Office, 1962
20. "Materials Testing Reactor, Engineering Test Reactor Technical Branches' Quarterly Report, July 1 - September 30, 1962," USAEC Report IDO-16827, Idaho Operations Office, January 1963
21. R. Tromp, "Swelling Threshold Temperature for Irradiated Beryllium," pp. 12.14-12.15 in Quarterly Progress Report: Irradiation Effects on Reactor Structural Materials, May, June, July 1965, USAEC Report BNWL-128, Battelle-Northwest Laboratory, August 1965

LIST OF REFERENCES (CONT'D)

22. G. J. Dienes and A. C. Damask, "An Estimate of the Effects of Radiation on the Thermal Conductivity of Beryllium," Journal of Nuclear Materials, 3: 16-21 (1961)
23. G. T. Stevens and B. S. Hickman, "Effects of Irradiation on the Mechanical Properties of Beryllium Metal," Australian Atomic Energy Commission Research Establishment Report, AAEC-E-133, Lucas Heights, January 1965
24. G. P. Walters, "Effect of Neutron Irradiation on the Mechanical Properties of Hot Pressed and Extruded Beryllium," Journal of the Less - Common Metals, 11: 72-78 (1966)
25. J. B. Rich and G. P. Walters, "The Mechanical Properties of Beryllium Irradiated at 350 and 600°C," pp. 362-371 in The Metallurgy of Beryllium, Proceedings of International Conference on Beryllium in London, England, October 16-18, 1961, Chapman and Hall, London, 1963
26. B. S. Hickman, "The Effects of Neutron Irradiation on Beryllium Metal," pp. 410-417 in The Metallurgy of Beryllium, Proceedings of International Conference on Beryllium in London, England, October 16-18, 1961, Chapman and Hall, London, 1963
27. G. P. Walters, "Effects of Neutron Irradiation on the Mechanical Properties of Hot Pressed and Extruded Beryllium," pp. 138-159 in Conference on the Physical Metallurgy of Beryllium, Gatlinburg, Tennessee, April 30-May 1, 1963, USAEC Report CONF-170, UDSEC Division of Technical Information
28. C. F. Ellis and J. R. Weir, "The Effect of Neutron Irradiation on the Plastic Deformation and Fracture of Beryllium," Journal of Nuclear Materials, 13(1): 45-46 (1964)
29. R. P. Shogan, "Irradiation Effects on the Mechanical Properties of S-200-Grade Beryllium and Lockalloy at Cryogenic Temperatures", WANL-TME-1655, September 1967
30. J. W. Allen, Personal Communication (November, 1970)
31. Palmer, E. E., Bell, J. R., Crabtree, R. D., Wattier, J. B., Springer, H. L., and Thomas, E. A., "Thermal Conductivity and Electrical Resistivity of Selected NERVA Materials," Report FJK 351-3, General Dynamics, Fort Worth Division (Nov., 1969)

LIST OF REFERENCES (CONT'D)

32. Williams, J., Oak Ridge National Laboratory, Solid State Division, Personal Communication (August, 1970)
33. Williams, J., Oak Ridge National Laboratory, Solid State Division, Personal Communication (November, 1970)
34. Schollenberger, R.S., DRM 51205, Westinghouse Astronuclear Laboratory (August, 1970)
35. Makin, M. J., and Minter, F. J., "Irradiation Hardening in Copper and Nickel," *Acta Met*, 8 (October, 1960)
36. Greenfield and Wilsdorf, "Observations Concerning the Radiation Hardening in Cu and Ni, *Naturw*" (1960)
37. Thompson, D. O., and Blewitt, T. H., et al, "Low Temperature Measurements of Young's Modulus and Internal Friction of Copper During Irradiation," *J. Appl. Phys.*, 21 (6) (1957)
38. Thompson, D. O. and Holmes, D. K., "The Effect of Gamma Irradiation on the Young's Modulus of Copper," *Phys. Chem*, 1 (4) (1957)
39. Josin, A., and Bienvenue, L. L., "Effect of Electron Irradiation and Subsequent Thermal Treatment on Young's Modulus of Copper," *J. Appl. Phys.*, 31 (2) (February, 1960)
40. Barnes R. S. and Hancock, N. H., "The Effect of Neutron Irradiation Upon the Internal Friction of Copper Single Crystals at Liquid Nitrogen Temperatures," AERE-M/R-2436A (November, 1957)
41. Coltman, R. R., Blewitt, T. H., Klabunde, C. F., and Redman, J. K., "Low Temperature Radiation Damage in Metals," *Bulb Phys. Soc.* 4 (3), (March, 1959)
42. Corbett, J. W. and Walker, R. M., "Energy Dependence of Recovery in Irradiated Copper," *Phys. Rev*, 115 (1) 67 (July, 1959)
43. Winterich, D., Plum Brook Reactor Facility, NASA, Sandusky, Ohio, Private Communication (September, 1970)

LIST OF REFERENCES (CONT 'D)

44. Schollenberger, R. S., DRM 51214, Westinghouse Astronuclear Laboratory
(August, 1970)
45. Schollenberger, R. S., DRM 51232, Westinghouse Astronuclear Laboratory
(August 1970)
46. Schollenberger, R. S., DRM 51250, Westinghouse Astronuclear Laboratory
(August, 1970)
47. McAndrew, J. B., DRM 53174, Westinghouse Astronuclear Laboratory
(October, 1970)

RADIATION EFFECTS INFORMATION REQUESTS (REIRS)

(Chapter 8 - Reflector Assembly)

No.

12	Aluminum 6061-T61 Barrel Forging - Tensile and Fracture Properties
13	Aluminum 6061-T61 Barrel Forging - Low Cycle Fatigue
14	Beryllium - Dynamic Modulus
15	Beryllium - Cyclic Crack Growth
16	Beryllium - Thermal Conductivity
17	Copper Boron - Fracture Strength
18	Copper Boron - Thermal Conductivity
19	A - 286 - Spring Stress Relaxation

IDENTIFICATION NO. 12

RADIATION EFFECTS INFORMATION REQUEST

DATE November 10, 1970

SUBJECT Aluminum 6061-T61 Barrel Forging FUNCTIONAL CATEGORY Test & Evaluation

REF. ITEM SUPPORT EC-677559 REF. DATA ITEMS S-100

INFORMATION DESIRED Effects of irradiation on strength and ductility as well as fracture toughness (if the
(PURPOSE OF STUDY OR EXPERIMENT)
material is brittle in any direction

DATE REQUIRED CY 71

RELATION OF ITEM TO ENGINE Forging which encases beryllium reflector segments
(LOCATION)
PER DRAWINGS, DATA ITEMS Dwg. 939J733

ENVIRONMENT (4): NEUTRON FLUX _____ NEUTRON FLUENCE $5.6 \times 10^{18-19}$ n/cm² (E > 1 Mev)
(Ref. data items) GAMMA DOSE RATE _____ GAMMA DOSE _____
TEMPERATURE 140°R - 750°R NUCLEAR HEATING RATE _____
PRESSURE _____ OTHER _____
FLUID _____
CYCLE REQUIREMENTS TBD

RELIABILITY ESTIMATE Screening* REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL Screening APPROVED BY _____ DATE _____

APPROVED BY _____ DATE _____

SPECIAL CONSIDERATIONS _____ APPROVED BY _____ DATE _____

PERTINENT REFERENCES _____

* If the material is brittle under any use conditions, K_{IC} data will be required to 99% reliability at 95% CL

IDENTIFICATION NO. 13

RADIATION EFFECTS INFORMATION REQUEST

DATE November 10, 1970

SUBJECT Aluminum 6061-T61 Barrel Forging FUNCTIONAL CATEGORY Test & Evaluation

REF. ITEM SUPPORT EC-677559 REF. DATA ITEMS S-110

INFORMATION DESIRED Effects of radiation on low cycle fatigue properties of this material
(PURPOSE OF STUDY OR EXPERIMENT)

DATE REQUIRED CY 71 - 72

RELATION OF ITEM TO ENGINE Forging which is primary support structure of reflector assembly
(LOCATION)

PER DRAWINGS, DATA ITEMS Dwg. 939J733

ENVIRONMENT (4): NEUTRON FLUX _____
(Ref. data items)

NEUTRON FLUENCE _____

GAMMA DOSE RATE _____

GAMMA DOSE 5.6×10^{18} & 5.6×10^{19} n/cm² (E > 1 Mev)

TEMPERATURE 140°R - 750°R

NUCLEAR HEATING RATE _____

PRESSURE _____

OTHER _____

FLUID _____

CYCLE REQUIREMENTS _____

RELIABILITY ESTIMATE 99% REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL 95% APPROVED BY _____ DATE _____

APPROVED BY _____ DATE _____

SPECIAL CONSIDERATIONS _____ APPROVED BY _____ DATE _____

PERTINENT REFERENCES _____

IDENTIFICATION NO. 14

RADIATION EFFECTS INFORMATION REQUEST

DATE November 10, 1970

SUBJECT Beryllium FUNCTIONAL CATEGORY Test & Evaluation

REF. ITEM SUPPORT EC-677559 REF. DATA ITEMS S-110

INFORMATION DESIRED Effects of radiation on Dynamic Modulus of beryllium
(PURPOSE OF STUDY OR EXPERIMENT)

DATE REQUIRED CY 71-72

RELATION OF ITEM TO ENGINE Reflector segments outside core periphery and within pressure vessel
(LOCATION)

PER DRAWINGS, DATA ITEMS Dwg. 939J733

ENVIRONMENT (4): NEUTRON FLUX _____
(Ref. data items)

NEUTRON FLUENCE 4×10^{17} & 4×10^{19} n/cm² (E > 1 Mev)

GAMMA DOSE RATE _____

GAMMA DOSE _____

TEMPERATURE 140°R - 750°R

NUCLEAR HEATING RATE _____

PRESSURE _____

OTHER _____

FLUID _____

CYCLE REQUIREMENTS _____

RELIABILITY ESTIMATE TBD REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL TBD APPROVED BY _____ DATE _____

APPROVED BY _____ DATE _____

SPECIAL CONSIDERATIONS _____ APPROVED BY _____ DATE _____

PERTINENT REFERENCES _____

8-41

IDENTIFICATION NO. 15

RADIATION EFFECTS INFORMATION REQUEST

DATE November 10, 1970

SUBJECT Beryllium FUNCTIONAL CATEGORY Test & Evaluation

REF. ITEM SUPPORT EC-677559 REF. DATA ITEMS S-110

INFORMATION DESIRED Effects of irradiation on cyclic crack growth
(PURPOSE OF STUDY OR EXPERIMENT)

DATE REQUIRED CY 71-72

RELATION OF ITEM TO ENGINE Reflector segments outside core periphery and within pressure vessel
(LOCATION)

PER DRAWINGS, DATA ITEMS Dwg. 939J733

ENVIRONMENT (4): NEUTRON FLUX _____ NEUTRON FLUENCE 4×10^{18} & 4×10^{19} n/cm² (E > 1 Mev)
(Ref. data items)

GAMMA DOSE RATE _____ GAMMA DOSE _____

TEMPERATURE 140°R - 750°R NUCLEAR HEATING RATE _____

PRESSURE _____ OTHER _____

FLUID _____

CYCLE REQUIREMENTS _____

RELIABILITY ESTIMATE TBD REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL TBD APPROVED BY _____ DATE _____

APPROVED BY _____ DATE _____

SPECIAL CONSIDERATIONS _____ APPROVED BY _____ DATE _____

PERTINENT REFERENCES _____

IDENTIFICATION NO. 16

RADIATION EFFECTS INFORMATION REQUEST

DATE November 10, 1970

SUBJECT Beryllium FUNCTIONAL CATEGORY Test & Evaluation

REF. ITEM SUPPORT EC-677559 REF. DATA ITEMS _____

INFORMATION DESIRED Effect of irradiation on thermal conductivity
(PURPOSE OF STUDY OR EXPERIMENT)

DATE REQUIRED CY 71

RELATION OF ITEM TO ENGINE Reflector segments outside core periphery and within pressure vessel
(LOCATION)

PER DRAWINGS, DATA ITEMS Dwg. 939J733

ENVIRONMENT (4) : NEUTRON FLUX _____ NEUTRON FLUENCE 4×10^{17} & 4×10^{19} n/cm² (E > 1 Mev)
(Ref. data items)

GAMMA DOSE RATE _____ GAMMA DOSE _____

TEMPERATURE 140°R - 750°R NUCLEAR HEATING RATE _____

PRESSURE _____ OTHER _____

FLUID _____

CYCLE REQUIREMENTS _____

RELIABILITY ESTIMATE Screening REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL Screening APPROVED BY _____ DATE _____

APPROVED BY _____ DATE _____

SPECIAL CONSIDERATIONS _____ APPROVED BY _____ DATE _____

PERTINENT REFERENCES _____

8-43

IDENTIFICATION NO. 17

RADIATION EFFECTS INFORMATION REQUEST

DATE November 10, 1970

SUBJECT Copper-Boron-10 (enriched and natural) Composite FUNCTIONAL CATEGORY Test & Evaluation

REF. ITEM SUPPORT EC-677559 REF. DATA ITEMS S-110

INFORMATION DESIRED Effect of irradiation on fracture strength
(PURPOSE OF STUDY OR EXPERIMENT)

DATE REQUIRED CY 71

RELATION OF ITEM TO ENGINE Vanes attached to beryllium drums in Reflector Assembly
(LOCATION)

PER DRAWINGS, DATA ITEMS Dwg. 939J733

ENVIRONMENT (4) : NEUTRON FLUX _____ NEUTRON FLUENCE 4×10^{18} & 4×10^{19} n/cm² (E>1Mev)
(Ref. data items)

GAMMA DOSE RATE _____ GAMMA DOSE _____

TEMPERATURE 140°R - 750°R NUCLEAR HEATING RATE _____

PRESSURE _____ OTHER _____

FLUID _____

CYCLE REQUIREMENTS _____

RELIABILITY ESTIMATE 99% REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL 95% APPROVED BY _____ DATE _____

APPROVED BY _____ DATE _____

SPECIAL CONSIDERATIONS _____ APPROVED BY _____ DATE _____

PERTINENT REFERENCES _____

IDENTIFICATION NO. 18

RADIATION EFFECTS INFORMATION REQUEST

DATE November 10, 1970

SUBJECT Copper-Boron-10 (enriched and natural) FUNCTIONAL CATEGORY Test & Evaluation

REF. ITEM SUPPORT EC-677559 REF. DATA ITEMS _____

INFORMATION DESIRED Effect of radiation on thermal conductivity
(PURPOSE OF STUDY OR EXPERIMENT)

DATE REQUIRED CY 71 - 72

RELATION OF ITEM TO ENGINE Vanes attached to beryllium drums in Reflector Assembly
(LOCATION)

PER DRAWINGS, DATA ITEMS Dwg. 939J733

ENVIRONMENT (4) : NEUTRON FLUX _____ NEUTRON FLUENCE 4×10^{18} & 4×10^{19} n/cm² (E>1 Mev)
(Ref. data items)

GAMMA DOSE RATE _____ GAMMA DOSE _____

TEMPERATURE 140°R - 750°R NUCLEAR HEATING RATE _____

PRESSURE _____ OTHER _____

FLUID _____

CYCLE REQUIREMENTS _____

RELIABILITY ESTIMATE Screening REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL Screening APPROVED BY _____ DATE _____

APPROVED BY _____ DATE _____

SPECIAL CONSIDERATIONS _____ APPROVED BY _____ DATE _____

PERTINENT REFERENCES _____

8-45

IDENTIFICATION NO. 19RADIATION EFFECTS INFORMATION REQUESTDATE November 10, 1970SUBJECT A - 286 Stainless Steel FUNCTIONAL CATEGORY Test & EvaluationREF. ITEM SUPPORT EC - 677559 REF. DATA ITEMS S - 110INFORMATION DESIRED Effects of irradiation on the Stress Relaxation properties of springs
(PURPOSE OF STUDY OR EXPERIMENT)DATE REQUIRED CY 70 - 71RELATION OF ITEM TO ENGINE Springs in Reflector Assembly
(LOCATION)PER DRAWINGS, DATA ITEMS Dwg 939J733ENVIRONMENT (4):
(Ref. data items) NEUTRON FLUX _____ NEUTRON FLUENCE 4.2×10^{18} & 4.2×10^{19} (E > Mev)

GAMMA DOSE RATE _____ GAMMA DOSE _____

TEMPERATURE 140 - 750 °R NUCLEAR HEATING RATE _____

PRESSURE _____ OTHER _____

FLUID _____

CYCLE REQUIREMENTS _____

RELIABILITY ESTIMATE TBD REQUESTED BY _____ DATE _____CONFIDENCE LEVEL TBD APPROVED _____ DATE _____

APPROVED _____ DATE _____

SPECIAL CONSIDERATIONS _____ APPROVED _____ DATE _____

PERTINENT REFERENCES _____

REPORTS (T-11%) RESPONDING TO REIRS

(None prepared at this issuance)

9.0 CONTROL DRUM DRIVE ASSEMBLY

9.1 SUMMARY AND CONCLUSION

In the NSS, there are eighteen (18) separate actuators for eighteen separate Control Drum Drive Assemblies (CDDA). Each CDDA consists of ninety-two (92) separate subcomponents which are fabricated from twelve (12) different materials. These materials have been selected for their individual applications after discarding others which were less acceptable. Therefore, significant Radiation Effects analysis has already been incorporated into the CDDA design.

Each remaining material or subcomponent has been analyzed in the projected thermal and nuclear environment for its ability to perform without significant degradation. Although directly applicable information is sparse, one can conclude that most of the metals employed will not undergo unacceptable changes. This is largely reasonable because of the relatively low radiation levels.

Less is known about the remaining materials. Thus, REIRs have been issued for these materials (copper wire, shrinkable tubing, Alnico V, Relay 5, Diallyl Phthallate, etc.). In addition, some questions remain for the major subcomponents, such as the motor and pot, as well as the complete actuator, to function in the projected environment and to allow the development of reliability assessments. Thus REIRs have also been issued for these major subcomponents.

9.2 INTRODUCTION

Actuator mechanisms and override devices for the Control Drum Drive Assembly (CDDA; EC-677556) will be provided by WANL in the NSS. The CDDA consists of eighteen (18) individual subassemblies which are mounted externally on the pressure vessel forward closure as shown in Figure 1.

The purpose of this analysis is to either substantiate the capability of the referenced design to withstand the projected radiation environment or to specify where additional radiation effects testing will be required. The twelve (12) different materials which are

utilized in the CDDA have been selected both from the standpoint of their functional application and relative resistance to radiation. Materials which clearly would be unable to withstand the referenced design radiation environment have already been discarded in a prior iteration.

Table 1 is a drum actuator assembly and actuator malfunction override device (AMOD) parts list which specifies each part, function, nomenclature, material selection and quantity. Table 2 is a material summary. Additional investigative effort is identified in Radiation Effects Information Requests (REIRs) where the analysis shows inconsistency between available evidence and the material application.

The CDDA consists of the following:⁽¹⁾

1. Activating mechanism - electric actuator which supplies torque to a control drum, imparting rotation to the drum in response to signals from the actuator controller;
2. Actuator malfunction override device (AMOD) which:
 - a. Acts as a drive shaft and transmits torque between the actuator and control drum;
 - b. If energized by the I & C system it disengages the actuator and rotates to a predetermined angle within $\pm 2.5\%$ from any position in the working range with a minimum torque value of 100 in-lbs;
3. Controller - provides signals to the actuator in response to input demand signals.

The eighteen (18) individual CDDA mechanisms and AMOD pairs are equally spaced at the pressure vessel circumference on the centerline of the control drum. Controllers are remotely located. Some important characteristics of the CDDA are as follows:

1. Working range - 170° (angular range)

2. Accuracy - 1° relative to command position throughout the working range;
3. Resolution - static resolution of 0.2° maximum;
4. Position indicator - $\pm 0.5^\circ$ of absolute;
5. Velocity - $12^\circ/\text{second}$;
6. Life - 150 hours operating life; 10^7 - 10^8 actuator motor steps.

A more complete description of the CDDA may be found in the E-105 data item.

9.3 ENVIRONMENTAL AND OPERATIONAL REQUIREMENTS

9.3.1 Nuclear Environment

9.3.1.1 Unshielded⁽²⁾

Neutron fluence

$E > 1.0 \text{ Mev}$	$- 3.4 \times 10^{17} \text{ n/cm}^2$
$0.4 \text{ ev} < E < 1.0 \text{ Mev}$	$- 3.4 \times 10^{18} \text{ n/cm}^2$
$E < 0.4 \text{ ev}$	$- 3.4 \times 10^{17} \text{ n/cm}^2$

Gamma - $1.1 \times 10^{11} \text{ ergs/gm (c)}$

9.3.1.2 Shielded (BATH plus lead)

Neutron Fluence

$E > 1.0 \text{ Mev}$	$- 1.0 \times 10^{16} \text{ n/cm}^2$
$0.4 \text{ ev} < E < 1.0 \text{ Mev}$	$- 2.2 \times 10^{16} \text{ n/cm}^2$
$E < 0.4 \text{ ev}$	$- 3.8 \times 10^{15} \text{ n/cm}^2$

Gamma - $2.9 \times 10^9 \text{ ergs/gm-C}$

(Note: In this analysis, the R-1 referenced design or shielded configuration shall be assumed; however, the testing suggested in Section 9.4 and summarized in the attached REIRs specify the unshielded environment).

9.3.2 Thermal Environment

Temperatures for each component are shown in Table 3.⁽³⁾ These are the steady-state operating temperatures for each material, but do not include uncertainties or temperature ranges during cooldown and storage because they are not necessary for the analysis presented below.

9.4 ANALYSIS

9.4.1 Stainless Steel (SS) 302

9.4.1.1 Non-NERVA Data and Experience

There has been very little investigative work carried out on SS 302. However SS 304 is nearly identical (0.15 carbon weight percent in SS 302 and 0.08 Carbon weight percent in SS 304) and it has become one of the standard materials utilized in the nuclear industry because its mechanical properties following irradiation are well understood. At temperatures near ambient, very little change in its mechanical properties have been observed.⁽⁴⁾ As expected, tensile strength and yield strength improve significantly at higher fast fluences ($>10^{20}$ n/cm²) and ductility decreases. Saturation is reached at approximately 2×10^{20} n/cm² ($E > 1$ Mev) if irradiation is performed at 50°C.⁽⁵⁾ However, no saturation is reached at 1.5×10^{21} n/cm² if the irradiation temperature is 290°C.⁽⁵⁾

Creep properties of Type 304 Stainless Steel as well as stress rupture tests have also shown negligible effects at these levels.^(6,7) Other properties do not appear to be significantly affected by irradiation, especially at these levels.

Less is known at cryogenic temperatures where strength, ductility and toughness should be maximized. Austenitic stainless steels, which are facecentered alloys, generally exhibit properties at cryogenic temperatures which are desirable. Stainless Steel 304 irradiated at 1.0×10^{17} n/cm² ($E > 0.5$ Mev) and at -257°C showed significant improvement in yield strength and ultimate tensile strength when tested at the same temperature. Notched tensile strength decreased slightly, from 193 to 173 psi, and elongation was virtually unchanged. Other related stainless steels showed similar trends.⁽⁴⁾

Lockheed Nuclear Products Company⁽¹⁰⁾ performed a rather extensive investigation of NERVA-type materials at 257°C (30°R). Stainless steel 304 was seen to "degrade" by approximately 10% following irradiation to 1×10^{17} n/cm² ($E > 0.5$ Mev) at 30°R and post-irradiation testing at the same temperature. Tests carried out in hydrogen show that Stainless 304, and Stainless 302, are not degraded significantly as a result of hydrogen embrittlement (unirradiated).⁽¹⁵⁾

9.4.1.2 NERVA Data and Experience

This material and similar stainless steels have been used in a variety of NRX components without apparent difficulty. No reports referencing these observations are available.

In addition, WANL has estimated the irradiated properties of SS 304 for NSS PDR. In Chapter 6.4.2 these estimates have been graphically compared to those cited in this data item. Over the range of NSS CDDA fluences and temperatures, this material exhibits generally increased ductility and decreased strength. Good agreement may be observed between the DRM and the cited data.

9.4.1.3 Conclusion

Thus, it may be concluded that stainless steels, including SS 304, show little detrimental mechanical property changes at the fluences projected for the CDDA. SS 302 has been examined at similar temperatures with little detrimental change. Due to their similarity, conclusions drawn for SS 302 are applicable to SS 304. It can be safely concluded that SS 302 performance will not degrade to any significant extent under these conditions.

9.4.2 Stainless Steel (SS) 440C (modified for use in cryogenic environment)

9.4.2.1 Non-NERVA Data and Experience

SS 440C is planned for the CDDA bearings. As is the case with SS 302, there is very little irradiation effects data available. However, REIC Report No. 45⁽⁴⁾ points out that SS 440C mechanical properties do improve with radiation at 7×10^{17} n/cm² ($E > 1$ Mev)

where both the irradiation and testing is performed at room temperature. Although SS 440C may be expected to be somewhat more radio-sensitive, due to its high carbon content (0.6 - 1.2% by weight), this fluence is certainly below the point where damage may be expected. (8,10)

9.4.2.2 NERVA Data and Experience

This material was used in bearing races and balls in NRX and XE Prime drum bearings without apparent difficulty. Similarly, ANSC has used this material in bearings without difficulty.

9.4.2.3 Conclusion

Although there is little data available at cryogenic temperatures, it can be inferred that radiation-induced degradation will be minimal at projected fluences.

9.4.3 A-286

9.4.3.1 Non-NERVA Data and Experience

This material has been irradiated at room temperature and above to fluences from 5×10^{19} to 7×10^{21} n/cm² ($E > 1$ Mev) with no significant degradation observed (11,12) in tensile properties. On the other hand, Moteff, et. al. (13) has shown significant embrittlement at a fast fluence of 1×10^{16} n/cm², at 1200°R, most probably due to the Boron-10 n, α reaction. Irradiation appears to significantly enhance fatigue properties. (14) The cryogenic work carried out by Lockheed (10) shows that A-286 is not significantly affected by irradiation at 1×10^{17} n/cm² ($E > 0.5$ Mev) where irradiation and test temperatures were approximately 30°R.

Similarly, A-286 does not appear to undergo hydrogen embrittlement to any significant extent. (16)

9.4.3.2 NERVA Data and Experience

An extensive program is currently underway to measure the effects of radiation on this material at a variety of applicable fluences and temperatures. In addition, WANL has estimated the irradiated properties of A-286 for NSS PDR⁽⁹⁾. These data show minor changes in strength and decreased ductility following irradiation. The data derived from NERVA studies underway as well as data available from the literature, will be thoroughly analyzed in Chapter 4, cluster hardware, which represents the most important application of this material in the NSS.

9.4.3.3 Conclusion

After the data discussed in 6.4.3.2 is generated, firm conclusions may be drawn. However, it can certainly be concluded that this material will withstand the projected CDDA environment.

9.4.4 Polyimide

9.4.4.1 Non-NERVA Data and Experience

This material has been recommended by the vendors for a variety of applications. It has been demonstrated to be resistant to projected NERVA gamma levels since it does not exhibit any radio sensitivity below 10^{10} ergs/gm (c).^(17, 20, 21)

9.4.4.2 NERVA Data and Experience

This material was used in actuators for NRX A-6 side-by-side tests⁽²²⁾ and in an irradiation test program at Plum Brook.⁽²³⁾ No failure or degradation can be traced to this material. However, there has been no quantitative evaluation of polyimide performance.

9.4.4.3 Conclusion

Polyimide has been investigated to a sufficient extent to conclude that radiation will not cause significant actuator degradation under the projected conditions.

9.4.5 PSC Durock

9.4.5.1 Non-NERVA Data and Experience

There is no directly applicable non-NERVA data.

9.4.5.2 NERVA Data and Experience

This material was used in actuators for NRX A-6 side-by-side tests⁽²²⁾ and at Plum Brook.⁽²³⁾ No failure or degradation can be traced to this material. However, there has been no quantitative evaluation performed for this material.

9.4.5.3 Conclusion

Additional information is necessary on this material to assess its ability to survive the projected environment. REIRs which summarize the needed information are attached to this chapter.

9.4.6 Copper Wire

9.4.6.1 Non-NERVA Data and Experience

Copper wire is known to undergo significant changes in resistivity during irradiation. At the projected fluences, changes up to 100% might be anticipated, although changes of a lower magnitude are more likely.^(25,26) This property has been measured at cryogenic temperatures in GTR 21 where there is additional evidence of substantial changes in the electrical resistivity of copper at low temperatures.⁽²⁷⁾ Similarly, investigations at General Electric have demonstrated significant changes in the electrical resistivity of copper at low temperatures.^(28,29) However, most damage will anneal out at NSS storage temperatures.

9.4.6.2 NERVA Data and Experience

NERVA test experience in transducer testing suggests that failures and degradation due to changes in the electrical resistance of copper wires have occurred.⁽²⁶⁾ However,

degradation has not been of sufficient magnitude to cause actuator performance degradation. ⁽²²⁾

9.4.6.3 Conclusion

There is sufficient evidence of significant changes in the thermophysical properties of copper to cause concern, despite the fact that actual tests have not shown actuator malfunction due to detrimental changes in copper. Thus, Radiation Effects testing appears necessary and REIRs which summarize the needed information are attached to this chapter.

9.4.7 Silicon Iron

9.4.7.1 Non-NERVA Data and Experience

No data on the irradiated properties of this material is available. However, it should be unaffected at these fluences except for minor changes in its magnetic properties. ⁽²⁵⁾

9.4.7.2 NERVA Data and Experience

No data is available.

9.4.7.3 Conclusions

Based on the dearth of data, Radiation Effects testing or analyses appears necessary and REIRs which summarize the needed information are attached to this chapter.

9.4.8 Alnico V

9.4.8.1 Non-NERVA Data and Experience

No data is available for this material. Based upon the performance of similar materials when irradiated, little change in its properties is expected at the projected fluence.

9.4.8.2 NERVA Data and Experience

This material has been utilized extensively in NERVA with no degradation noted. However, there has been no quantitative evaluation performed on the effects of radiation on ALINCO V.

9.4.8.3 Conclusion

Based on the dearth of data, Radiation Effects testing or analyses appears necessary and REIRs which summarize the needed information are attached.

9.4.9 Diallyl Phthalate

9.4.9.1 Non-NERVA Data and Experience

This material has been recommended by the vendor for this application (track elements) and there is some evidence that it will not be affected at the projected gamma fluences.

9.4.9.2 NERVA Data and Experience

This material has been used in past NERVA actuators with no apparent degradation. However, there is no quantitative evidence available.

9.4.9.3 Conclusion

It seems likely that diallyl phthalate will not degrade in the projected environment. However, additional Radiation Effects testing or analyses is necessary and REIRs which summarize these requirements are attached.

9.4.10. Relay 5

9.4.10.1 Non-NERVA Data and Experience

There is no specific data available on this material. However, some changes (0-20%) in magnetic properties can be expected based on work on related materials. ⁽²⁵⁾

9.4.10.2 NERVA Data and Experience

NERVA use of this material has indicated that no failures have occurred as a result of its usage. No quantitative evidence is available.

9.4.10.3 Conclusion

Based on the dearth of available data, Radiation Effects testing or analyses appears necessary and REIRs summarizing requirements are attached.

9.4.11 Brass with Gold Plate

9.4.11.1 Non-NERVA Data and Experience

No data is available. However, generalizing from other materials, no degradation is expected at the projected fluences.⁽²⁵⁾

9.4.11.2 NERVA Data and Experience

None Available

9.4.11.3 Conclusions

Due to the lack of information, Radiation Effects analysis or testing is required and REIRs summarizing requirements are attached to this chapter.

9.4.12 Shrinkable Tubing

9.4.12.1 Non-NERVA Data and Experience

No non-NERVA data is available.

9.4.12.2 NERVA Testing and Experience

In past XE Prime tests, this material has been utilized and no performance degradation has been traced to its usage. However, no quantitative information is available.

9.4.12.3 Conclusion

Based on the dearth of data, Radiation Effects testing or analysis is required and REIRs summarizing requirements are attached.

9.4.13 System or Component Data and Experience

9.4.13.1 Non-NERVA Data and Experience

Motors and potentiometers have been extensively used in nuclear systems and their ability to withstand radiation and thermal environments similar to the NERVA engine is well documented.^(30, 31, 32, 33, 34)

9.4.13.2 NERVA Data and Experience

There is no directly applicable NERVA experience. The tests at Plum Brook⁽²³⁾ and the NRX A-6⁽²²⁾ side-by-side are primarily qualitative but demonstrate that many of the proposed actuator components will withstand the proposed environment.

9.4.13.3 Conclusions

Although there is good evidence that the majority of materials utilized will withstand the projected environment, it is impossible to conclude irrevocably that the CDDA will function properly. Furthermore, a reliability determination must also be eventually made prior to R-1. Thus, REIRs have been suggested for the motor, potentiometer and complete actuator and are attached to this chapter.

TABLE 9-1
DRUM ACTUATOR AND AMOD PARTS LIST

<u>Part Nomenclature</u>	<u>Present Material Selection</u>	<u>Quantity</u>
Control Drum Actuator Assembly		1
Female Connector Assembly		1
Cable Clamp	SS 302	2
Cable Clamp Screws	SS 302 or SS 400	2
Screw Lock Wire	SS 302	AR
Connector Shell	SS 302	1
Female Connector Pin	SS 302 with Au Plate	22
Pin Insulator and Retainer	PSC Durock (XE Experience)	1
Connector Lock Spring	SS 302	1
Connector Retaining Nut Assembly		1
Retaining Nut	SS 302	1
Bayonett Pin	SS 302	3
Insulator Retaining Ring	SS 302	1
Motor Housing and Connector Assembly		1
Housing	SS 302	1
Male Connector Pin	SS 302 with Au Plate	22
Pin Insulator/Seal	PSC Durock (XE Exp.)	AR
Potentiometer Lead	Cu Wire-Polyimide Ins.	6
AMOD Trigger Lead	Cu Wire-Polyimide Ins.	2
Pin/Lead Junction Shrinkable Insulator	Shrinkable Tubing (XE Experience)	AR
Actuator Mounting Bolt	A-286	3
Actuator/Pedestal Interface Seal	SS 302 Pb Plate	1
Actuator/Pedestal Interface Dowels	SS 302	2
Potentiometer Indexing Dowel	SS 302	1

TABLE 9-1
DRUM ACTUATOR AND AMOD PARTS LIST
(CONTINUED)

<u>Part Nomenclature</u>	<u>Present Material Selection</u>	<u>Quantity</u>
Motor Assembly		1
Motor Forward End Bell	SS 302	1
Motor Stator Key	SS 302	1
Motor Stator Assembly		1
Stator Lamination	Silicon Iron	70
Lamination Bonding Material	TBD	AR
Stator Phase Winding	Cu with Polyimide	4
Stator Impregnant	Polyimide	AR
Stator Lead Insulating Sleeve	Polyimide	AR
Guard Plate	SS 302	1
Motor Rotor Assembly		1
Rotor Shaft	SS 302	1
Rotor Key	SS 302	1
Rotor Magnet	Alnico V, Alnico IX, Samarium Cobalt	1
Motor Bearing Preload Spring	SS 302	1
Forward Motor Bearing	SS 440C	1
Aft Motor Bearing	SS 440C	1
Motor Aft End Bell	SS 302	1
Transmission Planet Assembly		1
Planet Duplex Gear	A-286	1
Bearing Seat Ring	SS 302	1
Thrust Washer	SS 302	1
Planet Bearing Preload Spring	SS 302	1
Retaining Ring Lock	SS 302	1
Planet Retaining Ring	SS 302	1
Planet Bearing	SS 440C;	1

TABLE 9-1
DRUM ACTUATOR AND AMOD PARTS LIST
(CONTINUED)

<u>Part Nomenclature</u>	<u>Present Material Selection</u>	<u>Quantity</u>
Stationary Ring Gear	A-286	1
Output Mechanism Assembly		1
Output Mechanism Housing	SS 302	1
Output Shaft Bearing	SS 440C	2
Potentiometer Assembly	Polyimide	1
Potentiometer Resistance Track Assembly	Diallyl	1
Resistance Track Element	Phthalate	1
Potentiometer Terminal	TBD	6
Terminal Pressure Contact	TBD	6
Angular Indexing Washer	Polyimide	2
Potentiometer Clamp Insulation Ring	Polyimide	1
Potentiometer Clamp Spring	SS 302	1
Potentiometer Clamp Retaining Ring Lock	SS 302	1
Potentiometer Retaining Ring	SS 302	1
Potentiometer Wiper Assembly		2
Wiper Arm	SS 302	1
Wiper Insulator Plate	Polyimide	1
Wiper Contact Finger Support	TBD	1
Wiper Contact Finger	TBD	1
Wiper Eyelet Insulator	Polyimide	2
Wiper Eyelet	TBD	2
Wiper Arm Screws	SS 302	4
Output Shaft	A 286	1

TABLE 9-1
 DRUM ACTUATOR AND AMOD PARTS LIST
 (CONTINUED)

<u>Part Nomenclature</u>	<u>Present Material Slection</u>	<u>Quantity</u>
Output Shaft Bearing Preload Spring	SS 302	1
Output Shaft Retaining Ring Lock	SS 302	1
Output Shaft Retaining Ring	SS 302	1
Wiring Grommet	Polyimide	3
Output Mechanism Assembly	SS 302	6
AMOD Trigger Assembly		1
Trigger Coil Assembly		1
Trigger Coil Bobbin	Polyimide	1
Trigger Coil Terminal	Brass with Au Plate	2
Trigger Coil Winding	Cu with Polyimide Ins.	1
Trigger Coil Encapsulant	Polyimide	AR
Trigger Field Magnet Segment	Alnico V	6
Trigger Pole Face	Relay 5 (Alg. Lud.)	1
Trigger Spring	SS 302	1
Trigger Hammer	Relay 5	1
Trigger Cover Assembly		1
Hammer Retriever Cup	SS 302	1
Wiring Guard Tube	SS 304	1
Trigger Cover Screws	SS 302	6

TABLE 9-1
DRUM ACTUATOR AND AMOD PARTS LIST
(CONTINUED)

<u>Part Nomenclature</u>	<u>Present Material Selection</u>	<u>Quantity</u>
Drive Shaft Assembly		1
Ram Housing	A-286	1
Ram	A-286	1
Ram Housing/Shaft Tube Adaptor	SS 302	1
Drive Shaft Tube	SS 304	1
Thrust Collar	SS 302	1
Dash Pot Assembly		1
Piston	SS 302	1
Piston Ring	Polyimide	1
Coolant Filter Element	SS 302	1
AMOD Drive Spring	A-286	1
Drive Shaft/Drum Male Spline Coupling	A-286	1
Latch Ball	SS-440C	9
Latch Target Ring	A-286	1
Latch Spring	A-286	1
AMOD Cam Assembly		1
AMOD Cam	A-286	2
Cam Indexing Dowel	SS 302	2
Cam Mounting Screws	SS 302	6

TABLE 9-2
MATERIAL SUMMARY

SS 302

SS 440C

A-286

Polyimide

PSC Durock

Copper Wire with Polyimide Insulation

Shrinkable Tubing

Silicon Iron

Alnico V

Diallyl Phthalate

Brass with Gold Plate

Relay 5

TABLE 9-3
SUMMARY OF RADIATION EFFECTS ON ACTUATOR MATERIALS

<u>Part Nomenclature</u>	<u>Preliminary Material Selection</u>	<u>Quantity</u>	<u>Category</u> ⁽¹⁾	<u>Steady-State Operating Temperature °R</u>	<u>Remarks</u>
Control Drum Actuator Assembly		1			
Female Connector Assembly		1			
Cable Clamp	SS 302	2	I	530	
Cable Clamp Screws	SS 302 or SS 400	2	I	"	
Screw Lock Wire	SS 302	AR	I	"	
Connector Shell	SS 302	1	I	"	
Female Connector Pin	SS 302 with Au Plate	22	I	"	
Pin Insulator and Retainer	PSC Durock XE Experience	1	-	"	Good XE Experience
Connector Lock Spring	SS 302	11	I	"	
Connector Retaining Nut Assembly		1	I	"	
Retaining Nut	SS 302	1	I	"	
Bayonett Pin	SS 302	3	I	"	
Insulator Retaining Ring	SS 302	1	I	"	
Motor Housing and Connector Assembly		1		"	
Housing	SS 302	1	I	"	
Male Connector Pin	SS 302 with Au Plate	22	I	"	

(1) Category I - Material is not expected to exhibit significant changes under specified conditions

Category II - Material might be expected to exhibit significant changes under specified conditions and tests are recommended

TABLE 9-3 (CONTINUED)
SUMMARY OF RADIATION EFFECTS ON ACTUATOR MATERIALS

<u>Part Nomenclature</u>	<u>Preliminary Material Selection</u>	<u>Quantity</u>	<u>Category</u>	<u>Steady-State Operating Temperature °R</u>	<u>Remarks</u>
Pin Insulator/Seal	PSC Durock (XE Exp.)	AR	-	530	Good XE Experience
Potentiometer Lead	Cu Wire-Polyimide Ins.	6	II	"	Change in electrical resistance anticipated; REIR issued (No. <u>4</u>)
AMOD Trigger Lead	Cu Wire-Polyimide Ins.	2	II	"	Change in electrical resistance anticipated; REIR issued (No. <u>4</u>)
Pin/Lead Junction Shrinkable Insulator	Shrinkable Tubing (XE experience)	AR	-	"	Good XE Experience
Actuator Mounting Bolt	A-286	3	I	"	
Actuator/Pedestal Inter- face Seal	SS 302 Pb Plate	1	I	"	
Actuator/Pedestal Inter- face Dowels	SS 302	2	I	"	
Potentiometer Indexing Dowel	SS 302	1	I	"	
Motor Assembly		1			
Motor Forward End Bell	SS 302	1	I	448-717	
Motor Stator Key	SS 302	1	I	"	
Motor Stator Assembly		1			

TABLE 9-3 (CONTINUED)
SUMMARY OF RADIATION EFFECTS ON ACTUATOR MATERIALS

<u>Part Nomenclature</u>	<u>Preliminary Material Selection</u>	<u>Quantity</u>	<u>Category</u>	<u>Steady-State Operating Temperature °R</u>	<u>Remarks</u>
Stator Lamination	Silicon Iron	70	II	448-717	Little is known; REIR issued. (No. <u>4</u>)
Lamination Bonding Material	TBD	AR	-	"	
Stator Phase Winding	Cu with Polyimide	4	II	"	Same as Remark in Potentiometer Lead
Stator Impregnant	Polyimide	AR	I	"	
Stator Lead Insulating Sleeve	Polyimide	AR	I	"	
Guard Plate	SS 302	1	I	"	
Motor Rotor Assembly		1			
Rotor Shaft	SS 302	1	I	"	
Rotor Key	SS 302	1	I	"	
Rotor Magnet	Alnico V, Alnico IX Samarium Cobalt	1	II	"	Some change in magnetic properties anticipated; REIR issued (NO. <u>4</u>); use of Alnico V appears best due to past experience.
Motor Bearing Preload Spring	SS 302	1	I	"	
Forward Motor Bearing	SS 440C; Polyimide Retainer	1	I	"	
Aft Motor Bearing	SS 440C; Polyimide Retainer	1	I	"	

TABLE 9-3 (CONTINUED)
SUMMARY OF RADIATION EFFECTS ON ACTUATOR MATERIALS

<u>Part Nomenclature</u>	<u>Preliminary Material Selection</u>	<u>Quantity</u>	<u>Category</u>	<u>Steady-State Operating Temperature °R</u>	<u>Remarks</u>
Motor Aft End Bell	SS 302	1	I	448-717	
Transmission Planet Assembly		1			
Planet Duplex Gear	A-286	1	I	664	
Bearing Seat Ring	SS 302	1	I	"	
Thrust Washer	SS 302	1	I	"	
Planet Bearing Preload Spring	SS 302	1	I	"	
Retaining Ring Lock	SS 302	1	I	"	
Planet Retaining Ring	SS 302	1	I	"	
Planet Bearing	SS 440C; Polyimide	1	I	"	
Stationary Ring Gear	A-286	1	I	"	
Output Mechanism Assembly		1			
Output Mechanism Housing	SS 302	1	I	831	
Output Shaft Bearing	SS 440C	2	I	"	
Potentiometer Assembly	Polyimide Retainer	1	I	"	
Potentiometer Resistance Track Assembly		1			
Resistance Track Element	Diallyl Phthalate	1	II	"	Due to dearth of information, REIR has been issued (NO. <u>4</u>)

TABLE 9-3 (CONTINUED)
SUMMARY OF RADIATION EFFECTS ON ACTUATOR MATERIALS

<u>Part Nomenclature</u>	<u>Preliminary Material Selection</u>	<u>Quantity</u>	<u>Category</u>	<u>Steady-State Operating Temperature °R</u>	<u>Remarks</u>
Potentiometer Terminal	TBD	6	-	831	
Terminal Pressure Contact	TBD	6	-	"	
Angular Indexing Washer	Polyimide	2	I	"	
Potentiometer Clamp Insulating Ring	Polyimide	1	I	"	
Potentiometer Clamp Spring	SS 302	1	I	"	
Potentiometer Clamp Retaining Ring Lock	SS 302	1	I	"	
Potentiometer Retaining Ring	SS 302	1	I	"	
Potentiometer Wiper Assembly		2			
Wiper Arm	SS 302	1	I	"	
Wiper Insulator Plate	Polyimide	1	I	"	
Wiper Contact Finger Support	TBD	1	-	"	
Wiper Contact Finger	TBD	1	-	"	

TABLE 9-3 (CONTINUED)
SUMMARY OF RADIATION EFFECTS ON ACTUATOR MATERIALS

<u>Part Nomenclature</u>	<u>Preliminary Material Selection</u>	<u>Quantity</u>	<u>Category</u>	<u>Steady-State Operating Temperature °R</u>	<u>Remarks</u>
Wiper Eyelet Insulator	Polyimide	2	I	831	
Wiper Eyelet	TBD	2	-	"	
Wiper Arm Screws	SS 302	4	I	"	
Output Shaft	A-286	1	I	"	
Output Shaft Bearing Preload Spring	SS 302	1	I	"	
Output Shaft Retaining Ring Lock	SS 302	1	I	"	
Output Shaft Retaining Ring	SS 302	1	I	"	
Wiring Grommet	Polyimide	3	I	"	
Output Mechanism Assembly Mounting Screws	SS 302	6	I	"	
AMOD Trigger Assembly		1			
Trigger Coil Assembly		1			
Trigger Coil Bobbin	Polyimide	1	I	340	
Trigger Coil Terminal	Brass with Au Plate	2	I	"	
Trigger Coil Winding	Cu with Polyimide Ins.	1	II	"	Same as Potentiometer Lead.
Trigger Coil Encapsu- lant	Polyimide	AR	I	"	

TABLE 9-3 (CONTINUED)
SUMMARY OF RADIATION EFFECTS ON ACTUATOR MATERIALS

<u>Part Nomenclature</u>	<u>Preliminary Material Selection</u>	<u>Quantity</u>	<u>Category</u>	<u>Steady-State Operating Temperature °R</u>	<u>Remarks</u>
Trigger Field Magnet	Alnico V	6	II	340	Some change in magnetic properties expected; REIR issued (NO. <u>4</u>).
Trigger Pole Face	Relay 5 (Alg. Lud.)	1	II	"	Same as above; Reir No. <u>4</u>
Trigger Spring	SS 302	1	I	"	Some change in spring constants expected; no REIR issued because magnitude of change will not affect spring function.
Trigger Hammer	Relay 5	1	II		Same as Trigger Field Magnet.
Trigger Cover Assembly		1			
Hammer Retriever Cup	SS 302	1	I		
Wiring Guard Tube	SS 302	1	I	"	
Trigger Cover Screws	SS 302	6	I	"	
Actuator Malfunction Override Device Assembly		1			
Drive Shaft Assembly		1			
Ram Housing	A-286	1	I	"	
Ram	A-286	1	I	"	
Ram Housing/Shaft Tube Adaptor	SS 302	1	I	"	

TABLE 9-3 (CONTINUED)
SUMMARY OF RADIATION EFFECTS ON ACTUATOR MATERIALS

<u>Part Nomenclature</u>	<u>Preliminary Material Selection</u>	<u>Quantity</u>	<u>Category</u>	<u>Steady-State Operating Temperature °R</u>	<u>Remarks</u>
Drive Shaft Tube	SS 304	1	I	340	
Thrust Collar	SS 302	1	I	"	
Dash Pot Assembly		1	I	"	
Piston	SS 302	1	I	"	
Piston Ring	Polyimide	1	I	"	
Coolant Filter Element	SS 302	1	I	"	
AMOD Drive Spring	A-286	1	I	"	
Drive Shaft/Drum Male Spline Coupling	A-286	1	I	"	
Latch Ball	SS-440C	9	I	"	
Latch Target Ring	A-286	1	I	"	
Latch Spring	A-286	1	I	"	
AMOD Cam Assembly	A-286	1	I	"	
AMOD Cam	A-286	2	I	"	
Cam Indexing Dowel	SS 302	2	I	"	
Cam Mounting Screws	SS 302	6	I	"

Figure 9-1. Diagram of CDDA in Relation to NSS

REFERENCES

(Chapter 9 - CDDA)

1. "NERVA Specification No. CP-677555; Detail Specification, Part 1, Performance/Design and Qualification Requirements for the NERVA Nuclear Subsystem," Westinghouse Astronuclear Laboratory (July, 1970)
2. "Radiation Environment External to the Combined Reflector Shield Concept," DRM 51938, Westinghouse Astronuclear Laboratory (February, 1970)
3. "Thermal Analysis of Control Drum Actuator Device," DRM 52914, Westinghouse Astronuclear Laboratory, (August, 1970)
4. Kangilaski, M., "The Effects of Neutron Radiation on Structural Materials," REIC Report No. 45, (June, 1967)
5. Irvin, J. E., Bement, A. C., and Hoogland, R. G., "The Combined Effect of Temperature and Irradiation on the Mechanical Properties of Austenitic Stainless Steels," BNWL-1 (January, 1965)
6. Williams, J. A., and Carter, J. W., "The Creep of Annealed 304 Stainless Steel During Irradiation and its Engineering Significance," Symposium on Effects of Radiation on Structural Materials, ASTM-STP 426 (1967)
7. Bloom, E. E., and Weir, J. R., "In-Reactor and Post-Irradiation Creep Rupture Properties of Type 304 Stainless Steels," Trans. Am. Nuc. Soc. 10: 131-132 (June, 1967)
8. Aerojet General Corporation, RN-S-0290, "GTR-16 Radiation Effects Test on Structural Materials at -423 F," (November, 1965)
9. Schollenberger, R. S., DRM 51234, Westinghouse Astronuclear Laboratory (July, 1970)
10. Schwanbeck, C. A., Effect of Nuclear Radiation on Materials at Cryogenic Temperatures, Final Report NASA-CR-54881 (N66-24695) (January, 1965)
11. High Temperature Materials and Reactor Component Development Programs, Volume 1, Materials, Second Annual Report, GEMP-177A (February, 1963)
12. Collins, C. C., Conts, W. H., Hammons, G. H., and Robertshaw, F. C., "Radiation Effects on the Stress Rupture Properties of Inconel X and A-286," Apex 676 (December 1961).

REFERENCES (CONT'D)

13. Moteff, J., Robertshaw, F. C., and Kingsbury, F. D., "Effects of Neutron Irradiation on the Stress Rupture Properties of High Temperature Precipitation Hardening Alloys," J. Nuclear Materials, 17: 245-258 (1965)
14. "High Temperature Materials Program Progress Report No. 25, Part A," GEMP-25A (July, 1970)
15. "High Temperature Materials Program Progress Report No.27," Part A, GEMP-27A (September, 1970)
16. Chandler, W. Rocketdyne, Private Communication (June, 1970)
17. Welleff, W. and Accinelli, J. B., "Effects of Radiation on Bearing Cage Materials and Retainers at Liquid Hydrogen Environment," Trans. Am. Nuc. Soc. 10: 116-117 (June, 1967)
18. Schollenberger, R. S., DRM 51250, Westinghouse Astronuclear Laboratory (April, 1970)
19. Schollenberger, R. S., DRM 51298, Westinghouse Astronuclear Laboratory (August, 1970)
20. Sanders, R. H., and Welleff, W., "Final Report on GTR-17 Effects of Radiation on Organic Materials Irradiated in Liquid Nitrogen," Aerojet General Corporation, RN-S-0327, (March, 1967)
21. "Radiation Effects State of the Art, 1965-1966, REIC Report No. 42 (June, 1966)
22. NRX-A6 Final Report, Westinghouse Astronuclear Laboratory, TNR-224 (January, 1969)
23. Robinson, D. J., Mayer, J. T., and Wyer, R. R., "NERVA Control Drum Actuator Irradiation Test Program," NASA TN D-5347 (July, 1969)
24. Kirchner, J. F., and Bowman, R. E., "Effects of Radiation on Materials and Components," Reinhold Publishing Corporation, New York (1964)

REFERENCES (CONT'D)

25. Billington, D. S., and Crawford, J. H., "Radiation Damage in Solids," Princeton University Press, Princeton, New Jersey (1961)
26. Bush, N., and T. P. Merrick, DRM 51911, Westinghouse Astronuclear Laboratory (July, 1970)
27. "NERVA Irradiation Program, GTR Test 21, Volume 3 - Thermal Conductivity and Electrical Resistivity of Selected NERVA Materials," General Dynamics/Fort Worth Division Report FZK 351-3, NJD-20 (November, 1969)
28. Lucassen, P. G., and Walker, R. M., "Research Directed Toward the Study of the Radiation Damage Thresholds of the Elements," General Electric Company, Research Laboratory, Schenectady, New York, Scientific Report No. 5 (June, 1962)
29. Corbett, J. W., and Walker, R. M., "Discrete Recovery Spectrum Below 65°K in Irradiated Copper," *Phy. Rev.*, 110(3): 767-768 (May, 1958)
30. SNAP-8 Radiation Effects Test Program, Volume II, Lockheed Georgia Co., Marietta, Georgia (November, 1964)
31. "A Study of Nuclear Radiation Effects on Telemetry, Volume II, Design Allowable," L.T. V. Aeronautics (February, 1964)
32. Stapp, W. J., and Collins, C. G., "Analysis of Radiation Damage to Electric Components Containing Wire-Wound Coils," General Electric Co., (November, 1965)
33. SNAP-8 Material Report for January-June, 1964, Aerojet General Corp., Azusa, California (July, 1964)
34. "Electrical Component Technology for a 0.25 to 10 Megawatt Space Power System, Design Study," Westinghouse Electric Corporation, Aero. Electrical Div., Quarterly Progress Report (August, 1967)

RADIATION EFFECTS INFORMATION REQUESTS (REIRS)
(Chapter 9 - CDDA)

<u>No.</u>	<u>Description</u>
1	CDDA Potentiometer
2	CDDA Motor
3	CDDA and SSCV Actuator (complete)
4	CDDA Miscellaneous Materials

IDENTIFICATION NO. 1

RADIATION EFFECTS INFORMATION REQUEST

DATE August 20, 1970

SUBJECT Irradiation Testing of NERVA Control Drum Actuator Potentiometer FUNCTIONAL CATEGORY -

REF. ITEM SUPPORT = REF. DATA ITEMS S-110, Chapter 9 (November, 1970)

INFORMATION DESIRED To obtain data under environmental conditions of temperature, radiation, low pressure
(PURPOSE OF STUDY OR EXPERIMENT)

vibration, if possible) and establish concentrated wear characteristics for the control drum actuator potentiometer

DATE REQUIRED CDR

RELATION OF ITEM TO ENGINE Item(s) located externally on the pressure vessel forward closure & in the upper thrust structure/
(LOCATION)

PER DRAWINGS, DATA ITEMS Reference Drawing

ENVIRONMENT (4): NEUTRON FLUX N/A NEUTRON FLUENCE 1×10^{17} n/cm² (E > 1 Mev)
(Ref. data items)

GAMMA DOSE RATE N/A GAMMA DOSE 1×10^{11} ergs/cm(c)

TEMPERATURE 140°R - 540°R NUCLEAR HEATING RATE TBD

PRESSURE TBD OTHER N/A

FLUID LH₂ or LN₂

CYCLE REQUIREMENTS TBD

RELIABILITY ESTIMATE 99.99 REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL 90% APPROVED BY _____ DATE _____

APPROVED BY _____ DATE _____

SPECIAL CONSIDERATIONS N/A APPROVED BY _____ DATE _____

PERTINENT REFERENCES TIS No. 41

IDENTIFICATION NO. 2

RADIATION EFFECTS INFORMATION REQUEST

DATE August 20, 1970

SUBJECT Irradiation Testing of NERVA Control Drum Motor FUNCTIONAL CATEGORY -

REF. ITEM SUPPORT - REF. DATA ITEMS S-110, Chapter 9 (November, 1970)

INFORMATION DESIRED To determine the operating characteristics of the NERVA control drum motor under
(PURPOSE OF STUDY OR EXPERIMENT)
dynamic loads and expected environments

RELATION OF ITEM TO ENGINE Items located externally on the pressure vessel forward closure and in the
(LOCATION) upper thrust structure
PER DRAWINGS, DATA ITEMS Reference Drawing

ENVIRONMENT (4): NEUTRON FLUX N/A NEUTRON FLUENCE 1×10^{17} n/cm² (E > 1 Mev)
(Ref. data items) GAMMA DOSE RATE N/A GAMMA DOSE 1×10^{11} ergs/cm(c)
TEMPERATURE 140°R (LN₂) - 540°R NUCLEAR HEATING RATE TBD
PRESSURE TBD OTHER N/A
FLUID LN₂ or LH₂
CYCLE REQUIREMENTS TBD

RELIABILITY ESTIMATE 99.9 REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL 90% APPROVED BY _____ DATE _____

APPROVED BY _____ DATE _____

SPECIAL CONSIDERATIONS N/A APPROVED BY _____ DATE _____

PERTINENT REFERENCES TIS No. 42

IDENTIFICATION NO. 3

RADIATION EFFECTS INFORMATION REQUEST

DATE August 20, 1970

SUBJECT Irradiation Testing of NERVA Control Drum and SSCV Actuator FUNCTIONAL CATEGORY -

REF. ITEM SUPPORT - REF. DATA ITEMS S-110, Chapter 9 (November, 1970)

INFORMATION DESIRED To determine wear characteristics of control drum and SSCV actuator under environmental
(PURPOSE OF STUDY OR EXPERIMENT)
conditions of temperature, radiation, pressure (and vibration if possible).

DATE REQUIRED CDR
Actuators located externally on the pressure vessel forward closure and the
RELATION OF ITEM TO ENGINE upper thrust structure.
(LOCATION)
PER DRAWINGS, DATA ITEMS Reference Drawing

ENVIRONMENT (4): NEUTRON FLUX N/A NEUTRON FLUENCE 1×10^{17} n/cm² (E > 1 Mev)
(Ref. data items) GAMMA DOSE RATE N/A GAMMA DOSE 1×10^{11} ergs/cm(c)
TEMPERATURE 140°R - 540°R NUCLEAR HEATING RATE TBD
PRESSURE TBD OTHER N/A
FLUID LH₂ or LN₂
CYCLE REQUIREMENTS TBD

RELIABILITY ESTIMATE 99.9 REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL 90% APPROVED BY _____ DATE _____

APPROVED BY _____ DATE _____

SPECIAL CONSIDERATIONS N/A APPROVED BY _____ DATE _____

PERTINENT REFERENCES TIS No.

IDENTIFICATION NO. 4

RADIATION EFFECTS INFORMATION REQUEST

DATE August 20, 1970

SUBJECT Irradiation testing of miscellaneous NERVA actuator materials FUNCTIONAL CATEGORY -

REF. ITEM SUPPORT - REF. DATA ITEMS S-110, Chapter 9 (November, 1970)

INFORMATION DESIRED To determine if radiation in LN₂ causes degradation in performance of PSC Durock
(PURPOSE OF STUDY OR EXPERIMENT)

copper wire, shrinkable tubing, diallyl phthallate, silicon iron, alnico V, Relay 5, and

brass with gold plate.

DATE REQUIRED CDR

RELATION OF ITEM TO ENGINE Materials located in actuators located externally on pressure vessel forward
(LOCATION) closure and the upper thrust structure.

PER DRAWINGS, DATA ITEMS Reference Drawing

ENVIRONMENT (4): NEUTRON FLUX N/A NEUTRON FLUENCE 1×10^{17} n/cm² (E > 1 Mev)
(Ref. data items)

GAMMA DOSE RATE N/A

GAMMA DOSE 1×10^{11} ergs/cm(c)

TEMPERATURE 140°R - 540°R

NUCLEAR HEATING RATE TBD

PRESSURE TBD

OTHER N/A

FLUID LN₂

CYCLE REQUIREMENTS N/A

RELIABILITY ESTIMATE N/A

REQUESTED BY DATE

CONFIDENCE LEVEL N/A

APPROVED BY DATE

APPROVED BY DATE

SPECIAL CONSIDERATIONS N/A

APPROVED BY DATE

PERTINENT REFERENCES N/A

REPORTS (T-119s) RESPONDING TO REIRs

(None prepared at this issuance)

10.0 SUPPORT STEM COOLANT VALVE

10.1 SUMMARY AND CONCLUSIONS

In the NSS, there are two (2) separate actuators for two separate support stem coolant valves (SSCV). Each SSCV actuator consists of approximately ninety (90) separate subcomponents which are fabricated from twelve (12) different materials. These materials have been selected for their individual applications after discarding others which were less acceptable. Therefore, significant Radiation Effects analysis has already been incorporated into the SSCV Actuator design.

In Chapter 9, each material or subcomponent for the Control Drum Drive Assembly (CDDA) has been analyzed in the projected thermal and nuclear environment for its ability to perform without significant degradation. It was concluded that most of the metals employed will not undergo unacceptable property changes at the low radiation levels anticipated. Less is known about the non-metals and REIRs were issued for these materials and the motor, pot and complete actuator.

The same conclusions must be applicable for the SSCV actuator because these two assemblies are still further from the NSS than the CDDA. Thus, these conclusions drawn for the CDDA, as well as the REIRs suggested, are applicable to the SSCV Actuator.

10.2 INTRODUCTION

Actuator mechanisms for the Structural Support Coolant Valve (SSCVA; EC-677576) will be provided by WANL. The Structural Support Coolant Valve will be provided by ANSC and the S-110 chapter analyzing the effects of radiation on this valve is currently being prepared. The SSCV consists of two (2) individual subassemblies which are located in the upper thrust structure approximately twenty (20) inches forward of the external shield. The actuating mechanisms are essentially identical to those used in the CDDA (EC-677556).

The purpose of this analysis is to either substantiate the capability of the referenced design to withstand the projected radiation environment or to specify where additional radiation effects testing will be required. The 12 materials utilized in the SSCV actuator have been selected both from the standpoint of their functional application and relative

resistance to radiation. Materials which clearly would be unable to withstand the referenced design radiation environment have already been discarded in a prior iteration.

Table 1 is a drum actuator assembly and parts list which specifies each part, function, nomenclature, material selection, and quantity. Table 2 is a material summary. Additional investigative effort is identified in Radiation Effects Information Requests (REIRs) where the analysis shows inconsistency between available evidence and the material application.

The SSCVA consists of the following:

1. Actuator mechanisms - electric actuator which supplies torque to the input shaft;
2. Coupling - transmits torque from actuator to valve; and
3. Controller - provides additional signals to the actuator.

Some important characteristics of the SSCVA are as follows:

1. Repeatability - 0.3° ;
2. Resolution - static resolution of 0.30 maximum;
3. Position indication - $\pm 0.5^{\circ}$ of absolute;
4. Life - 150 hours; 10^7 - 10^8 actuator motor steps; and
5. Weight - 15 lbs

A more complete description of the SSCVA may be found in the E-105 data item

10.3 ENVIRONMENTAL AND OPERATIONAL REQUIREMENTS

10.3.1 Nuclear

Neutron Fluence

$E > 1.0 \text{ Mev}$	$1.0 \times 10^{15} \text{ n/cm}^2$
$0.4 \text{ ev} < E < 1.0 \text{ Mev}$	$4.8 \times 10^{15} \text{ n/cm}^2$
$E < 0.4 \text{ ev}$	$1.6 \times 10^{15} \text{ n/cm}^2$

Gamma 1.8×10^7 Rad-C

(Note: This assumes no external shield; i. e., R-1 reference design.)

10.3.2 Temperature

TBD

10.3.3 Others

Not applicable

10.4 ANALYSIS

The analysis provided in 9.4 for the CDDA will apply. Since the radiation levels experienced in the SSCV are lower than is the case for the CDDA analyzed in 9.4, it shall be assumed that the analyses will be conservative.

Similarly, the conclusions for the SSCVA are identical to those reached for the CDDA. However, these conclusions are also conservative. Briefly stated, these conclusions are as follows:

1. No additional analyses or testing will be required on SS 302, SS 440C, A-286 and polyimide because these materials have been demonstrated to withstand the projected environment.

2. Limited applicable data are available for PSC Durock, copper wire, shrinkable tubing, Silicon Iron, Alnico V, diallyl phthalate, brass with gold plate, and Relay 5. Thus, some additional radiation effects testing or analysis is necessary. That proposed in Chapter 9 will suffice.

3. The systems' tests proposed for the motor, pot and complete actuator in Chapter 9 will be conservatively applicable to the SSCV actuator.

TABLE 10-1
(Identical to 9-1)

TABLE 10-2
(Identical to 9-2)

LIST OF REFERENCES
(Chapter 10 - SSCVA)

The List of References in Chapter 9 (CDDA) are Applicable

RADIATION EFFECTS INFORMATION REQUESTS (REIRs)
(Chapter 10 - SSCVA)

<u>No.</u>	<u>Description</u>
1	CDDA (SSCVA) Potentiometer
2	CDDA (SSCVA) Motor
3	CDDA (SSCVA) Actuator (complete)
4	CDDA (SSCVA) Miscellaneous Materials

(Note: These REIRs may be found in Chapter 9)

REPORT (T-119s) RESPONDING TO REIRS
None Prepared at this Issuance.

11.0 NSS INSTRUMENTATION

11.1 RESISTANCE TEMPERATURE THERMOMETER (RTT)

11.1.1 SUMMARY AND CONCLUSIONS

Analysis of radiation effects testing of RTTs in the NERVA Technology Program reveals that additional testing is necessary to assure satisfactory performance of the RTTs in the NERVA reactor.

Gamma and fast neutron radiation appear to cause the transducer output sensitivity shift observed for most of the tests. For the most recent RTT design, it is believed that fast neutrons ($E > 1.0$ Mev) cause the majority of the RTT damage.

Available test data indicate that RTTs have been exposed to only a small percentage of the total fast neutron fluence and total gamma dose expected in the NERVA reactor RTT location.

Additional testing is proposed on Radiation Effects Information Requests (REIRs) to isolate more definitively than previous testing the magnitude of radiation damage, the type of radiation causing the damage, and the RTT component(s) most susceptible to damage.

By progressively identifying and improving the components most susceptible to radiation damage, the RTT design may be "hardened".

11.1.2 INTRODUCTION

11.1.2.1 Transducer Description

The sensing end of the Resistance Temperature Thermometer (RTT) is composed of a platinum wire wound on a ceramic (MgO) mandrel which supports the platinum wire in a strain free configuration. Tem-Tech Model 1137 has a 1,000 ohm, 0.0006 inch diameter, 64 inch long platinum sensing wire. WANL Drawing 979D161⁽¹⁾ describes in further detail the configuration of the Tem-Tech transducer selected for NERVA - NSS utilization.

This type of transducer utilizes the principle whereby the resistivity of the wire varies linearly as a function of temperature. A constant current of about two milliamps is

provided as an input while the voltage generated is measured. A change in temperature at the sensing end provides a corresponding change in voltage, which, after calculation of the platinum wire resistance indicates the new temperature of the medium. Thus, a change in temperature causes a change in the platinum wire resistance.

11.1.2.2 Location of Transducer

Platinum wire sensing element RTTs will be mounted internal to the NERVA NSS pressure vessel (See Figure 11.1-1) and used to measure the reflector inlet plenum hydrogen temperature during reactor operations. Installation details of the RTT in the NSS Reflector inlet are described in WANL DRM 52033⁽²⁾. The RTTs must provide an accurate determination of the hydrogen temperature for computation of reactor thermal power and for evaluation of reflector thermal-hydraulic performance.

11.1.3 ENVIRONMENTAL AND OPERATIONAL REQUIREMENTS

The final design RTT to be used for measuring hydrogen temperature at the NERVA NSS reflector inlet plenum must be capable of withstanding the nuclear environment of 11.1.3.1 and the non-nuclear environment of 11.1.3.2. In addition, the transducer must comply with the performance characteristics designated in 11.1.3.3.

11.1.3.1 Nuclear Environment

Listed below are the approximate radiation levels anticipated at the reflector inlet plenum RTT location⁽³⁾ during rated reactor power:

<u>Neutron flux</u>	
E > 1.0 Mev	$5.1 \times 10^{13} \text{ n/cm}^2 - \text{sec}$
.4 ev < E < 1.0 Mev	$3.6 \times 10^{14} \text{ n/cm}^2 - \text{sec}$
E < .4 ev	$6.9 \times 10^{13} \text{ n/cm}^2 - \text{sec}$
<u>Neutron Fluence</u>	
E > 1.0 Mev	$1.9 \times 10^{18} \text{ n/cm}^2$
.4 ev < E < 1.0 Mev	$1.3 \times 10^{19} \text{ n/cm}^2$
E < .4 ev	$2.5 \times 10^{18} \text{ n/cm}^2$

Gamma dose rate	4.4×10^8 RADS(C)/HR
Total gamma dose	4.4×10^9 RADS(C)
Heating rate	1.23 W/gm(C)

11.1.3.2 Non-Nuclear Environment

At this time, the NERVA Engine design is preliminary. The reflector inlet plenum hydrogen flow rate, pressure, and temperature will depend on the final engine reference design⁽⁴⁾ and the engine mode of operation.

The present composite core, start-of-life, reflector inlet plenum, rated-engine power state points are⁽⁵⁾:

hydrogen mass velocity	TBD lbs/sec / in ² flow area
pressure	1134 psia
temperature	192°R
cycles	As defined in reference 5

If engine conditioning should be required for turbine operation, the reflector would require thermal conditioning. This would be accomplished by bringing the reactor to a low power level (under no flow conditions) and the RTTs could possibly experience a temperature of up to 700°R. In the proximity of Earth, RTT temperatures (non-rotating vehicle) would be about 450°R to 550°R (reactor not operating).⁽⁶⁾

11.1.3.3 Transducer Characteristics

The transducer and integral sheath cable must measure temperature to an accuracy of $\pm 5^\circ\text{R}$ ⁽⁷⁾ and must be capable of operation through the temperature range designated on WANL Drawing No. 979D161.⁽¹⁾ Equipment Specification E-677565⁽⁸⁾ describes the response time of the thermometer.

Additional detail design and equipment specification information for the transducer is obtainable from References (1) and (8).

WANL DRM 51912⁽⁹⁾ describes the logic employed to select RTTs, rather than thermocouples, for the NSS reflector inlet plenum application.

11.1.4 ANALYSIS

11.1.4.1 Transducer Assembly

11.1.4.1.1 Non-NERVA Technology

Little information exists in non-NERVA technology for RTTs tested at cryogenic temperatures and the high radiation levels expected during NERVA operation.

Lockheed Georgia Company^(10, 11) reported that no radiation effects occurred at LH₂ temperature on Rosemount (M/N 134EB400 and M/N 150BH) platinum wire sensing element RTTs exposed to a fast neutron fluence of 1×10^{15} n/cm² (E > .5 Mev) and a gamma dose rate of 2×10^8 RADS (C)/HR.

Bendix Corporation^(10, 12) reported that some radiation effects occurred at LH₂ temperature on Rosemount (M/N 134EB186) platinum wire sensing element RTTs exposed to a fast neutron fluence of 3.7×10^{16} n/cm² (E > 2.0 Mev) and a gamma dose rate of 5×10^9 RADS (C)/hr. The permanent temperature deviation ranged from .5°R to 1.4°R and the gamma heating error (usually temporary) was 1.3°R.

It appears that insufficient data is available in non-NERVA technology testing to predict platinum wire sensing element RTT performance in the NERVA nuclear environment.

11.1.4.1.2 NERVA Technology

Table 11.1-1 condenses pertinent information from previous NERVA technology program platinum wire sensing element RTT radiation testing.

The testing consisted of two types:

- (1) Transducer diagnostic
- (2) Reactor diagnostic

The transducer diagnostic tests were planned to determine what effect radiation had on the transducer electrical or mechanical output. The reactor diagnostic test primary objective was to use the transducer to control and evaluate performance of the reactor. Evaluation of the transducer performance was usually a secondary objective of the reactor diagnostic test.

Examination of Table 11.1-1 reveals the majority of the RTTs tested experienced a change in electrical resistance due to gamma dose rate and/or fast neutron fluence. WANL-DRM-51911⁽¹³⁾ reported that the failure mode mechanism (change in resistance) in Table 11.1-1 resulted in a transducer electrical output sensitivity shift during irradiation. Note that sensitivity shift is listed, with qualification, as a failure mode in Table 11.1-1 because none of the referenced documents referred to this behavior of the transducers as an output sensitivity shift.

To understand this phenomenon, consider the RTT sensitivity to be the ratio of dT/dE , where T is the measured temperature input and E is the transducer output in volts. When the electrical resistance of the RTT wire changes, at some constant temperature, the voltage output changes to result in a sensitivity shift. During irradiation, data from the references of Table 11.1-1 indicate that the resistance of the platinum wire and in turn the ratio dT/dE is changing instantaneously and continuously. This change has been referred to in WANL DRM 51911 to be a sensitivity shift in the transducer output and will be referred to hereafter in this report as a sensitivity shift. C. R. Tallman⁽⁴⁾ discusses in greater detail the technical aspects of RTTs.

The magnitude of the sensitivity shift is dependent on several variables:

- (1) The design of the RTT,
- (2) The gamma dose rate,
- (3) The integrated fast neutron flux (fast neutron fluence)

The transducer test radiation levels for each test are listed in Table 11.1-1. In some cases, the radiation exposure rates or total exposure is not given because this data is not reported in the documents referenced.⁽¹⁵⁻³³⁾

In several of the earlier tests (PBRF-4, GTR-12A, GTR-15), the gamma dose rate was the reported cause of the initial temperature deviations observed. RN-S-0157⁽¹⁵⁾ (GTR-12A) reported that when the effects of gamma radiation were analytically removed, further temperature deviations varied linearly with the integrated neutron flux ($E > 1.0$ Mev). Approximately 55% of the effects due to fast neutrons were annealed out at room temperature after irradiation during the GTR-12A testing.

As reported in RN-S-0251⁽¹⁶⁾ the magnitude of the temperature deviation caused by integrated fast neutrons appears inversely proportional to the platinum resistance wire diameter. Test data from GTR-7A and GTR-15 suggests that the optimum wire diameter, for radiation effects considerations, is about 0.001 inches (1 mil).

RTTs tested during the GTR-18 test were shielded for photons, thermal neutrons, and epithermal neutrons and experienced temperature deviations as great as unshielded units exposed to radiation. The reported test results⁽¹⁷⁾ indicated that the radiation damage to platinum resistance wire temperature transducers at cryogenic temperatures was caused by fast neutrons in the reactor spectrum of the GTR.

Figure 11.1-2 shows the reflector inlet plenum hydrogen temperature measured by the Tem Tech RTTs during the 60 minute full power hold NRX-A6 reactor test. The transducer outputs indicated a gradual 5°R to 10°R increase in temperature. Since thermocouples installed near the RTTs indicated relatively constant temperatures, the change in RTT output (sensitivity shift) was attributed to the radiation environment. The resistance change, as reported in TME 1863,⁽¹⁸⁾ was caused by the presence of fast neutrons ($E > 1.0$ Mev) and was observable above a fast neutron fluence of $5 \times 10^{14} \text{ n/cm}^2$ ($E > 1.0$ Mev).

Listed below are the fast neutron flux levels, the fast neutron fluence, the gamma dose rate, and the gamma dose which the selected Tem Tech transducers experienced in the GTR-18 and NRX-A6 tests. Also listed for comparison are the radiation levels expected at the NERVA RTT location.⁽⁷⁾

	<u>GTR-18</u>	<u>NRX-A6</u>	<u>NERVA</u>
Fast Neutron Flux, $n/cm^2 - sec$	6.5×10^{10}	3.0×10^{13}	5.1×10^{13}
Fast Neutron Fluence, n/cm^2	4.7×10^{16}	1.1×10^{17}	1.90×10^{18}
Gamma Dose Rate, RADS(C)/HR	9.0×10^6	8.0×10^8	4.4×10^8
Total Gamma Dose, RADS(C)	2.0×10^9	8.0×10^8	4.4×10^9
Temperature, $^{\circ}R$	37	165	192

Figure 11.1-3 presents the RTT maximum radiation levels in terms of percent of the radiation levels expected for NERVA. Obviously, the RTTs have been exposed to a small percentage of the total fast neutron fluence and total gamma dose expected in the NERVA reactor.

11.1.4.1.3 Conclusion

Post GTR-12A annealing tests showed that the effects due to fast neutrons were 55% removed (annealed out) after a temperature rise to room temperature.

The temperature extremes to be experienced by RTTs due to thermal cycling will vary for ground and flight reactor operation.

During reactor ground tests the RTTs will be initially at atmospheric temperature and will drop to a hydrogen temperature dependent on the reactor mode of operation. After reactor shutdown and cooldown, the RTT will warm up to atmospheric temperatures.

During flight operation, the initial temperature of any RTT will depend on the vehicle position in space and the passive temperature control methods adopted for the engine.

If engine conditioning should be required for turbine operation, the reflector would require thermal conditioning. This would be accomplished by bringing the reactor to a low power level (under no flow conditions) and the RTTs could possibly experience a temperature of up to $700^{\circ}R$.

Inconclusive test data, recent changes in NERVA Engine Reference Design and the variability of the RTT operational thermal environment necessitates that additional radiation testing and annealing testing be performed to assure satisfactory performance of the RTTs in the NERVA reactor. REIRs are included summarizing the test requirements for additional testing.

TABLE 11.1-1
TRANSDUCER TEST SURVEY, NERVA TECHNOLOGY PROGRAM

TEST AND DATE	DOCUMENT REFERENCES	TRANSDUCER MANUFACTURER AND MODEL NUMBER	NO. OF XDCRS TESTED	NO. OF XDCRS IN OBSERVED FAILURE MODE	DIAGNOSTIC TEST TYPE	TRANSDUCER ENVIRONMENT				OBSERVED FAILURE MODE							FAILURE MODE MECHANISM		GAMMA HEATING	TEST AND DATE		
						GAMMA DOSE RATE RADS (C)/HR	TOTAL GAMMA DOSE ~RADS (C)	FAST NEUTRON FLUX E > 1.0 Mev ~n/cm ² - sec	FAST NEUTRON FLUENCE E > 1.0 Mev ~n/cm ²	OR TEMPERATURE	SENSITIVITY SHIFT	ZERO SHIFT	DRIFT	NON-LINEARITY	ZERO OUTPUT	OFF-SCALE OUTPUT	OTHER	CHANGE IN RESISTANCE			CHANGE IN MAGNETIC PROPERTIES	OTHER
GTR-4 5/63	FZK-170-1 TNR-145	Winsco 2543 Engelhard 30707-5A	2	2	Transducer	4.1 (6)	4.1 (8)	5.7 (10)	2.0 (16)	142-450	←	---	---	---	---	NRE	---	---	---	---	GTR-4 5/63	
GTR-7A 8/63	FZK-170-4 TME-700	Engelhard 30707-5A	5	0		4.3 (6)	4.3 (8)	5.8 (10)	2.1 (16)	140	←	---	---	---	---	NRE	---	---	---	---	GTR-7A 8/63	
PBRF-4 9/63	TME-691	Winsco 2405	2	2	Transducer	5.3 (8)	3.2 (10)	9.4 (12)	2.0 (18)	572	** X	---	---	NP	---	---	X	---	---	X	PBRF-4 9/63	
GTR-9 10/63	FZK-170-7 TME-786	Winsco 2611 Microdot TT1415-2-1-T-3 Microdot TT1315-2-1-T-3	3	1	Transducer	2.8 (7)	2.8 (9)	3.9 (11)	1.4 (17)	150-160	X	---	---	NP	---	---	X	---	---	X	GTR-9 10/63	
GTR-12A 4/64	FZK-184-2 RN-S-0251 RN-S-0121 RN-S-0157	Rosemount 134EB250 Rosemount 110AA150 Winsco 2448-5	5	5		1.7 (7)	2.0 (9)	NP *	2.5 (16)	37	←	---	---	NP	---	---	X	---	---	X	GTR-12 3/64	
GTR-14 7/64	FZK-184-4 RNS-0251	Rosemount 134EBX1,X2,250 Rosemount 110AA-150	5	5	Transducer	NP*	8.3 (8)	NP *	1.0 (16)	37	** X	---	---	NP	---	---	X	---	---	X	GTR-14 7/64	
NRX-A2 9/64	FNS-0193 PNS-0318	Rosemount 134EB-250	7	0	Reactor	NP	NP	NP	6 (15)	NP	←	---	---	---	---	NRE	---	---	---	---	NRX-A2 9/64	
GTR-15 12/64	FZK-184-6 RN-S-0251	Rosemount 134GT Rosemount 134EB250 Rosemount 134EB1X Rosemount 134EB2X Rosemount 110AA150 Winsco 2448-5	2	2	Transducer	NP*	2.8 (9) to 3.8 (9)	NP *	3.9 (16) 6.4 (16)	37	** X	---	---	NP	---	---	X	---	---	---	GTR-15 12/64	
NRX-A3 5/65	RN-S-0258 PN-S-0318	Rosemount 134EB	8	8		NP *	NP *	NP *	NP **	NP	←	---	---	---	---	---	---	---	---	---	---	NRX-A3 5/65
NRX-EST 3/66	RNS-0396	Rosemount 134EB Rosemount 134EB	8	0		NP *	6 (8) 2 (6)	NP *	9.3 (16) 5 (13)	NP	←	---	---	---	---	NRE	---	---	---	---	---	NRX-EST 3/66
NRX-A5 5/66	RN-S-0397	Rosemount 134EB250	1	0		NP *	1 (8)	NP *	2 (16)	60	←	---	---	---	---	NRE	---	---	---	---	---	NRX-A5 5/66
GTR-18 6/66	FZK-310-1 RNS-0370	Rosemount 134EB186 Rosemount 137CJ Tem Tech 1881-2 Rosemount 134EB1X2X250 Rosemount 110AA	11	11		1 (7)	2.1 (9)	6.8 (10)	4.9 (16)	37	** X	---	---	NP	---	---	---	X	---	---	X	GTR-18 6/66
NRX-A6 12/67	RNS-0505 TME-1863	Tem Tech 1137-1	6	6	Reactor	8 (8)	8 (8)	3 (13)	1.1 (17)	165	** X	---	---	NP	---	---	X	---	---	---	NRX-A6 12/67	
NRX-A6 12/67	RN-S-0468	Rosemount 134JAT	4	2	Reactor	1.3 (8)	1.3 (8)	1.3 (13)	4.0 (16)	50	** X	---	---	---	---	---	X	---	---	X	NRX-A6 12/67	

NRE - No apparent radiation effects

* NP- Indicates that the data was not presented in available references.

FZK - General Dynamics/Fort Worth Report; TNR, TME-WANL Reports; RN-S- ANSC Reports

** Not presented in referenced report in this form. Documents described observed failure mode as a change in the ratio dT/dE which is referred to here as "sensitivity shift".

TABLE 11.1-1 (CONTINUED)

TRANSDUCER TEST SURVEY, NERVA TECHNOLOGY PROGRAM

TEST AND DATE	REACTOR ENVIRONMENT TRANSDUCER FAILURE SOURCE					JUSTIFICATION FOR DETERMINATION OF FAILURE SOURCE		NATURE OF RADIATION EFFECT			REMARKS	TEST AND DATE
	GAMMA DOSE RATE	GAMMA DOSE	FAST NEUTRON FLUX	FAST NEUTRON FLUENCE	OTHER	TEST DEMONSTRATED	ANALYTICAL ENGINEERING JUDGMENT	PERMANENT	TRANSIENT	SYNERGISM		
GTR-4 5/63	NRE										One Winsco unit circuit opened after unplanned test detonation. Second Winsco unit failed during irradiation. Satisfactory operation of Engelhard units.	GTR-4 5/63
GTR-7A 8/63	NRE										Post irradiation resistance values within 1% of pre-irradiation values.	GTR-7A 8/63
PDPF-4 9/63	X					X		X	X	X	Good data to 1.44 (B) RADS (C) HR after which excessive internal heat generation rates produced excessive error. Insulation resistance of one RTI increased with reactor power, the other decreased with power. One RTI output increased steadily at full reactor power. Insulation resistance > 1.0 Megohm recommended.	PDPF-4 9/63
GTR-9 10/63	X			X		X		X			Post irradiation values within 2% of pre-irradiation resistance values for 6 RTI. One micro circuit opened prior to test start. Rate sensitive decrease in one Winsco unit insulation resistance decreased less than one decade at full power. Rate sensitive decrease in 3 Microdot units, insulation resistance increased 2-3 decades at full power. All operating transducers showed temperatures greater than LN ₂ because of gamma heating. Comparison made with thermocouples.	GTR-9 10/63
GTR-12A 3/64	X			X		X	X	X	X	X	Fast neutron temperature deviations 2 to 6°R for 8 units. Gamma heating average temperature change + .6°R. Two Winsco units and one Rosemount unit failed during irradiation. Excessive expansion of connector potting compound due to gamma heating caused open circuits. 55% recovery of effects due to fast neutrons during annealing to room temperature. Rosemount 134EB performed most satisfactory.	GTR-12A 3/64
GTR-14 7/64	X		X	X		X	X	X			These units previously irradiated in GTF-12A. For 110AA, fast neutron temperature deviation = 1.0°R and gamma temperature deviation = .15°R. For 134EB's, fast neutron average temperature deviation = .2°R, gamma = .15°R.	GTR-14 7/64
NRX-A2 9/64				NRE							Post irradiation permanent change in absolute accuracy of .2°R at LH ₂ temperature.	NRX-A2 9/64
GTR-15 12/65				X		X	X	X			Average temperature deviations: Rosemount 134GT, + 3.2°R; Rosemount 134EB250, + 2.6°R; Rosemount 134FB1X, + 3.5°R; Rosemount 134EB2X, + 3.2°R; Rosemount 110AA150, 9.0°R; Winsco 2448-S, + 6°R; Rosemount 134GT + .46; Winsco 134FB250, .54; Rosemount 134EV1X, .63; Winsco 134EB2X, .65; Rosemount 110AA150, .54; Winsco 2448-S, .51	GTR-15 12/65
NRX-A3 5/65				X				NP			A small positive output increase during full power run which was predictable. Fast neutron degradation of platinum resulted in ± 8.0°R data system temperature uncertainty.	NRX-A3 5/65
NRX-EST 3/66					NRE						Post test disassembly revealed broken internal lead, on 5 PIT's and faulty connectors on 2 others. Radiation believed not to be responsible for failures. All RTI's located external to pressure vessel.	NRX-EST 3/66
NRX-A5 5/66						NRE					XDCR mounted on nozzle inlet manifold. This unit replaced one failing due to vibration during preirradiation checkout.	NRX-A5 5/66
GTR-18 6/66	X			X		X		X	X	X	9 units shielded with various materials. 2 units not shielded. Average temperature deviation 2.35°R to 5 x 10 ¹⁶ n/cm ² . Average temperature deviation 2.74°R to 5 x 10 ¹⁶ n/cm ² . Gamma heating temperature change .33°R. Temperature deviation 3.02°R to 5 x 10 ¹⁶ n/cm ² . Gamma heating temperature change .17°R. Average temperature deviation 2.51°R to 5 x 10 ¹⁶ n/cm ² . Gamma heating temperature change .35°R. Temperature deviation 4.8°R to 5 x 10 ¹⁶ n/cm ² . Gamma heating temperature change .10°R.	GTR-18 6/66
NRX-A6 12/67				X		X	X	X	X	X	Gradual 5 to 10°R increase in indicated temperature output during 60 minute full power hold above 5 x 10 ¹⁴ n/cm ² . Change in post irradiation resistance difficult to evaluate because of annealing effects at higher temperatures.	NRX-A6 12/67
NRX-A6 12/67	X			X			X	X			Average temperature deviation of + 1.72°R due to fast neutron fluence. Gamma heating temperature change + .55°R. The RTTs were mounted on the nozzle manifold. Redesigned XDCR moved connector closer to sensing element.	NRX-A6 12/67

Figure 11.1-1 Reactor Temperature Measurement Locations

11-11

Figure 11.1-2 NRX-A6 EP-III A Reflector Inlet Plenum RTT's Reflector P.V. Annulus 936B363H01

Figure 11.1-3 Platinum Sensing Element RTT Maximum Radiation Test Levels In Terms of Percent of the Expected NERVA Radiation Levels

11.1.2 REFERENCES

1. WANL Drawing No. 979D161C - Thermometer, Resistance - Instrumentation (August 1966).
2. E. K. Figenbaum, WANL DRM 52033 - NSS Transducer Installation Study - Reflector Low Temperature Measurements (September 1969).
3. M. A. Capo et al. S-100 Radiation and Shielding Analysis Report NERVA Nuclear Subsystem. WANL TME 2757 (To be published)
4. WANL Drawing 939J740, NERVA Nuclear Subsystem Reference Design Layout.
5. EC-677559, Detail Specification, Part I, Performance/Design and Qualification Requirements for Reflector Assembly Forming a Part of CP-677555, NERVA Nuclear Subsystem (July 1970).
6. A. C. Thadani. Temperature Distributions Within Non-Rotating Vehicle. WANL DRM 52943 (August 1970).
7. EC-677565, Detail Specification, Part I - Performance/Design and Qualification Requirements for Nuclear Subsystem Instrumentation Forming a Part of NERVA Nuclear Subsystem (July 1970).
8. EC-677565, Equipment Specification, Thermometer, Resistance (October 1967).
9. E. Figenbaum, WANL DRM 51912 - Instrumentation Comparison Matrix Study, Low Temperature Measurements (February 1970).
10. REIC Report Number 43, TIC Report Number 3 - The Effect of Nuclear Radiation on Transducers (October 1966).
11. ER-8295, Components Irradiation Test No. 17 - "Resistance Thermometers at Liquid Hydrogen Temperatures," Lockheed Aircraft Corporation, Lockheed, Georgia Company, Georgia Nuclear Labs (November 1965). NAS-85332.
12. Magee, R. M., et al, BSR-1057, "Design Criteria for Radiation Resistant Transducers in Cryogenic Environments," Bendix Corporation, Bendix Systems Division (August 1965).
13. Bush, N. E., - WANL DRM 51911 - Survey of Radiation Effects Testing History of NERVA Related Transducer Types (August 1970).

REFERENCES (CONTINUED)

14. Tallman, C. R., Performance Characteristics of Resistance Thermometers as Determined by the Defects Introduced in the Sensor During Fabrication and Use, and by Adverse Environments (Unpublished paper).
15. ANSC-RN-S-0157, Final Test Report RTT Tests 23/L101 (December 1964).
16. ANSC-RN-S-0251, Summary Report Resistance Temperature Transducer Irradiation Program (December 1965).
17. ANSC-RN-S-0370, Fast and Thermal Neutron and Proton Effects on Platinum Resistance Transducers at Liquid Hydrogen Temperature, (June 1967).
18. WANL-TME-1863, NRX-A6 Transducer Evaluation Report,CRD (December 1968).
19. WANL-TNR-145, Test Report on a WANL Radiation Test of NERVA Components (GTR Test Number 4), CRD (November 1963).
20. GD/FW FZK 170-4, NERVA Components, Irradiation Program, Volume 4 - GTR Test 7A (November 1963).
21. WANL-TME-700, Test Report on a WANL Radiation Test of NERVA Components (GTR Tests 7 and 7A), CRD (March 1964).
22. WANL-TME-691, NRX-A Candidate Diagnostic Transducers Irradiated at Plum Brook Reactor Facility, Cycle 4 (DM 1W) (April 1964).
23. WANL-TME-786, Test Report on a WANL Radiation Test of NERVA Components (GTR Test 9), CRD (May 1964).
24. ANSC-RN-S-0121, Preliminary Test Reports for Ground Test Reactor (GTR) Test No. 12A (August 1964).
25. GD/FW FZK-184-4, NERVA Components Irradiation Program, Volume 4, GTR Test 14 (November 1964).
26. ANSC-RN-S-0318, Summary of Radiation Effects Information Gained from NRX-A2 and NRX-A3 (September 1966).
27. GD/FW FZK-184-6, NERVA Components Irradiation Program, Volume 6, GTR Test 15 (March 1965).
28. ANSC-RN-S-0396, Summary of Radiation Effects Information Obtained from NRX/EST (July 1967).

REFERENCES (CONTINUED)

29. ANSC-RN-S-0397, Summary of Radiation Effects Information Gained from NRX-A5 (August 1967).
30. GD/FW FZK-310-1, NERVA Irradiation Program, Volume 1, GTR Test 18 Components Tests (August 1966).
31. ANSC-RN-S-0505, Summary of the Radiation Effects Information Gained from NRX-A6 (July 1969).
32. ANSC-RN-S-0468, Post Test Analysis of NRX A6 Evaluation RTTs Rosemount Model No. 134JA-1 (May 1968).
33. ANSC-S110, SR01-W181, NERVA Radiation Effects Summary Report, NERVA Program (May 1970).

RADIATION EFFECTS INFORMATION REQUESTS (REIRs)

(Chapter 11.1 - RTT)

<u>No.</u>	<u>Description</u>
20	Platinum Resistance Wire
21	Platinum Wire Wound Mandrel
22	RTT Insulation

IDENTIFICATION NO. 20

RADIATION EFFECTS INFORMATION REQUEST

DATE November 1970

SUBJECT Platinum Resistance Wire Evaluation FUNCTIONAL CATEGORY Engr. Test & Evaluation

REF. ITEM SUPPORT EC-677565 REF. DATA ITEMS S-110 (Chapter 11.1)

INFORMATION DESIRED The effects of gamma and fast neutron radiation and temperature cycling on platinum wire resistivity
(PURPOSE OF STUDY OR EXPERIMENT)

DATE REQUIRED _____

RELATION OF ITEM TO ENGINE Reflector inlet plenum
(LOCATION)

PER DRAWINGS, DATA ITEMS (EC-677559, DRM 52033)

ENVIRONMENT (4): Fast NEUTRON FLUX $5.1 \times 10^{13} \text{ n/cm}^2\text{-sec}$ NEUTRON FLUENCE $1.0 \times 10^{18} \text{ n/cm}^2$
(Ref. data items) E > 1.0 Mev Fast E > 1.0 Mev

GAMMA DOSE RATE $4.4 \times 10^{18} \text{ RADS(C)/HR}$ GAMMA DOSE $4.4 \times 10^9 \text{ RADS(C)}$

TEMPERATURE 192 NUCLEAR HEATING RATE 1.23 W/gm(C)

PRESSURE N/A OTHER N/A

FLUID LN₂ GN₂

CYCLE REQUIREMENTS 450-192-700°R

RELIABILITY ESTIMATE 0.9999 probability of success REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL N/A APPROVED _____ DATE _____

APPROVED _____ DATE _____

SPECIAL CONSIDERATIONS N/A APPROVED _____ DATE _____

PERTINENT REFERENCES N/A

11-17

IDENTIFICATION NO. 21

RADIATION EFFECTS INFORMATION REQUEST

DATE November 1970

SUBJECT RTT Platinum Resistance Wire Wound Mandrel Evaluation FUNCTIONAL CATEGORY Engr. Test & Evaluation

REF. ITEM SUPPORT EC-677565 REF. DATA ITEMS S-110 (Chapter 11.1)

INFORMATION DESIRED The effects of gamma and fast neutron radiation and temperature cycling on the electrical
(PURPOSE OF STUDY OR EXPERIMENT) resistance change of the insulation between the platinum wires wound on the

RTT mandrel

DATE REQUIRED _____

RELATION OF ITEM TO ENGINE Reflector inlet plenum
(LOCATION)

PER DRAWINGS, DATA ITEMS (EC-677559, DRM 52033)

ENVIRONMENT (4): Fast NEUTRON FLUX $5.1 \times 10^{13} \text{ n/cm}^2 \text{ - sec}$ NEUTRON FLUENCE $1.9 \times 10^{18} \text{ n/cm}^2 \text{ E} > 1.0 \text{ Mev}$
(Ref. data items) $E > 1.0 \text{ Mev Fast}$

GAMMA DOSE RATE $4.4 \times 10^8 \text{ RADS(C) / HR}$ GAMMA DOSE $4.4 \times 10^9 \text{ RADS(C)}$

TEMPERATURE 192 NUCLEAR HEATING RATE 1.23 W/gm (C)

PRESSURE N/A OTHER N/A

FLUID LN₂, GN₂

CYCLE REQUIREMENTS 450-192-700° R

RELIABILITY ESTIMATE 0.99990 probability of success REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL N/A APPROVED _____ DATE _____

APPROVED _____ DATE _____

SPECIAL CONSIDERATIONS N/A APPROVED _____ DATE _____

PERTINENT REFERENCES N/A

IDENTIFICATION NO. 22

RADIATION EFFECTS INFORMATION REQUEST

DATE November 1970

SUBJECT Platinum Resistance Wire RTT Evaluation FUNCTIONAL CATEGORY Engr. Test & Evaluation

REF. ITEM SUPPORT EC-677565 REF. DATA ITEMS S-110 (Chapter 11.1)

INFORMATION DESIRED The effects of gamma and fast neutron radiation and temperature cycling on the electrical
(PURPOSE OF STUDY OR EXPERIMENT) resistance change of the RTT insulation between the copper sheathed cable and RTT
case ground

DATE REQUIRED _____

RELATION OF ITEM TO ENGINE Reflector Inlet Plenum
(LOCATION)

PER DRAWINGS, DATA ITEMS (EC-677559, DRM 52033)

ENVIRONMENT (4): Fast NEUTRON FLUX $5.1 \times 10^{13} \text{ n/cm}^2 \text{ sec}$ NEUTRON FLUENCE $1.9 \times 10^{18} \text{ n/cm}^2$, $E > 1.0 \text{ Mev}$
(Ref. data items) Fast

GAMMA DOSE RATE $4.4 \times 10^8 \text{ RADS(C)/HR}$ GAMMA DOSE $4.4 \times 10^9 \text{ RADS(C)}$

TEMPERATURE 192 NUCLEAR HEATING RATE 1.23 W/gm(C)

PRESSURE N/A OTHER N/A

FLUID LN_2 , GN_2

CYCLE REQUIREMENTS $450-192-700^\circ \text{R}$

RELIABILITY ESTIMATE 0.99990 probability of success REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL N/A APPROVED _____ DATE _____

APPROVED _____ DATE _____

SPECIAL CONSIDERATIONS N/A APPROVED _____ DATE _____

PERTINENT REFERENCES N/A

11-19

REPORTS (T-119s) RESPONDING TO REIRs

(None prepared at this issuance)

11.2 LOW TEMPERATURE THERMOCOUPLES

11.2.1 SUMMARY AND CONCLUSIONS

The low temperature copper-constantan thermocouples (CC-T/C) considered for NERVA NSS use appear to be resistant to the radiation levels experienced to date in NERVA and non-NERVA technology testing. Previous testing reveals that the best overall system accuracy is maintained above 300°R.

None of the thermocouples considered for NERVA use experienced significant "failure". Since satisfactory performance is expected for the NERVA application, further radiation development testing for CC-T/Cs does not appear necessary.

11.2.2 INTRODUCTION

Copper/constantan sensing tip thermocouples (CC-T/C) have been selected for NERVA NSS gas temperature measurements at the component inlet plenums and for measuring material temperatures in the temperature range 35-1200°R.

The transducers must provide an accurate determination of hydrogen temperature and material temperatures to evaluate reactor components thermal/hydraulic/mechanical performance and to compute reactor thermal power.

11.2.2.1 Transducer Description

When two junctions (each composed of the same two dissimilar metals) are subjected to different temperatures, a voltage is generated. One, or both, of the two dissimilar metal junctions is always at some reference temperature. The voltage difference (measured with a high impedance device) between the two junctions, is proportional to the temperature difference between the two junctions.

Several CC-T/C designs are currently considered for NERVA NSS use:⁽¹⁾

- (a) Grounded junction with sheath diameter of 0.020"⁽²⁾
(WANL Sketch SK-I-391)
- (b) Grounded junction with sheath diameter of 0.045"⁽³⁾
(WANL Drawing 964B039)
- (c) Exposed junction with sheath diameter of 0.065"⁽⁴⁾
(WANL Drawing 945C906H01)

Each thermocouple design includes MgO insulation and 348 stainless steel sheaths.

The type of junction (exposed or grounded) for any particular NSS-low temperature measurement depends on the sensor frequency response, system accuracy, and routing configuration requirements.⁽¹⁾

11.2.2.2 Location of Transducers

WANL-DRM-51912 lists the copper/constantan thermocouples selected for use at the various NSS locations (See Figure 11.1-1). The WANL DRMs listed in References 5 to 8 describe the recommended installation at each location and the intended reactor and/or flight application.

<u>Installation DRM</u>		<u>Reactor Application</u>
	1. Grounded junction, 0.02" diameter sheath (SK-I-391)	
52036	a. Support stem supply plenum gas temperature	Ground Test
51903	b. Central shield inlet gas temperature	Ground Test & Flight
52033	c. Reflector cylinder (materials) temperature	Ground Test & Flight
52033	d. Reflector tie bolt material temperature	Ground Test
52033	e. Control drum annulus gas temperature	Ground Test
52033	f. Drum vane material temperature	Ground Test
52036	g. Support stem outlet plenum (individual) gas temperature	Ground Test
52033	h. Reflector coolant channel outlet gas temperature	Ground Test
52036	i. Support stem exhaust plenum gas temperature	Ground Test & Flight
52033	j. Drum vane exit gas and drum annulus exit	Ground Test
52021	k. Support stem coolant-aft gas temperature	Ground Test
52033	l. Reflector/pressure vessel annulus exit	Ground Test

	2. Grounded junction, 0.045" diameter sheath (964B039)	
52036	a. Core support plate material temperature	Ground Test
52033	b. Control drum material temperature	Ground Test
51903	c. Central shield material temperature	Ground Test & Flight
	3. Exposed junction, 0.065" diameter sheath (945C906H01)	
52036	a. Core inlet plenum gas temperature	Ground Test & Flight
	4. Grounded junction, 0.020" diameter at sensing end and routing cable of 0.04" diameter. Design similar to 978D186	
52031	a. Support item and cap material	Ground Test
52021	b. Support item outlet gas	Ground Test

11.2.3 ENVIRONMENTAL AND OPERATIONAL REQUIREMENTS

The thermocouples must be capable of withstanding the nuclear environment of Paragraph 11.2.3.1 and the non-nuclear environment of Paragraph 11.2.3.2. In addition, the transducers must comply with the performance characteristics designated in Paragraph 11.2.3.3.

11.2.3.1 Nuclear Environment

Listed below are the approximate radiation levels anticipated at the core inlet plenum centerline during rated reactor power.⁽⁹⁾

Neutron flux

E > 1.0 Mev	$7.0 \times 10^{14} \text{ n/cm}^2 - \text{sec}$
.4 ev < E < 1.0 Mev	$4.5 \times 10^{15} \text{ n/cm}^2 - \text{sec}$
E < .4 ev	$1.0 \times 10^{14} \text{ n/cm}^2 - \text{sec}$

Neutron fluence

E > 1.0 Mev	$2.5 \times 10^{19} \text{ n/cm}^2$
.4 ev < E < 1.0 Mev	$1.6 \times 10^{20} \text{ n/cm}^2$
E < .4 ev	$3.6 \times 10^{18} \text{ n/cm}^2$

Gamma dose rate

$$5.0 \times 10^9 \text{ RADS(C)/HR}$$

Total gamma dose

$$5.0 \times 10^{10} \text{ RADS(C)}$$

Heating rate

$$13.9 \text{ W/gm(C)}$$

11.2.3.2 Non-Nuclear Environment

At this time the NERVA engine design is preliminary.⁽¹⁰⁾ The NSS hydrogen temperatures and pressures will depend on the final engine reference design and the engine mode of operation. The most severe radiation environment on copper/constantan thermocouples will probably occur at the core inlet plenum.

The present start of life, composite core, inlet plenum, rated engine power state points are: ^(11, 12)

Hydrogen mass velocity	TBD lb/sec/in ² flow area
Temperature	267°R
Pressure	623 psia
Cycles	as defined in reference 12

500°R is the maximum core inlet fluid temperature expected during cooldown operations after a 10-minute full power hold.⁽¹³⁾ Calculations for longer run times have not been performed (for which cases the temperatures could be as high as 600 to 700°R). At the present time the maximum known temperature the thermocouples will experience is earth ambient.

11.2.3.3 Transducer Characteristics

The transducer and integral sheath cable must measure core inlet temperature to an accuracy of $\pm 5^\circ\text{R}$ and must be capable of operation through the temperature range(s) designated in WANL EC-677565.⁽¹⁴⁾

Equipment Specification, 675503,⁽¹⁵⁾ describes the temperature - EMF requirements and the calibration requirements for copper/constantan thermocouples.

11.2.4 ANALYSIS

11.2.4.1 Thermocouple Assembly

11.2.4.1.1 Non-NERVA Technology

Little information is available in non-NERVA technology for copper/constantan thermocouples tested at the high radiation levels expected during NERVA operation.

Blewitt and Jamison reported⁽¹⁶⁾ in 1953 that CC-T/C's irradiated to a fast neutron fluence of 5×10^{16} n/cm² (energy level not defined) at LN₂ temperature did not show any apparent radiation effects.

Battelle Northwest Laboratory reported⁽¹⁷⁾ in 1967 that preliminary analysis at the Fast Flux Test Facility (FFTF) indicated a gamma dose rate of 2.3×10^7 RADS(C)/HR caused a calibration shift in copper-constantan thermocouples. Test results indicated the magnitude of the calibration shift depended on temperature, thermocouple sheath size, and dose rate. The transducers were tested between 140°R and 670°R.

11.2.4.1.2 NERVA Technology

Table 11.2-1 condenses pertinent information from previous NERVA technology program copper/constantan thermocouple radiation testing. (18 thru 25, 27 thru 48).

The testing consisted of two types:

- (1) Transducer diagnostic
- (2) Reactor diagnostic

The transducer diagnostic tests were planned to determine what effect radiation had on the transducer electrical or mechanical output. The reactor diagnostic test primary objective was to use the transducer to control and evaluate performance of the reactor. Evaluation of the transducer performance was usually a secondary objective of the reactor diagnostic test.

Examination of Table 11.2-1 reveals that all but one of the GTR, PBRF and NRX (external to pressure vessel) CC thermocouples tested performed satisfactorily and did not experience temperature deviations or other anomalies due to radiation.

During the NRX-A2, A3, and EST tests some ANSC thermocouples indicated significant temperature deviations. Data analysis revealed that the test car mounted reference temperature electronic controllers were damaged by gamma radiation and were the contributing factor to the anomalous output. The thermocouples were considered to have performed satisfactorily. (18 thru 22)

ANSC thermocouples tested during GTR-14 experienced small temperature deviations due to the gamma dose rate.⁽²³⁾ The temperature deviations were not of sufficient magnitude (less than 1°R) to warrant consideration as a thermocouple failure.

During GTR-15,⁽²⁴⁾ the total temperature deviation for the thermocouple tested was from $+3.0^{\circ}\text{R}$ to -5.5°R and was attributed to gamma dose rate and fast neutron fluence ($E > 1.0 \text{ Mev}$). Gamma heating in the thermocouples caused initial temperature deviations of about $+2.0^{\circ}\text{R}$. As testing progressed the temperature deviation decreased linearly to -5.5°R and appeared to be a function of the integrated fast neutron flux for the 300 MW-HR test.

359 WANL copper/constantan thermocouples were utilized for measuring NRX-A2 through A6 reactor gas and material temperatures. Sixteen of these T/Cs experienced data anomalies or did not operate properly because of improper installation. The T/Cs experiencing data anomalies showed a normal post test electrical check and/or no visually observed damage. Seven NRX-A6 units measuring reflector cylinder material and reflector bolt material temperatures indicated data drifts of 15 to 35°R . The NRX-A6 test document⁽²⁵⁾ reported that the temperature rise may have been due to neutrons causing a change in the thermal conductivity of the reflector material.

Tests at the Oak Ridge National Laboratory (ORNL) and Ground Test Reactor (GTR) have shown that a decrease in the thermal conductivity of beryllium occurs with increasing fast neutron fluence at cryogenic temperatures. A decrease in beryllium thermal conductivity is known to be responsible for a corresponding increase in material temperature⁽²⁶⁾.

XE post test instrumentation disassembly and post test data analysis were not performed. The thermocouples appeared to operate satisfactorily although specific performance was not reported.⁽²⁷⁾

Two of the thermocouple designs utilized for the NRX reactor test series and XE reactor testing will be used for NERVA testing (11.2.2). These thermocouples (945C906, 964B039)

will not be used for NERVA reflector beryllium temperature measurements. Thermocouple SK-1-391 (untested design) is the current candidate for measuring reflector material temperatures.

11.2.4.1.3 Conclusion

Copper/constantan thermocouples appear to be quite resistant to the radiation levels experienced in NERVA and non-NERVA technology testing.

As indicated in Table 11.2-1 (GTR-15), gamma dose rate and fast neutron fluence are most likely to be the cause of failure, if failure occurs.

Copper/constantan thermocouples have an extremely low sensitivity (3 microvolts per degree) in the LH_2 temperature region. Coupled with poor system accuracy and associated "noise", it is difficult to obtain data less than 50°R in the LH_2 temperature region. At temperatures between 50°R and 300°R , the data system is inherently less accurate than in the range 300°R to 1200°R . At temperatures between 300°R and 1200°R , the overall accuracy is acceptable,⁽²⁸⁾ as illustrated in NRX-A3 and NRX-EST tests. (See remarks in Table 11.2-1.)

Further radiation development testing of the 945C906 and 964B039 copper-constantan thermocouples does not appear necessary for the following reasons:

1. Except for the reflector beryllium temperature measurements (NRX-A6), these thermocouples performed satisfactorily throughout the NRX reactor test series.
2. Reflector beryllium temperature measurements will be made with another thermocouple design (WANL-SK-1-391).

The major component materials in the four NERVA thermocouple designs are the same. Since geometric and sheath closure procedures are the major differences in the thermocouple designs, and the behavior of beryllium in a radiation environment is understood, testing of the SK-1-391 thermocouple, or the thermocouple similar to the 978D186 design, does not appear necessary.

TABLE 11.2-1

TRANSDUCER TEST SURVEY, NERVA TECHNOLOGY PROGRAM

TEST AND DATE	DOCUMENT REFERENCES	TRANSDUCER MANUFACTURER AND MODEL NO. OR DRAWING NO.	SUCCESS RATIO	TRANSDUCER ENVIRONMENT							REMARKS
				GAMMA DOSE RATE RADS(C)/HR	TOTAL GAMMA DOSE RADS(C)	THERMAL NEUTRON FLUX $E < 1.86 \text{ eV}$ n/cm ² -sec	THERMAL NEUTRON FLUENCE $E < 1.86 \text{ eV}$ n/cm ²	FAST NEUTRON FLUX $E > 1.0 \text{ MeV}$ n/cm ² -sec	FAST NEUTRON FLUENCE $E > 1.0 \text{ MeV}$ n/cm ²	°R TEMPERATURE	
GTR 7A 8/63	FZK 170-4 TME-700	a. Tuttle-Kift b. Pyroelectric	28/28	4.5 (6)	4.5 (8)	NP*	NP*	7.8 (10)	2.8 (16)	140	Diallyl-Phthalat connectors withstood radiation better than alumina ceramic connectors. NRE**
PBRF-4 9/63	TME-691	Pyroelectric	5/5	5.3 (8)	3.2 (10)	NP*	NP*	9.4 (12)	2.0 (18)	572	Two T/Cs had grounded junctions. Three T/Cs had exposed junctions. Test results inconclusive as to whether or not gamma heating caused small temperature deviations. No obvious radiation effect.
GTR-14 7/64	FZK-184-4	ANSC	4/4	NP*	8.4 (8)	NP*	NP*	NP*	1.0 (16)	37	Maximum temperature during irradiation $\pm 0.7^\circ\text{R}$. A small initial temperature deviation due to gamma dose rate. Maximum temperature deviation at post test ice point calibration $+0.3^\circ\text{R}$ after annealing to 560°R . Temperature deviation too small to be considered a failure. NRE**
NRX-A2	TME 629	WANL 945C906	25/23	1.8 (9)	NP*	NP*	NP*	1.8 (14)	NP*	NP*	Data off scale during power runs. Post test examination and electrical check disclosed no damage. NRE**
	TME 1079 TME 840 WRL	WANL 9648039 WANL 9638363	8/7 30/26	2.9 (9) 4.7 (9)	NP* NP*	NP* NP*	NP* NP*	1.8 (14) 4.7 (14)	NP* NP*	NP* NP*	Test data off scale. Visual and electrical checks normal. No indication of anomaly. NRE** Erratic data. Post test electrical checks indicated sensors good. Anomalies not resolved. NRE**
NRX-A2 9/64	R-NS-0318 R-NS-0193	Con. Controls Corp. Tematron Division ANSC Nos. a. 701530-701554 b. 706614-706615	30/30	NP*	NP*	NP*	NP*	NP*	1.0 (16)	37-650	Six immersion T/C data indicated temperatures to be lower than LH_2 and is attributable to computer programming and signal conditioning errors. Twenty-four surface mounted T/C pre and post calibration data indicated 40°R shift at 35°R shift at 662°R . Indicated temperature shifts due to radiation effects on reference temperature electronic controllers mounted in privy.
GTR-15 12/64	FZK-184-6	Con. Controls Corp. Temptron Division T-2951-1	1/0	NP*	3.7 (9)	NP*	NP*	NP*	4.5 (16)	37	Temperature deviation during irradiation $+3.0$ to -5.7°R . Initial temperature deviation of $+2.0^\circ\text{R}$ due to gamma heating. An LH_2 trailer change appeared to allow radiation damage to anneal out since temperature deviation decreased. Temperature deviation after a second reactor startup (because of an inadvertent shutdown) was -5.5°R ; increased to $+3.0^\circ\text{R}$, and decreased linearly to -3.0°R during irradiation.
NRX-A3 5/65	TME 1245	WANL 945C906	35/31	NP*	NP*	NP*	NP*	NP*	NP*	NP*	Erratic test data. One unit post resistance measurement change $+16\%$. No visual examination. NRE**
	WRL	WANL 9648039 WANL 9638363	10/10 47/44	NP* NP*	NP* NP*	NP* NP*	NP* NP*	NP* NP*	NP* NP*	NP*	NRE** Erratic test data on two sensors with normal post test electrical check. 4.7% resistance change on other sensor. No visible damage. NRE**
NRX-A3 5/65	R-NS-0318 R-NS-0307 R-NS-0238	NP*	49/49	2.0 (9)	NP*	NP*	NP*	2.0 (14)	NP*	37-1200	Immersion T/Cs experienced $-12^\circ\text{R}/\text{sec}$. Indicated temperature drift due to gamma radiation effects on reference temperature junction. Measurement uncertainties least between 400 and 1200°R . Only a small change in loop resistance due to irradiation. NRE**

TABLE 11.2-1 (CONTINUED)
TRANSDUCER TEST SURVEY, NERVA TECHNOLOGY PROGRAM

TEST AND DATE	DOCUMENT REFERENCES	TRANSDUCER MANUFACTURER AND MODEL NO. OR DRAWING NO.	SUCCESS RATIO	TRANSDUCER ENVIRONMENT							REMARKS
				GAMMA DOSE RATE RADS(C)/HR	TOTAL GAMMA DOSE RADS (C)	THERMAL NEUTRON FLUX $E < 1.86 \text{ ev}$ $n/cm^2\text{-sec}$	THERMAL NEUTRON FLUENCE $E < 1.86 \text{ ev}$ n/cm^2	FAST NEUTRON FLUX $E > 1.0 \text{ MEV}$ $n/cm^2\text{-sec}$	FAST NEUTRON FLUENCE $E > 1.0 \text{ MEV}$ n/cm^2	OR TEMPERATURE	
NRX-EST 3/66	TME 1210	WANL 945C906	54/53	5.4 (9)	NP*	2.9 (14)	NP*	2.0 (15)	NP*	NP*	Reversed polarity on one unit (improper installation) NRE** NRE** NRE**
	TME 1447	WANL 964B039	10/10	3.2 (9)	NP*	1.0 (15)	NP*	2.2 (14)	NP*		
	MRL	WANL 963B363	24/24	4.3 (9)	NP*	1.3 (15)	NP*	2.9 (14)	NP*		
NRX-EST 3/66	R-NS-0396	NP*	43/43	NP*	2.0 (9)	NP*	3.7 (17)	NP*	3.4 (17)	35-1000	No apparent radiation effects on surface and immersion T/Cs but some drift occurred due to radiation effects on solid state electronic controllers. Measurement uncertainties least between 300 and 800°R. NRE**
NRX-A5 5/66	TME 1389	WANL 945C906	44/43	5.0 (9)	NP*	2.3 (14)	NP*	6.5 (14)	NP*	NP*	Bent probe on one unit. NRE** NRE** NRE**
	TME 1509	WANL 964B039	6/6	3.5 (9)	NP*	1.0 (15)	NP*	2.1 (14)	NP*		
	MRL	WANL 963B363	8/8	4.5 (9)	NP*	1.2 (15)	NP*	3.0 (14)	NP*		
NRX-A5 5/66	R-NS-0397	ANSC 1115422-39	6/6	NP*	1.0 (9)	NP*	NP*	NP*	1.0 (17)	35-600	Failure of one unit not due to radiation effects. Thermocouple reference temperature electronic controller experienced radiation effects which required T/C data correction.
	R-NS-0330	ANSC 1115419-39	22/22								
NRX-A6 12/67	TME-1863	WANL 945C906	23/23	6.7 (9)	6.7 (9)	2.0 (14)	7.2 (17)	6.5 (14)	2.3 (18)	170 to 300	NRE** Three reflector cylinder units and four reflector bolt units had 15-30°R temperature rise which, as reported by TME 1863, may be due to neutrons causing a change in the thermal conductance of the reflector material at cryogenic temperature. NRE**
	MRL	WANL 964B039	11/4	3.9 (9)	3.9 (9)	2.2 (15)	7.9 (18)	2.3 (14)	8.3 (17)	200	
		WANL 963B363	24/24	3.0 (9)	3.0 (9)	1.1 (15)	4.0 (18)	2.7 (14)	9.7 (17)	130 to 445	
XE	TME 1805	WANL 945C906	17/?	6.2 (9)	NP*	1.0 (14)	NP*	8.0 (14)	NP*	200	Success ratio not reported. Invalid data for one unit. Reason not reported. NRE** Success ratio not reported
	R-NS-0510	WANL 964B039	4/3	3.8 (9)	NP*	1.0 (15)	NP*	2.2 (14)	NP*	220	
	MRL	WANL 963B363	14/?	4.5 (9)	NP*	1.5 (15)	NP*	3.5 (14)	NP*		
NERVA		WANL 945C906		5.9 (9)	5.0 (10)	1.0 (4)	3.6 (18)	7.0 (14)	2.5 (19)	267	
		WANL 964B039		4.1 (9)	4.1 (10)	5.1 (14)	1.8 (19)	7.1 (14)	2.6 (19)	225	
		WANL SK-I-391		4.1 (9)	4.1 (10)	5.1 (14)	1.8 (19)	7.1 (14)	2.6 (19)	250	

*NP - not presented in referenced documents.

**NRE - no significant or apparent radiation effects.

Figure 11.2-1. Reactor Temperature Measurement Locations

REFERENCES

(Chapter 11.2- Cu/Con T/C)

- (1) E. K. Figenbaum - "Instrumentation Comparison Matrix Study, Low Temperature Measurements," WANL DRM 51912 (February 1970)
- (2) Thermocouple - 0.020 Inch Diameter Copper-Constantan, Grounded Junction, WANL Sketch SK-1-391
- (3) Thermocouple - Copper-Constantan Sheathed; Instrumentation, WANL Dwg. 964B039
- (4) Thermocouple - Exposed Junction; Instrumentation, WANL Dwg. 945C906
- (5) E. K. Figenbaum - "NSS Transducer Installation Study, Reflector Low Temperature Measurements," WANL DRM 51933 (September 1969)
- (6) E. K. Figenbaum - "NSS Transducer Installation Study, Core Support Plate and Plena Low Temperature Measurements," WANL DRM 52036 (September 1969)
- (7) E. K. Figenbaum - "NSS Transducer Installation Study, Cluster Support Stem Low Temperature Measurement," WANL DRM 52021 (August 1969)
- (8) E. K. Figenbaum - "NSS Transducer Installation Study, Central Shield and Cluster Plate Low Temperature Measurements," WANL DRM 51903 (October 1969)
- (9) M. A. Capo, et al, "S-100 Radiation and Shielding Analysis Report, NERVA Nuclear Subsystem," WANL TME-2757 (to be published).
- (10) NERVA Nuclear Subsystem Reference Design Layout, WANL Drawing 939J740
- (11) NSS State Points - Flight Engine Composite Core, WANL Drawing 957A982
- (12) Detail Specification, Part 1 - Performance/Design and Qualification Requirements for Cluster Hardware Forming a Part of CP-677555, NERVA Nuclear Subsystem, EC-677558 (July 1970)
- (13) D. L. Black - Trade Study 757 - Reactor Cooldown Criteria. WANL-TME-2728 (CRD), November 1970.

REFERENCES (CONTINUED)

- (14) Detail Specification, Part 1 - Performance/Design and Qualification Requirement for Nuclear Reactor Subsystem Instrumentation Forming a Part of NERVA Nuclear Subsystem, EC-677565 (July 1970).
- (15) Equipment Specification, Thermocouple, Copper-Constantan, E-675503.
- (16) Behavior of Two Types of Thermocouples Under Pile Irradiation at Low Temperatures. Review of Scientific Instruments, Vol. 24, T. H. Blewitt, R. E. Jamison (1953).
- (17) Electromotive Force in Thermocouples, Fast Flux Test Facility Quarterly Progress Report, January, February, March, 1967, BNWL-501 Pacific Northwest Laboratory, Richland, Washington.
- (18) Summary of Radiation Effects Information Gained from NRX-A2 and NRX-A3, ANSC-RN-S-0318 (September 1966).
- (19) Instrumentation Evaluation, NRX-A2 Tests ANSC-RN-S-0193 (February 1965).
- (20) Instrumentation Evaluation, NRX-A3 Tests ANSC-RN-S-0238 (December 1965).
- (21) Summary of Radiation Effects Information Obtained from NRX-EST, ANSC-RN-S-0396 (July 1967).
- (22) NRX-EST Post Test Instrumentation Report, ANSC-RN-S-0307 (October 1966).
- (23) NERVA Components Irradiation Program, Vol. 4, GTR Test 14, GD/FW-FZK-184-4 (November 1964).
- (24) NERVA Components Irradiation Program, Vol. 6, GTR Test 15 (March 1965).
- (25) NRX-A6 Transducer Evaluation Report, WANL-TME-1863 (CRD) (December 1965).
- (26) Radiation Effects Analysis Report, Reflector Assembly, Chapter 8. S-110. J. A. DeMastry (To be published November 1970).
- (27) XE-2 Measurement Requirements List. (CRD) (June 1966).
- (28) NERVA Components Irradiation Program, Vol. 4, GTR Test 7A, GD/FW-FZK 170-4 (November 1963).
- (29) Test Report on a WANL Radiation Test of NERVA Components (GTR Tests 7 and 7A) (CRD) (March 1964).

REFERENCES (CONTINUED)

- (30) NRX-A Candidate Diagnostic Transducer Irradiated at Plum Brook Reactor Facility, Cycle 4 (DM-1W), WANL-TME-691 (April 1964)
- (31) Summary of Radiation Effects Information Gained from NRX-A5, ANSC-RN-S-0397 (August 1967)
- (32) NRX-A5 Post Test Instrumentation Report, ANSC-RN-S-0330 (January 1967)
- (33) Radiation Effects Program Review. WANL-TME-629 (CRD) (December 1963)
- (34) NRX-A2 Instrumentation Disassembly and Post Operative Report. WANL-TME-1079 (CRD) (January 1965)
- (35) Reactor Analysis Data Book, 3 Volumes. WANL-TME-840 (CRD) (1965)
- (36) NRX-A2 Measurements Requirements List (August 1964)
- (37) NRX-A3 Instrumentation Disassembly and Post Operative Report. WANL-TME-1245 (CRD) (August 1965)
- (38) NRX-A3 Measurements Requirements List. (CRD) (May 1965)
- (39) NRX-A4/EST Reactor Analysis Data. WANL-TME-1210. (CRD) (August 1965)
- (40) NRX/EST Disassembly and Post Operative Examination Report. WANL-TME-1447 (CRD) (June 1966)
- (41) NRX/EST Measurements Requirements List. (CRD) (March 1966)
- (42) NRX-A5 Reactor Analysis Data. WANL-TME-1389 (CRD) (April 1966)
- (43) NRX-A5 Sensor Disassembly and Post-Operative Examination Report. WANL-TME-1509 (CRD) (September 1966)
- (44) NRX-A5 Measurements Requirements List. (CRD) (August 1966)
- (45) NRX-A6 Measurements Requirements List. (CRD) (January 1968)
- (46) XE-2 Nuclear Subsystem Thermal and Nuclear Design Data Book. WANL-TME-1805 (CRD) (July 1968)
- (47) XE-Prime Engine Final Report. Vol. III Book 1, Test Data Analysis. ANSC-RN-S-0510 (May 1970)
- (48) R-1 Measurement Requirements List. (CRD) (February 1968)

11.3 MEDIUM TEMPERATURE THERMOCOUPLES

11.3.1 SUMMARY AND CONCLUSIONS

The medium temperature Chromel/Alumel thermocouples considered for NERVA use appear to be resistant to the radiation levels experienced to date in NERVA and non-NERVA technology testing. A possible exception to this is the 150°R temperature rise experienced by the core material temperature sensors during the NRX-A6 reactor test.

Several of the thermocouple designs considered for NERVA use have not experienced the higher radiation levels and low and high temperature extremes expected during NERVA reactor operations and post shutdown cooldown operations.

Additional testing is recommended to determine:

1. The cause of a temperature rise at the NRX-A6 core material temperature measurement location (Station 0.2).
2. If the thermocouples considered for the support stem liner temperature measurements will perform adequately at the high NERVA radiation levels and the temperature cycling expected.
3. If the new thermocouple design considered for the unfueled core element (Station 8.0) temperature measurement will perform satisfactorily in the NERVA environment.

The additional testing suggested is included in this report on Radiation Effects Information Request (REIR) sheets.

11.3.2 INTRODUCTION

Chromel/Alumel thermocouples (Cr/Al T/C) are being considered for various reactor gas and material temperature measurements in the temperature range 140°R to 2200°R.

The transducers must provide an accurate determination of hydrogen temperatures and material temperatures to aid in evaluating the thermal/hydraulic/mechanical performance of the NSS core, lateral support system, and the structural support assembly.

11.3.2.1 Transducer Description

Several Chromel/Alumel thermocouple designs are currently considered for NERVA NSS use. These include:

1. Grounded junction, MgO insulated with a 0.020 inch diameter 348 stainless steel sheath, WANL Drawing 978D186H01⁽¹⁾.

2. Grounded junction, MgO insulated with a 0.045 inch diameter 348 stainless steel sheath, WANL Drawing 963B361⁽²⁾.

3. Grounded junction, MgO insulated with a 0.020 inch or 0.045 inch diameter stainless steel sheath, modified WANL Drawing 978D186.

4. Grounded junction, MgO insulated, 8 inch long molybdenum sheath at the sensing end (sheath diameter not defined). No WANL drawing is available⁽⁴⁾.

The selection of a specific thermocouple for use at any particular location depends on the sensor frequency response, system accuracy and routing configuration requirements⁽³⁾.

11.3.2.2 Location of Transducers

References 4,5 and 6 describe the recommended installation, transducer design, intended reactor and/or flight application for the Chromel/Alumel thermocouples described in Paragraph 11.3.2.1. See Figure 11.3-1 for reactor location.

<u>Installation DRM</u>		<u>Reactor Application</u>
	1. 0.020" dia. 348 stainless steel sheath (978D186H01)	
52021	a) support stem liner material temp.	Ground Test
52033	b) drum vane material temp.	Ground Test
	2. 0.045" dia. 348 stainless steel sheath (963B361H01)	
52038	a) seal chamber gas temp.	Ground Test
52038	b) final seal exit gas temp.	Ground Test
	3. 0.020" or 0.045" dia. stainless steel sheath (Modified 978D186)	
52034	a) Fueled core element	Ground Test and Flight
	4. 8" Molybdenum sheath (sensing end)	
52034	a) unfueled core element	Ground Test and Flight

11.3.3. ENVIRONMENTAL AND OPERATIONAL REQUIREMENTS (U)

(U) The thermocouples must be capable of withstanding the nuclear environment of paragraph 11.3.3.1 and the non-nuclear environment of paragraph 11.3.3.2. In addition the transducers must comply with the performance characteristics designated in paragraph 11.3.3.3.

11.3.3.1 Nuclear Environment (U)

(U) Listed below are the approximate radiation levels anticipated at the support stem liner (material) temperature ⁽⁷⁾ location. This is the predicted NSS location for maximum Cr/AL thermocouple radiation exposure.

Neutron Flux

$E > 1.0 \text{ Mev}$
 $.4\text{ev} < E < 1.0\text{Mev}$
 $E < .4\text{ev}$

$4.8 \times 10^{15} \text{ n/cm}^2\text{-Sec}$
 $2.4 \times 10^{16} \text{ n/cm}^2\text{-Sec}$
 $7.5 \times 10^{12} \text{ n/cm}^2\text{-Sec}$

Neutron Fluence

$E > 1.0 \text{ Mev}$
 $.4\text{ev} < E < 1.0\text{Mev}$
 $E < .4\text{ev}$

$1.7 \times 10^{20} \text{ n/cm}^2$
 $8.5 \times 10^{20} \text{ n/cm}^2$
 $2.7 \times 10^{17} \text{ n/cm}^2$
 $2.5 \times 10^{10} \text{ RADS (C) / Hr}$
 $2.5 \times 10^{11} \text{ RADS(C)}$
 6.88 W/gm (C)

Gamma Dose Rate

Total Gamma Dose

Heating Rate

11.3.3.2 Non-Nuclear Environment (U)

(U) At this time the NERVA engine design is preliminary. The NSS material temperatures will depend on the final engine reference design ⁽⁸⁾ and the engine mode of operation. The most severe radiation environment for Chromel/Alumel thermocouples will probably occur at one of the support stem liner (material) temperature measurement locations.

(CRD) The present rated power, start of life, composite core, support stem liner temperature is about 900°R ⁽⁹⁾ (The nominal design limit during reactor operations is about 1200°R ⁽¹⁰⁾).

(CRD) Temperatures as low as $250^{\circ}\text{R}^{(9)}$ are expected during post reactor shutdown cooldown operations. The absolute liner temperature limit is $1600^{\circ}\text{R}^{(10)}$. Thermocouple cycling is defined in Reference 11.

11.3.3.3 Transducer Characteristics (U)

(U) The transducer and integral sheath cable must measure the support stem liner (material) temperature to an accuracy of $\pm 20^{\circ}\text{R}^{(11)}$ and must be capable of operation through the temperature ranges designated in Specification EC677565 with the minimum response time identified in the same document. WANL Equipment Specification 675504⁽¹²⁾ describes the temperature - EMF requirements and the calibration requirements, in addition to other detailed data, for the Chromel/Alumel thermocouples.

11.3.4 ANALYSIS (U)

(U) Irradiation effects on thermocouples are usually classified as either cumulative or transient in nature. Cumulative effects occur by transmutation of one isotope to another by thermal neutron absorption and by permanent lattice alterations caused by bombardment of fast neutron particles. These effects are fluence dependent (proportional to the total integrated flux). Transient effects, if present, are relatively short-term, reversible changes in calibration and are proportional to the dose rate or the flux level (dose rate or flux dependent).

11.3.4.1 Thermocouple Assembly (U)

11.3.4.1.1 Non-NERVA Technology (U)

(U) Some of the earliest work reported on dose rate dependence for Chromel/Alumel thermocouple performance was conducted in England. In separate tests by Boorman⁽¹³⁾ and Madsen⁽¹⁴⁾ the mean of the fusion point temperatures for various materials was greater during reactor operations ($10^{12}\text{ n/cm}^2\text{-sec}$ and 530°R - 750°R) than when the reactor was shutdown. A maximum calibration shift of $.9^{\circ}\text{R}$ was observed.

(U) Terry,⁽¹⁵⁾ using a pulsed reactor, concluded that Chromel/Alumel thermocouples were not affected by transient radiation as high as $10^{16}\text{ n/cm}^2\text{-sec}$. Leonard and Crutchfield⁽¹⁶⁾ reported that a 10°R calibration shift occurred in Chromel/Alumel thermocouples at 672°R when exposed to a reactor flux of $6.3 \times 10^{12}\text{ n/cm}^2\text{-sec}$ ($E > 0.85\text{ MeV}$).

~~CONFIDENTIAL~~

This Page Is Unclassified

~~RESTRICTED DATA~~

~~Atomic Energy Act of 1954~~

Leonard and Hunkar⁽¹⁷⁾ reported that temporary shifts of $\leq 18^{\circ}\text{R}$ in the calibration of Chromel/Alumel thermocouples at boiling water temperature persisted for several hours after exposure to a fast-neutron flux of $6.3 \times 10^{12} \text{ n/cm}^2\text{-sec}$. Their results were not attributable to gamma-heating effects. The cause of the shifts was not reported.

Another Leonard experiment⁽¹⁸⁾ showed that Chromel/Alumel thermocouples experienced similar temporary calibration shifts which persisted for at least an hour following cessation of radiation exposure. Comparison with data of a previous experiment⁽¹⁷⁾ indicated that the relative location of flux and temperature gradients along the thermocouple leads had a major influence on the magnitude of decalibration encountered. The effect was attributed to radiation produced scattering centers subject to self-annealing processes. Leonard concluded that in pile thermocouple installations should be such that temperature gradients are restricted to locations outside the high flux region to minimize potential radiation-dependent, decalibrating effects.

Dau, et al⁽¹⁹⁾ recently observed EMF shifts equivalent to $< 4^{\circ}\text{R}$ at the melting point of tin for thermocouples exposed to radiation pulses $\leq 10^{16} \text{ n/cm}^2\text{-sec}$ (fast and thermal) which they attributed to gamma heating of the thermocouple bead. In contrast to the results of Leonard and Hunkar, they found no significant transient changes in thermocouple calibration following exposure to $7 \times 10^{12} \text{ n/cm}^2\text{-sec}$ (fast and thermal).

Kelly, et al⁽²⁰⁾ found that Chromel/Alumel thermocouples exposed to a fluence of $4.1 \times 10^{20} \text{ n/cm}^2$ showed no changes in post exposure calibration that correlated with degree of exposure. Gottschlich and Bellezza⁽²¹⁾ reported an average negative EMF drift of .8% after similar exposure.

Ehringer, et al⁽²²⁾ concluded that for the temperature range 490°R to 2300°R Chromel/Alumel thermocouples can be used up to $5 \times 10^{20} \text{ n/cm}^2$ without significant problems.

~~CONFIDENTIAL~~

~~RESTRICTED DATA~~

~~Atomic Energy Act of 1954~~

Bianchi, et al⁽²³⁾ found that calibration shifts of $\leq + 36^{\circ}\text{R}$ occurred because of local gamma heating effects.

Some of these experiments involved only comparison of pre- and post-irradiation calibrations, by which technique any transient EMF changes (occurring during irradiation and disappearing when irradiation ceases) could not be detected. Aside from some of the above conflicting reports, in pile thermocouple performance shows no significant radiation rate effects below neutron fluxes of about $10^{13} \text{ n/cm}^2 \text{ - sec}$ or to neutron fluences of about 10^{20} n/cm^2 .

11.3.4.1.2 NERVA Technology

Table 11.3-1 condenses pertinent information from previous radiation testing^(24 thru 43) of Chromel/Alumel thermocouples during the NERVA Technology Program.

A total of thirty-five Chromel/Alumel thermocouples were tested in the GTR-7A, GTR-14 and GTR-15 reactor tests.

The Tuttle-Kift and Pyro-electric thermocouples tested in GTR-7A⁽²⁵⁾ (a total of 31) operated satisfactorily to the radiation levels and temperature shown in Table 11.3-1.

Small temperature deviations due to gamma heating and fast neutrons were reported⁽²⁶⁾ for GTR-14 testing. However, because of inherent data system inaccuracies somewhat greater than the maximum temperature deviation of 5°R reported, it appears unlikely that true measurable temperature deviations existed for the two thermocouples.

Significant temperature deviations were observed for the two thermocouples tested during GTR-15⁽²⁷⁾. An initial maximum temperature deviation of 5.0°R due to reactor startup gamma heating was observed. The maximum temperature deviation reported attributable to fast neutron fluence was about $+4^{\circ}\text{R}$ per 1000 MW-min. and reached about 4°R at 10,000 MW-min. ($3.5 \times 10^{16} \text{ n/cm}^2$).

A significant decrease in temperature deviation occurred at about 7000 MW-min. as shown on Figures 11.3-2 and 11.3-3. The GTR-15 test report did not indicate the cause for the decrease. At 11,800 MW-min. the GTR was retracted to enable an LH₂ trailer change over. At this time a decrease in temperature deviation of about 40% occurred (see Figure 11.3-2). The temperature of the test articles and the time required for the LH₂ trailer change over were not reported. If the test articles were still at LH₂ temperature, then the 40% decrease in temperature deviation may have occurred due to the cessation of gamma dose rate and hence gamma heating in the test thermocouples. But if the test thermocouples reached some temperature near room temperature then the 40% decrease could have been due to a combination of:

1. the elimination of gamma heating effects, and
2. some annealing out of the effects due to fast neutrons.

A similar decrease in temperature deviation occurred after reactor shutdown. The temperature environment of the thermocouples was not reported at the time (zero and 30 minutes after reactor shutdown) the measurements were made.

A total of one hundred sixty-five WANL Chromel/Alumel thermocouples were utilized for temperature measurements during the NRX reactor test series.

WANL Drawing 963B361 units were used to measure NRX-A2 and NRX-A3 STA 8.0 unfueled core element temperatures. Two NRX-A3 units experienced changes in "loop" resistance during the tests. The cause for the resistance change was not reported⁽²⁸⁾. As a result of the post-test electrical checks, the data obtained from the thermocouples were deemed questionable. One 963B361 unit was used in each of the NRX-A3⁽²⁹⁾, EST⁽³⁰⁾ and A5⁽³¹⁾ reactor tests to measure the titanium core band material temperature and did not experience any anomalies.^(28, 32, 33)

~~CONFIDENTIAL~~
~~RESTRICTED DATA~~
~~Atomic Energy, Department of Energy~~

Astronuclear
Laboratory

(U) Eight 963B361 units were used during the NRX-A6 reactor tests to measure seal chamber gas temperature and seal chamber exit gas temperature^(34, 35) without incident. This unit will be used for NERVA seal chamber gas and seal chamber exit gas temperature measurements. The gamma dose rate and fast neutron flux level should be about the same for NERVA as those experienced during the NRX-A6 reactor tests at these measurement locations.

(U) WANL Drawing 978D186 Chromel/Alumel thermocouples were used to measure NRX-A2, NRX-A3 and NRX-EST tie rod exit gas temperatures (see Table 11.3-2).

Six of these units experienced changes in "loop" resistance and four others experienced carbiding and sheath melting during the NRX-A3 reactor tests. The test document⁽²⁸⁾ did not report the causes of the observed thermocouple physical anomalies.

(CRD) 978D186 units were utilized (see Table 11.3-2; Table 11.3-2 is presented to show additional detailed transducer location and environment information not shown in Table 11.3-1) for the NRX-A5, NRX-A6 and XE reactor tests to measure tie rod exit material and the forward fueled core element temperatures. The peak shutdown tie tube temperature experienced by the Cr/Al T/C's during NRX-A6 was 1015°R. The nominal NERVA support stem liner shutdown temperature limit is expected to be 1090°R. Four of these units, utilized to measure the NRX-A6 fueled element temperature (Station 1), experienced temperature drifts of about 150°R during the EP111A 60 minute full power hold.

(U) WANL TME-1863 reported that the Copper/Constantan thermocouple used to measure the reflector material temperatures could have been as much as 35 °R high (due to radiation heating effects) if the T/C junctions did not remain in physical contact with the reflector material.

(U) It is conceivable that a similar situation existed with the Cr/Al T/C fueled core element temperature measurement.

~~CONFIDENTIAL~~
~~RESTRICTED DATA~~
~~Atomic Energy, Department of Energy~~

~~CONFIDENTIAL~~
~~RESTRICTED DATA~~
1954

TME-1863 recommended that radiation testing of these thermocouples be performed to determine:

1. if temperature drift is due to radiation effects on the T/C,
2. if the temperature rise is due to radiation effects on the fueled core element,
3. if the temperature rise is due to the T/C not being in physical contact with the core element (as probably occurred in the NRX-A6 reflector cylinder material) ⁽³⁵⁾.

RNS-0510 ⁽³⁶⁾ reported that XE thermocouples performed as expected. T/C problems encountered during reactor operations were not significant enough to warrant post-test disassembly and evaluation. Consequently radiation effects data for XE thermocouples are not available.

Figure 11.3-4 presents the NRX-A6 radiation exposure for 963B361 Chromel/Alumel seal chamber gas thermocouples in terms of percent of the radiation levels presently expected for the NERVA seal chamber gas thermocouples. The reported gamma dose rate and neutron fluxes experienced during NRX-A6 exceeded the radiation levels expected at the NERVA location. However, the gamma dose and neutron fluences were about 12 percent of the levels expected at the NERVA location.

Figure 11.3-5 presents the NRX-A6 radiation exposure for 978D186 Chromel/Alumel fueled core element (Station 1.0) thermocouples in terms of percent of the radiation levels presently expected for the NERVA support stem liner thermocouples. The NRX-A6 reported that the gamma dose rate and the fast neutron flux were about 36 percent of the levels expected at the NERVA location.

The NRX-A6 thermal neutron flux exceeded the anticipated NERVA thermal neutron flux by a factor of three. The gamma dose and neutron fluences experienced during NRX-A6 were 3 to 32 percent of the levels expected for 10-hour NERVA operation.

~~CONFIDENTIAL~~
~~RESTRICTED DATA~~
Atomic Energy Commission 1954

11.3.4. 1.3 Conclusion

Most of the Chromel/Alumel thermocouple experiments conducted in non-NERVA radiation testing resulted in pre- and post-reactor test transducer calibration temperature differences significantly less than the $\pm 20^{\circ}\text{R}$ NERVA requirement. The test conditions and T/C designs utilized do not lead to conclusions as to the applicability of the test results for NERVA applications.

Although questionable performance of the 963B361 units has occurred for material temperature measurements but not for gas temperature measurements, the thermocouples should perform satisfactorily in the anticipated NERVA temperature and nuclear gamma dose rate environment at the seal chamber temperature measurement location.

Calculations by Browning and Miller⁽⁴⁴⁾ indicate thermal neutrons cause very little transmutation (less than 2% at $10^{14} \text{ n/cm}^2 - \text{sec}$ for 20 years) in Chromel or Alumel. Additional testing of the 963B361 T/C's in a thermal neutron environment should not be required even though the NRX-A6 thermal neutron fluence was but 11 percent of that expected for the NERVA location.

The effect of fast neutron radiation on the 963B361 thermocouple has not been demonstrated for the NERVA fast neutron fluence expected. Radiation tests should be performed to determine the effects of extended fast neutron exposure and temperature cycling on the thermocouple performance.

The source of the 150°R temperature rise indicated by the 978D186 units during the NRX-A6 (radiation effects on the thermocouple or core material) reactor tests is not certain. Tests should be conducted with this thermocouple to determine:

1. if drift is due to radiation effects on the thermocouple,
2. if the temperature rise is due to radiation effects on the core material,
3. if the temperature rise is due to the thermocouple not being in physical contact with the material where the temperature measurement is made.

The temperatures (during reactor operations and post-reactor operations (i.e. cooldown) expected during NERVA will be similar to those experienced during the NRX reactor test series. Testing should be performed on the 978D186 thermocouples to determine the annealing effects (temperature cycling between 250 and 1450°R) on the transducers after exposure to the high radiation levels expected at the support stem liner temperature measurement locations.

The thermocouple considered for use for the unfueled core element temperature measurement (see 11.3.2.2) is a new and untested design. Initially, testing should be performed on a prototype (molybdenum sheath) to determine if radiation effects on this thermocouple can be expected during NERVA reactor operations.

REIRs are included summarizing the test requirements for the recommended testing.

CONFIDENTIAL
~~RESTRICTED DATA~~
 Atomic Energy Act, 1954

11-45

TABLE 11.3-1
 NERVA TEST SURVEY, NERVA TECHNOLOGY PROGRAM

Test and Date	Document References	Transducer Manufacturer and Model Number	No of XDCRS Tested	No. of XDCRS in Observed Failure Mode	TRANSDUCER ENVIRONMENT						OBSERVED FAILURE MODE						Failure Mode Mechanism			Gamma Heating	Test and Date		
					Gamma Dose Rate RADS(C)/HR	Total Gamma Dose RADS(C)	Thermal Neutron Flux $n/cm^2 \text{ sec}$ $E < 48 \text{ ev}$	Thermal Neutron Fluence n/cm^2 $E < 48 \text{ ev}$	Fast Neutron Flux $n/cm^2 \text{ sec}$ $E > 1.0 \text{ Mev}$	Fast Neutron Fluence n/cm^2 $E > 1.0 \text{ Mev}$	(CRD) Temperature $^{\circ}R$	Sensitivity Shift	Zero Shift	Drift	Non-Linearity	Off-Scale Output	Zero Output	Other	Change in Resistance			Change in Magnetic Properties	Other
GTR-7A 8/63	FZK-170-4 TME-700	a. Tuttle-Kitt b. Pyroelectric	31	0	4 5(6)	4 5(8)	4 2(9)	1.5(15)	7 8(10)	2 8(16)	140	←						** NRE					GTR-7A 8/63
GTR-14 7/64	FZK-184-4	ANSC	2	2	NP*	8.0(8)	NP*	NP*	NP*	2 3(16)	37	←			NP*							→ X	GTR-14 7/64
NRX-A2 9/64	NRX-A2 MRL TME-1079 TME-629 TNR-199	963B362H01 945C906H02 963B361H01 978D186H01	7 8 5 4	0 0 0 0	1 6(10)	NP*	NP*	NP*	2.2(15)	NP*	160 to 1900	←			NP*			▶ X					NRX-A2 9/64
GTR-15 12/64	FZK-184-6	Coriolated Controls Corp.	2	2	NP*	3 8(8)	NP*	NP*	NP*	6 4(16)	37	←			NP*			▶ X					GTR-15 12/64
NRX-A3 5/65	TME-1245 NRX-A3 MRL TME-629	963B362H01 963B361H01 945C906H02 978D186H01	12 7 16 11	1 2 2 10	1 6(10)	NP*	NP*	NP*	2 2(15)	NP*	NP*	←			NP*			▶ X					NRX-A3 5/65
NRX-EST 3/66	TME-1447 NRX-EST MRL TME-1210	963B362H01 945C906H02 978D213H01 978D186H01 978D213H01	12 16 6 1 9	0 0 0 0 0	5 4(9)	NP*	2 2(14)	NP*	6 5(14)	NP*	NP*	←			** NRE							▶	NRX-EST 3/66
NRX-A5 5/66	TME-1509 NRX-A5 MRL TME-1389	963B362H01 978D186H01 978D186H02 978D213H01 963B361H01	4 7 8 8 1	0 0 0 0 0	1.9(10)	NP*	1.3(14)	NP*	2.6(5)	NP*	NP*	←			** NRE							→	NRX-A5 5/66
NRX-A6 12/67	TME-1863 NRX-A6 MRL	963B361H01 978D186H03 978D186H02	8 4 10	0 (see remark) 0	1.2(10) 9.0(9)	1.2(10) 9.0(9)	2.0(15) 5.5(13)	7.2(18) 2.0(17)	1.1(15) 1.1(15)	3 9(18) 3.9(18)	1015 685 570	←			** NRE X ** NRE				← NP			→	NRX-A6 12/67
XE	TME-1805	978D186	13	0	7.0(v)	NP*	8.0(13)	NP*	9.0(14)	NP*	233												

NP* - Data not presented in document(s) referenced.
 NRE** - No apparent radiation effects.

CONFIDENTIAL
~~RESTRICTED DATA~~
 Atomic Energy Act, 1954

~~CONFIDENTIAL~~
~~RESTRICTED DATA~~
Atomic Energy Act of 1954

This Page Is Unclassified

TABLE 11.3-1 (CONTINUED)
TRANSDUCER TEST SURVEY, NERVA TECHNOLOGY PROGRAM

Test and Date	REACTOR ENVIRONMENT TRANSDUCER FAILURE SOURCE							Justification for Deter- mination of Failure Source		Nature of Radiation Effect			REMARKS
	Gamma Dose Rate	Gamma Dose	Thermal Neutron Flux	Thermal Neutron Fluence	Fast Neutron Flux	Fast Neutron Fluence	Other	Test Demonstrated	Analytical Engr. Judgement	Nature of Radiation Effect			
										Permanent	Transient	Synergistic	
GTR-7A 8/63							** NRE						
GTR-14 7/64	X					X		X		X	X		Initial temp deviation of about .2°R due to gamma radiation. At full reactor power temp deviations random and about .3°R. Pre- and post-test calibration temp deviation (in LH2) was .1°R. Pre- and post-test ice point calibration temp deviation (4 hr. at 710°R before post-test calibration) was .45°R average.
NRX-A2 9/64				** NRE									20 to 25% change in electrical loop resistance for three units not attributed to radiation effects.
GTR-15 12/64	X					X		X		X	X		Max initial temp deviation of +5.0°R due to gamma radiation. Max temp deviation due to fast neutron fluence was about +0.5°R per 1000 MW-min. Max absolute temp deviation reached during 18 x 10 ³ MW-min reactor run was only 10°R because of an LH2 trailer change (T/C's lost cryogenic cond) and a liquid air detonation. 30 min post-test calibration showed max 8.0°R temp deviation.
NRX-A3 5/65				NP				X		X			Change in loop resistance for 11 thermocouples was from about 2% to 152%. No conclusion report as to cause of pre- and post-test resistance differences. Data questionable. 4 tie rod exit gas T/C's (978D186H01) experienced carbiding and melting of sheathing (cause of anomalies unknown).
NRX-EST 3/66						** NRE							
NRX-A5 5/66						** NRE							One 963B362H01 unit experienced data drifts (data not presented) during EP-I. Satisfactory operation during EP-II and EP-IV. No post-test electrical resistance change in thermocouple.
NRX-A6 12/67							NP						Linear 150°R positive drift in indicated fueled element temp during 60 min. full power run of EP-III-A. TME-1863 not conclusive concerning source of temp drift, i.e. radiation effects on thermocouple or fueled element material.

~~CONFIDENTIAL~~
~~RESTRICTED DATA~~
Atomic Energy Act of 1954

CONFIDENTIAL
RESTRICTED DATA

11-47

TABLE 11.3-2
REACTOR LOCATION AND NUCLEAR ENVIRONMENT, 963B361 AND 978D186
CHROMEL-ALUMEL THERMOCOUPLES

Thermocouple Dwg. (WANL)	Reactor Test	Reactor Temperature Measurement	NUCLEAR ENVIRONMENT						(CRD)		
			Gamma Dose Rate, RADS(C) per Hour	Total Gamma Dose, RADS(C)	Thermal Neutron Flux, $E < .4 \text{ ev}$ $n/cm^2\text{-sec}$	Thermal Neutron Fluence, $E < .4 \text{ ev}$ n/cm^2	Fast Neutron Flux, $E > 1.0 \text{ Mev}$ $n/cm^2\text{-sec}$	Fast Neutron Fluence, $E > 1.0 \text{ Mev}$ n/cm^2	Temp at Full Power	Temp at Startup, 963B361 Temp - Min., Max. during Pulse Cooling & Startup (978D186)	Success Ratio
963B361	NRX-A2	Unfueled Core Ele. Sta 8.0	1.6(10)				2.2(15)				5/5
	NRX-A3	Unfueled Core Ele. Sta 8.0 Titanium Core Band	1.6(10) 3.8(9)				2.2(15) 3.8(14)				6/4 1/1
	NRX-EST	Titanium Core Band	3.8(9)		2.2(14)		3.8(14)				1/1
	NRX-A5	Titanium Core Band	3.8(9)		8.0(13)		3.8(14)				1/1
	NRX-A6	Seal Chamber Gas Seal Chamber Exit Gas	1.2(10) 2.1(9)	1.2(10) 2.1(9)	2.0(15) 3.2(14)	7.2(18) 1.2(18)	1.1(15) 2.0(14)	3.9(18) 7.2(17)	725, 1015 870, 1015	335, 370 375, 410	8/8
	NERVA	Seal Chamber Gas Seal Chamber Exit Gas	1.0(10) 2.4(9)	1.0(11) 2.4(10)	1.4(15) 9.3(13)	5.0(19) 3.4(18)	1.0(15) 4.2(14)	3.6(19) 1.6(19)			
978D186	NRX-A2	Tie Rod Exit Gas	3.6(9)				5.4(14)				4/4
	NRX-A3	Tie Rod Exit Gas	3.6(9)				5.4(14)				11/1
	NRX-EST	Tie Rod Exit Gas	5.4(9)		1.4(13)		6.5(14)				10/10
	NRX-A5	Tie Rod Exit Material Fueled Core Ele. Sta 8.0	5.0(9) 1.9(10)		1.3(14)		6.0(14) 2.6(15)				15/15
	NRX-A6	Tie Rod Material Fueled Core Ele. Sta 1.0	4.0(9) 9.0(9)	4.0(9) 9.0(9)	2.0(13) 5.5(13)	7.2(16) 2.0(17)	6.0(14) 1.1(15)	2.2(18) 3.9(18)	570, 750 519, 685	370, 485* 210, 255	10/10 4/0
	XE	Tie Rod Material Fueled Core Element	4.0(9) 7.0(9)		2.3(13) 8.0(13)		6.0(14) 9.0(14)		233	1135 470	7/7 6/6
	NERVA	Fueled Core Ele. Sta 0.2 Support Stem Liner Temp	8.0(9) 2.5(10)	8.0(10) 2.5(11)	1.8(13) 2.5(12)	6.4(17) 2.7(17)	1.2(15) 3.4(15)	4.5(19) 1.2(20)	270 900		
Moly Sheath	NERVA	Unfueled Core Element	2.5(10)	2.5(11)	6.1(12)	2.2(17)	4.8(15)	1.7(20)			

(CRD) Peak shutdown temperature was 1015°R

CONFIDENTIAL
RESTRICTED DATA

Astronautical
Laboratory

Figure 11.3-1. Reactor Temperature Measurement Locations

Figure 11.3-2. Temperature Deviation vs Radiation Exposure:
T/C 2 (CrA1, CCC T2951-2, S/N 53243)

NPC 22,515

Figure 11.3-3. Temperature Deviation vs Radiation Exposure:
T/C 3 (CrA1, CCC T2951-2, S/N 53244)

Figure 11.3-4. 963B361 Chromel-Alumel Thermocouple NRX-A6
Radiation Levels in Terms of Percent of Expected NERVA Radiation Levels

Figure 11.3-5. 978D186 Chromel-Alumel Thermocouple NRX-A6
Radiation Levels in Terms of Percent of Expected NERVA Radiation Levels

Previous Radiation Test Total Dose

Previous Radiation Test Maximum Dose Rate

100% is the Expected NERVA Requirement

REFERENCES

(Chapter 11.3 - Cr/AL T/C)

1. Thermocouple - Two Diameter Grounded Junction - Instrumentation. WANL Dwg. 978D186.
2. Thermocouple, Chromel-Alumel sheathed. WANL Dwg. 963B361.
3. E.K.Figenbaum. Instrumentation Comparison Matrix Study, Low Temperature Measurement, WANL DRM 51912 (February 1970).
4. E.K.Figenbaum. NSS Transducer Installation Study - Cluster Support Stem Low Temperature (CRD), WANL DRM 52021, (August 1967).
5. T.C.Burlas. NSS Transducer Installation Study - R-1 Seal Chambers and Final Seal Exit Gas (CRD), WANL DRM 52038, (September 1969).
6. T.C.Burlas. NSS Transducer Installation Studies - Core Fueled Element, STA 0.2 and Core Unfueled Element, STA 0.8, WANL DRM 52034, (September 1969).
7. M. A. Capo, et. al., S-100 Radiation and Shielding Analysis Report, NERVA Nuclear Subsystem, WANL-TME-2757 (To Be Published)
8. NERVA Nuclear Subsystem Reference Design Layout, WANL Dwg.939J740
9. B.L.Pierce. PAM Digital Computer Code (1 July 1970) (Unpublished).
10. R.F.Lampe. Temperature Limits for Stem Assembly Material (CRD), WANL DRM 52462 (March 1970).
11. Detail Specification, Part 1 - Performance/Design and Qualification Requirements for Nuclear Reactor Subsystem Instrumentation Forming a Part of NERVA Nuclear Subsystems, CP677555. WANL EC677565 (July 1970).
12. Equipment Specification ES675504, Chromel-Alumel Thermocouples, (June 1964).
13. M.Boorman. In-Pile Thermocouple Performance, AERE-E/R-572, United Kingdom Atomic Energy Authority, (January 1951).

REFERENCES (CONTINUED)

14. P.E.Madsen. The Calibration of Thermocouples Under Irradiation in BEPO, AERE-M/R-649, United Kingdom Atomic Energy Authority, (January 1951).
15. F.D.Terry. Effects of Transient Nuclear Radiation on Transducers and Electrical Cables, IDO-16914, National Reactor Testing Station, Phillips Petroleum Company, (1963).
16. J.H.Leonard and D.S.Crutchfield. Transient Radiation Effects on Thermocouples, University of Cincinnati, Cincinnati, Ohio, NASA-CR-83079. Paper presented at the 1966 Winter Meeting American Nuclear Society, October 30 - November 3, 1966, Pittsburgh, Pa., (NASA N67-23465).
17. J.H.Leonard and D.B.Hunkar. Direct Radiation Effects on Thermocouples, University of Cincinnati, Cincinnati, Ohio, Nuclear Applications, Vol. 3, December 1967.
18. J.H.Leonard. Conditions Contributing to Radiation-Induced Thermocouple Decalibration. University of Cincinnati, Cincinnati, Ohio, Nuclear Applications, Vol. 6, March 1969.
19. G.J.Dau, R.R.Bourassa, and S.C.Keeton. Nuclear Radiation Dose Rate Influence on Thermocouple Calibration, Nuclear Applications, Vol. 5, November 1968. Battelle Memorial Institute, Richland, Washington.
20. M.J.Kelly, W.W.Johnston, and C.D.Bauman. The Effects of Nuclear Radiation on Thermocouples, in Temperature - Its Measurement and Control in Science and Industry, Vol. 3, Part 2, Reinhold, N.Y., (1962).
21. R.Gottschlich and M.Belezza. EMF Deviation of Chromel/Alumel Thermocouples caused by Neutron Irradiation, EUR 3257e, European Atomic Energy Community, (December 1966).

REFERENCES (CONTINUED)

22. H.Ehringer, C.Monigini-Tamaggoni, and C.Ponti. Thermocouple Composition Changes Due to Neutron Irradiation, European Atomic Energy Community, Euratom, Ispra, Italy, EUR 3156.e (NASA N67-23093), (1966).
23. G.Bianchi, P.Lupoli, and S.Moretti. Behavior of Irradiated Thermocouple-Experimental Investigations, Symposium, - In-Core Instrumentation, Oslo, Norway, June 1964, CONF-640607, Vol. II Paper D-4.
24. NERVA Components Irradiation Program, Vol. 4, GTR Test 7A, GD/FW-FZK-170-4 (November 1963).
25. Test Report on a WANL Radiation Test of NERVA Components, GTR Tests 7 and 7A, TME-700 (CRD), March 1964.
26. NERVA Components Irradiation Program, Vol. 4, GTR Test 14, GD/FW-FZK-184-4 (November 1964).
27. NERVA Components Irradiation Program, Vol. 2, GTR Test 15, FZK-184-6 (March 1965).
28. NRX-A3 Instrumentation Disassembly and Post-Operative Report, WANL-TME-1245 (CRD), (August 1965).
29. Measurement Requirements Test, NRX-A3 (CRD) (May 1965).
30. Measurement Requirements List, NRX/EST (CRD), (March 1966).
31. Measurement Requirements List, NRX-A5 (CRD) (August 1966).
32. NRX/EST Disassembly and Post-Operative Examination Report, WANL-TME-1447 (CRD), (June 1966).
33. NRX-A5 Sensor Disassembly and Post-Operative Examination Report, WANL-TME-1509 (CRD) (September 1966).
34. Measurements Requirements List, NRX-A6 (CRD) (January 1968).
35. NRX-A6 Transducer Evaluation Report, WANL-TME-1863 (CRD) (December 1968).

REFERENCES (CONTINUED)

36. XE-Prime Engine Final Report, Vol. II, Assembly, Test, and Disassembly Operations, ANSC-RNS-0510 (April 1970).
37. Radiation Effects Program Review, WANL-TME-629 (CRD) (December 1963).
38. Measurement Requirements List, NRX-A2 (August 1964).
39. NRX-A2 Instrumentation Disassembly and Post-Operative Report, WANL-TME-1079 (CRD) (January 1965).
40. Analysis of NRX-A2 Reactor Test Results, WANL TNR-199 (CRD) (April 1965), Supplement 1 (CRD) (May 1965), Supplement 2 (CRD) (July 1965).
41. XE-2 Nuclear Subsystem Thermal and Nuclear Design Data Book. WANL-TME-1805 (CRD) (August 1968).
42. NRX-A4/EST Reactor Analysis Data. WANL-TME-1210 (CRD) (August 1965).
43. NRX-A6 Reactor Analysis Data. WANL-TME-1389 (CRD) (April 1966).
44. W.E.Browning and C.E.Miller, Jr. Calculated Radiation Induced Changes in Thermocouple Composition. From: Temperature-Its Measurement and Control in Science and Industry - Vol. 3, Part 2 (1962).

RADIATION EFFECTS INFORMATION REQUESTS (REIRs)
(Chapter 11.3 - Cr/Al T/C)

<u>No.</u>	<u>Description</u>
23	CR/Al T/C 963B361
24	Cr/Al T/C 978D186
25	Cr/Al T/C Moly Sheath

~~CONFIDENTIAL~~
~~RESTRICTED DATA~~
~~Atomic Energy, January 1964~~

11-57

IDENTIFICATION NO. 23

RADIATION EFFECTS INFORMATION REQUEST

DATE October 1970

SUBJECT Chromel-Alumel Thermocouple, 9638361 FUNCTIONAL CATEGORY Engr. Test & Evaluation

REF. ITEM SUPPORT EC 677565 REF. DATA ITEMS S-110 (Chapter 11.3)

INFORMATION DESIRED The effect of fast neutrons and temperature cycling on the thermocouple electrical output.
(PURPOSE OF STUDY OR EXPERIMENT)

DATE REQUIRED: _____

RELATION OF ITEM TO ENGINE Seal Chamber and Seal Chamber Exit (EC 677564, DRM 5203C)
(LOCATION)

PER DRAWINGS, DATA ITEMS _____

ENVIRONMENT (Ref. data items) NEUTRON FLUX 1.0(15)n/cm²-sec E>1.0 Mev NEUTRON FLUENCE 316(19)n/cm² E > 1.0 Mev

GAMMA DOSE RATE N/A GAMMA DOSE N/A

TEMPERATURE 900°R NUCLEAR HEATING RATE N/A

PRESSURE N/A OTHER N/A

FLUID TBD

CYCLE REQUIREMENTS 900°R - 1100°R - 250°R

RELIABILITY ESTIMATE 0.99990 Probability of Success REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL N/A APPROVED _____ DATE _____

APPROVED _____ DATE _____

SPECIAL CONSIDERATIONS N/A APPROVED _____ DATE _____

PERTINENT REFERENCES _____

~~CONFIDENTIAL~~
~~RESTRICTED DATA~~
~~Atomic Energy, January 1964~~

This Page Is Unclassified

IDENTIFICATION NO. 24

RADIATION EFFECTS INFORMATION REQUEST

DATE October 1970

SUBJECT Chromel-Alumel Thermocouple, 978D186 FUNCTIONAL CATEGORY Eng. Test & Evaluation

REF. ITEM SUPPORT EC 677565 REF. DATA ITEMS S-110 (Chapter 11.3)

INFORMATION DESIRED The effects of radiation and temperature cycling on the thermocouple electrical output.
(PURPOSE OF STUDY OR EXPERIMENT)

DATE REQUIRED: _____

RELATION OF ITEM TO ENGINE Fueled Core Element, Sta 0.2 (EC 677566, DRM 52034)
(LOCATION)

PER DRAWINGS, DATA ITEMS and Support Stem Liner (EC 677575, DRM 52021)

ENVIRONMENT FAST NEUTRON FLUX 3.4(15)n/cm²-sec, E > 1.0 Mev FAST NEUTRON FLUENCE 1.2(20)n/cm², E > 1.0 Mev
(Ref. data items) Thermal Neutron Flux 1.9(13)n/cm²-sec E < .4 ev Thermal Neutron Fluence 6.4(17)n/cm² E < .4 ev
GAMMA DOSE RATE 2.5(10) RADS(C)/HR GAMMA DOSE 2.5(11) RADS(C)

(CRD) TEMPERATURE 270, 900°R NUCLEAR HEATING RATE 69.5 W/gm(C)

PRESSURE N/A OTHER N/A

FLUID TBD

(CRD) CYCLE REQUIREMENTS 900-1200-250°R

RELIABILITY ESTIMATE 0.99990 Probability of Success REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL N/A APPROVED _____ DATE _____

APPROVED _____ DATE _____

SPECIAL CONSIDERATIONS N/A APPROVED _____ DATE _____

PERTINENT REFERENCES _____

11-58

IDENTIFICATION NO. 25

RADIATION EFFECTS INFORMATION REQUEST

DATE October 1970

SUBJECT Chromel-Alumel Molybdenum Sheath Thermocouple FUNCTIONAL CATEGORY Engr. Test & Evaluation

REF. ITEM SUPPORT EC 677565 REF. DATA ITEMS S-110 (Chapter 11.3)

INFORMATION DESIRED The affects of radiation and temperature cycling on the thermocouple electrical output.
(PURPOSE OF STUDY OR EXPERIMENT)

DATE REQUIRED: _____

RELATION OF ITEM TO ENGINE Unfueled core element temperatures.
(LOCATION)

PER DRAWINGS, DATA ITEMS (EC 677566, DRM 52034)

ENVIRONMENT FAST NEUTRON FLUX 4.8(15)n/cm²-sec E > 1.0 Mev FAST NEUTRON FLUENCE 1.7(20)n/cm² E > 1.0 Mev
(Ref. data items) Thermal Neutron Flux 6.1(12)n/cm²-sec E < .4 ev Thermal Neutron Fluence 2.2(17)n/cm² E < .4 ev
GAMMA DOSE RATE 2.5(10)RADS(C)/HR GAMMA DOSE 2.5(11) RADS(C)

TEMPERATURE TBD NUCLEAR HEATING RATE 69.5 W/gm(C)

PRESSURE N/A OTHER N/A

FLUID TBD

CYCLE REQUIREMENTS TBD

RELIABILITY ESTIMATE 0.99990 Probability of Success REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL N/A APPROVED _____ DATE _____

APPROVED _____ DATE _____

SPECIAL CONSIDERATIONS N/A APPROVED _____ DATE _____

PERTINENT REFERENCES _____

11-59

REPORTS (T-119s) RESPONDING TO REIRs

(None at this issuance)

11.4 HIGH TEMPERATURE THERMOCOUPLE

11.4.1 SUMMARY AND CONCLUSIONS

Two different thermocouple designs will be utilized for measuring NERVA reactor core exit gas and unfueled core element temperatures in the high temperature range. The thermocouple design for the core exit gas temperature measurement is established but the design of the thermocouple to be utilized for the unfueled core element temperature measurement has not been determined. Hence, this report includes only radiation effects analysis of the core exit gas thermocouple. The design selected is significantly different than the NRX-A6/XE thermocouple design and has not been radiation tested. Although the NERVA thermocouple is not expected to provide excess temperature measurement errors due to radiation effects, the importance of the measurement for reactor control and evaluation dictates that radiation testing be performed. The testing should be designed to determine the effects of temperature, fast neutrons, thermal neutrons, and gamma rays on the performance of the NERVA design thermocouple and/or its components. Radiation Effects Information Requests are included in this report summarizing the test requirements.

11.4.2 INTRODUCTION

The measurement of high temperatures in the NERVA reactor involves in-core material and cluster exit gas (CEG) measurements. The thermocouple design for the cluster exit gas temperature measurement is established but the design of the thermocouple to be utilized for the unfueled core element temperature measurements has not been determined.⁽¹⁾ Hence, this report will include only radiation effects analysis of the cluster exit gas thermocouple.

CEG thermocouples must provide an accurate (see 11.4.3.3) determination of hydrogen temperatures for computation of the mixed mean chamber temperature.

11.4.2.1 Transducer Description

The cluster exit gas exposed junction thermocouple (Figure 11.4-1) currently considered for NERVA NSS use consists of: (1) (2)

1. thermoelements (wires) of pure tungsten and tungsten with 26 percent rhenium (W/W-26% Re),
2. sheathing made of thoriaated tungsten (2% Thoria) over the sensing end length exposed in the core exit plenum and stainless steel from the cluster hot end cap to the core support plate,
3. Beryllia (BeO) insulation.

At the present time a detailed design drawing for the NERVA W/W-26% Re thermocouple is not available.

11.4.2.2 Location of Transducer

WANL DRM 52032 describes the recommended installation details of the cluster exit gas thermocouples considered for R-1 utilization. Figure 11.4-2 shows the installation location.

Reference 3 dictates that the cluster exit gas temperature measurement shall be a flight reactor parameter.

11.4.3 ENVIRONMENTAL AND OPERATIONAL REQUIREMENTS

The thermocouple must be capable of withstanding the nuclear and non-nuclear environments and must comply with the performance characteristics described in the following paragraphs.

11.4.3.1 Nuclear Environment

Listed below are the approximate radiation levels anticipated for the cluster exit gas thermocouples.⁽⁴⁾ The values given are the maximum expected for the 10-hour NERVA reactor:

Neutron flux

$E > 1.0 \text{ Mev}$	$3.3 \times 10^{14} \text{ n/cm}^2 - \text{sec}$
$0.4 \text{ ev} < E < 1.0 \text{ Mev}$	$1.2 \times 10^{15} \text{ n/cm}^2 - \text{sec}$
$E < .4 \text{ ev}$	$1.1 \times 10^{13} \text{ n/cm}^2 - \text{sec}$

Neutron fluence

$E > 1.0 \text{ Mev}$	$1.2 \times 10^{19} \text{ n/cm}^2$
$0.4 \text{ ev} < E < 1.0 \text{ Mev}$	$4.1 \times 10^{19} \text{ n/cm}^2$
$E < .4 \text{ ev}$	$4.0 \times 10^{17} \text{ n/cm}^2$
Gamma dose rate	$1.3 \times 10^9 \text{ RADS(C)/HR}$
Total Gamma dose	$1.3 \times 10^{10} \text{ RADS(C)}$
Heating rate	3.6 w/gm (C)

11.4.3.2 Non-Nuclear Environment

At this time the NERVA engine design⁽⁵⁾ is preliminary. The NSS average hydrogen temperatures, pressures and mass velocity will depend on the final engine reference design and the engine mode of operation.

The present start of life, composite core, cluster exit gas, rated power state points are:⁽⁶⁾

hydrogen mass velocity	TBD lb/sec/in ² flow area
temperature	4290°R
pressure	450 psia
cycles	As defined in Reference 3

The maximum temperature the thermocouple could experience, within the NSS Specification⁽⁷⁾, is about 4483°R⁽⁸⁾. For a nominal chamber temperature of 4500°R the maximum temperature, within specification, is 4733°R. During normal post reactor shutdown cooldown operations the W/W-26% Re thermocouple temperature is expected to drop to about 1500°R (600 seconds at full reactor power with 30 seconds at throttled power)⁽⁹⁾. The temperatures expected for an Earth orbital mission (non-rotating vehicle) are about 450°R to 550°R⁽¹⁰⁾ (reactor not operating).

11.4.3.3 Transducer Characteristics

The thermocouple assembly must measure the cluster exit gas temperature to an accuracy of $\pm 50^{\circ}\text{R}$ and must be capable of operation through the temperature ranges described in 11.4.3.2. The thermocouple must conform to the reliability, response, hydrogen embrittlement, and operating service requirements designated in WANL EC677565.

Equipment Specification, 675505,⁽¹¹⁾ describes the temperature-EMF requirements and the calibration requirements for the tungsten/tungsten-26% rhenium thermocouple.

11.4.4 ANALYSIS

11.4.4.1 Thermocouple Assembly

Little radiation effects information is available for the NERVA selected type of T/C composed of W/W-26% Re thermoelements. Data for T/Cs with thermoelements composed of W/W-Re other than 26 or 25 percent Re are available but not considered here; i.e., 3 percent, 5 percent, etc. Re in one thermoelement leg. This report will be primarily concerned with radiation effects work performed for the NERVA material combinations of thermoelements, insulation and sheathing.

11.4.4.1.1 Non-NERVA Technology

The gradual transmutation, with time, of tungsten and rhenium into osmium causes the thermocouple thermoelectric output to change. The majority of transmutation of tungsten and rhenium is produced by thermal neutrons.⁽¹²⁾ Browning and Miller calculated the theoretical composition of tungsten and tungsten-26% rhenium after exposure to a thermal flux of 1×10^{14} n/cm²-sec. The alloy compositions for up to six months exposure are shown in Table 11.4-1.

Kuhlman⁽¹³⁾ utilized the calculated alloy compositions after irradiation to the times indicated in Table 11.4-1 by preparing specimens composed of the elements in the proportions shown in Table 11.4-1. The deviation in thermoelectric output between laboratory prepared reference specimens and synthesized transmuted thermoelements represented the change which could be expected in W and W-25% Re thermoelements when used in a nuclear reactor under comparable neutron flux (10^{14} n/cm²-sec) and time. It should be stressed here that Kuhlman did not irradiate his specimens but merely measured the indicated temperature difference between standard W/W-25% Re thermoelements and T/C's composed of the thermoelements shown in Table 11.4-1. Of interest is the error curve for a W-1%Os/W-25%Re thermocouple shown in Figure 11.4-3.

The data presented by Kuhlman indicates that an exposure time of 10 hours could result in temperature errors due to transmutation much less than the $\pm 50^\circ\text{R}$ NERVA accuracy requirement (it is recognized that the inaccuracy due to radiation effects will contribute to only a portion of the total accuracy deviation of a thermocouple).

Skaggs, et al⁽¹⁴⁾ reported that a single W/W-25% Re thermocouple radiated to a thermal neutron fluence of 3.8×10^{20} n/cm² at an average temperature of 3600°R experienced a post radiation calibration error of about $\pm 22^\circ\text{R}$ at 3900°R ($\pm 16^\circ\text{R}$ at 2160°R).

The results concluded by Skaggs, et al, indicated that the magnitude in calibration change was dependent on:

1. the amount of transmutation occurring
2. the time-temperature conditions during irradiation

11.4.4.1.2 NERVA Technology

Core exit gas temperature measurements were first performed on the NRX-A6 and XE reactors. The thermocouple (WANL Dwg. 979D149H02) utilized for providing this measurement was significantly different than the design currently considered for the NERVA cluster exit gas temperature measurement (see Figure 11.4-1). Consequently, test data for the NERVA thermocouple design does not exist.

Other than geometry, the major differences between the NRX-A6/XE and NERVA thermocouples are:

1. The 979D149H02 design utilized a tungsten-26% rhenium sheath at the exposed sensing end and the NERVA design will probably use a thoriated tungsten sheath at the hydrogen gas exposed sensing portion of the thermocouple.
2. The NERVA thermocouple will not use temperature limited brazing materials exposed to the hot core exit gas as did the 979D149 design.

Table 11.4-2 lists the radiation and temperature environments for the twenty-one 979D149H02 W/W-26% Re thermocouples used in each NRX-A6⁽¹⁵⁾ and XE reactor⁽¹⁶⁾ test. Also listed for comparison is the anticipated NERVA environment for the thermocouple in Figure 11.4-1

The NRX-A6 thermocouples performed satisfactorily. Although radiation effects analysis was not conducted on XE instrumentation, RN-S-0510⁽¹⁷⁾ did report that two cluster exit gas thermocouple data channels were questionable. The cause (radiation effects

on thermocouples or data acquisition system problems) was not identified. The thermocouples experienced steady state temperatures between 3000 and 4600° R during the reactor tests.

The NERVA reactor environment is expected to be nearly that of the NRX-A6 and XE reactors for this measurement location (Table 11.4-2). If the thermocouple design utilized in the NRX-A6 and XE reactors would be used in the NERVA reactors, it could be concluded that additional radiation effects testing would not be required. However, since the NERVA reactor cluster exit gas thermocouples are significantly different in design and installation than the NRX-A6 and XE thermocouples, radiation effects testing on the new thermocouple appears necessary.

11.4.4.1.3 Conclusion

The NERVA thermocouple design specified for the core exit gas temperature measurement has not been radiation tested. Although this thermocouple is not expected to provide excess temperature measurement errors due to radiation effects, the importance of the measurement for reactor control and evaluation and the high NERVA radiation levels expected dictate that additional testing be performed. The testing should be designed to determine the effects of temperature, fast neutrons, thermal neutrons and gamma rays on the performance of the NERVA design thermocouple and/or its components.

The REIR(s) included describe the test requirements for the recommended testing.

TABLE 11.4-1
CALCULATED COMPOSITIONS OF W AND W - 25 Re
AFTER EXPOSURE TO A 10^{14} n/cm² - sec FLUX⁽¹⁰⁾

<u>Time, Months</u>	<u>Tungsten Alloy Compositions</u>	<u>W - 25Re Alloy Composition</u>
0	W - 0Re - 0 Os	W - 25Re - 0 Os
0.5	W - 0.09Re - 0 Os	W - 24.83Re - 0.24 Os
1.0	W - 0.18Re - 0 Os	W - 24.65Re - 0.48 Os
3.0	W - 0.55Re - 0.01 Os	W - 23.98Re - 1.45 Os
6.0	W - 1.17Re - 0.05 Os	W - 23.13Re - 2.78 Os

TABLE 11.4-2
RADIATION AND TEMPERATURE ENVIRONMENT, NRX-A6 NAD XE REACTOR TESTS AND NERVA REACTOR

Reactor Test	Gamma Dose Rate RADS(C)/HR	Total Gamma Dose RADS(C)	Thermal Neutron Flux n/cm^2 -sec	Thermal Neutron Fluence n/cm^2	Fast Neutron Flux n/cm^2 -sec $E > 1.0$ Mev	Fast Neutron Fluence n/cm^2 $E > 1.0$ Mev	Temperature °R
NRX-A6	2.0(9)	2.0(9)	1.8(13) $E < 1.86$ ev	6.5(16) $E < 1.86$ ev	2.0(14)	7.2(17)	3035 to 4575
XE	2.0(9)		2.2(13) $E < 1.86$ ev		3.0(14)		3630 to 4400
NERVA	1.5(9)	1.5(10)	1.1(13) $E < .4$ ev	4.0(17) $E < .4$ ev	3.3(14)	1.2(19)	4286 avg.

Figure 11.4-1. Cluster Exit Gas Candidate Thermocouple Design Profile

Figure 11.4-2. Reactor Temperature Measurement Locations

Figure 11.4-3. Deviation from Initial Calibration in Degrees Rankine of a W/W - 25 Re Thermocouple as a Function of Temperature after 0.5, 1, 3, 6 Months at 10^{14} nv (thermal) and also the Deviation of a W - 1 Os/W - 25 Re Thermocouple after 6 Months at 10^{14} nv (thermal). (from Reference 13)

REFERENCES

(Chapter 11.4 - W/W-26 Re T/C)

1. T. C. Burlas, Trade Study Report on NERVA High Temperature Thermocouples; Cluster Exit Gas (CEG) and In-Core. WANL DRM 51914 March 1970 .
2. T. C. Burlas, NSS Transducer Installation Studies - Cluster Exit Gas Thermocouple, WANL DRM 52032 (CRD) September 1969.
3. Detail Specification, Part 1 - Performance/Design and Qualification Requirement for Nuclear Reactor Subsystem Instrumentation Forming a Part of NERVA Nuclear Subsystem, EC 677565 July 1970 .
4. M. A. Capo et al, S-100 Radiation and Shielding Analysis Report, NERVA Nuclear Subsystem, WANL TME-2757 (To be published).
5. NERVA Nuclear Subsystem Reference Design Layout, WANL Drawing 939J723.
6. NSS State Points - Flight Engine Composite Core, WANL Drawing 957A982.
7. NERVA Nuclear Subsystem, CEI Specification 4, CP-677555, September 1970.
8. Thermal and Fluid Environmental Drawing, WANL 939J746 (To be published).
9. D. L. Black, Trade Study 757 - Reactor Cooldown Criteria, WANL TME-2728 (CRD) (To be published).
10. A. C. Thadani, Temperature Distributions Within Non-Rotating Vehicle, WANL DRM-2943 (August 1970).
11. Equipment Specification, Thermocouple, Tungsten/Tungsten-Rhenium- EC 6775505 January 1967 .
12. W. E. Browning, Jr. and C. E. Miller, Jr., Calculated Radiation Induced Changes in Thermocouple Calibration. Reactor Chemistry Division, ORNL, Tenn. From "Temperature, Its Measurement and Control in Science and Industry", Vol. 3, Part 2 (1962).
13. W. C. Kuhlman, Thermocouple Research at GE-NMPO, presented at Technical Meeting on High Temperature Thermometry, Washington, D. C., WASH 1067, February 1965.

REFERENCES (Continued)

14. S. R. Skaggs, W. A. Rankin and A. J. Patrick, "Decalibration of a W/W-25% Re Thermocouple in a Neutron Flux", University of California, Los Alamos Scientific Laboratory, New Mexico, LA 3662, January 1967.
15. NRX-A6 Transducer Evaluation Report, WANL TME-1863 (CRD), December 1968.
16. XE-2 Nuclear Subsystem Thermal and Nuclear Design Data Book, WANL TME 1805 (CRD) August 1968.
17. XE-PRIME Engine Final Report, Test Data Analysis, ANSC RNS-0510, May 1970.

RADIATION EFFECTS INFORMATION REQUESTS (REIRs)

(Chapter 11.4 - W/W 26 Re T/C)

<u>No.</u>	<u>Description</u>
26	W/W 26 Re T/C

IDENTIFICATION NO. 26

RADIATION EFFECTS INFORMATION REQUEST

DATE October 1970

SUBJECT Tungsten/Tungsten-26% Rhenium FUNCTIONAL CATEGORY Eng. Test & Evaluation

REF. ITEM SUPPORT EC 677565 REF. DATA ITEMS S-110 (Chapter 11.4)

INFORMATION DESIRED To determine the effects of radiation and temperature cycling on thermocouple electrical
(PURPOSE OF STUDY OR EXPERIMENT) output

DATE REQUIRED: _____

RELATION OF ITEM TO ENGINE Cluster Exit Gas (EC-677558, DRM 52032 (CRD))
(LOCATION)

PER DRAWINGS, DATA ITEMS _____

ENVIRONMENT : NEUTRON FLUX 3.3(14)n/cm²-sec E>1.0 Mev NEUTRON FLUENCE 1.2(19) n/cm² E>1.0 Mev
(Ref. data items) Thermal Neutron Flux 1.1(13)n/cm²-sec E <.4 ev Thermal Neutron Fluence 4.0(17)n/cm² E <.4 ev
GAMMA DOSE RATE 1.3(9) RADS(C)/HR GAMMA DOSE 1.3(10) RADS(C)

TEMPERATURE 4400°R NUCLEAR HEATING RATE 3.6 W/gm(C)

PRESSURE N/A OTHER N/A

FLUID GN₂

CYCLE REQUIREMENTS 450-4400-1500

RELIABILITY ESTIMATE 0.99990 Probability of Success REQUESTED BY _____ DATE _____

CONFIDENCE LEVEL N/A APPROVED _____ DATE _____

APPROVED _____ DATE _____

SPECIAL CONSIDERATIONS N/A APPROVED _____ DATE _____

PERTINENT REFERENCES N/A

REPORTS (T-119s) RESPONDING TO REIRs

(None at this issuance)

CHAPTER 12

PROPELLANTS

(Chapter not required for NSS PDR)

TITLE Radiation Effects Analysis Report Nuclear Subsystem	
		DOCUMENT NO. WANL-TME-2751	DATE 12-10-70
ABSTRACT <p>This document presents the effects of nuclear radiation on the materials and components of the Nuclear Subsystem. The materials used in the NSS Engineering Critical Components are evaluated to determine if significant mechanical/physical property changes can be expected to occur under conditions experienced in the anticipated thermal and radiation environments.</p> <p>In those cases where there is insufficient data to make a correlation, Radiation Effects Information Requests are presented.</p>			
DATA ITEM NO. S-110		SECURITY CLASSIFICATION OF DOCUMENT C - RD	
CDRL LINE NO. N/A		TOTAL NO. OF PAGES 240	
KEY WORDS <p>Radiation Effects, Support Plate & Plena, Aluminum Alloys, Stainless Steels 302, 304, 321, A-286, Internal Shield, BATH, Lead, Reflector Assembly, Beryllium, Copper-Boron (18%), Graphite, Control Drum Drive Assembly, Structural Support Coolant Assembly, Instrumentation.</p>			
AUTHOR/EDITOR T.P. Merrick, J.A. DeMastry, G.A. Davis		PREPARING DEPT. Systems Integration	
W Form 59943		DOCUMENT INFORMATION SUMMARY	

Fold
or Cut

Fold
or Cut