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RIFT HAZARDS
SUMMARY REVIEW

(Title Unclassified)

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INTRODUCTION

This document presents the Westinghouse Astronuclear Laboratory contribution to the RIFT flight safety review. While the information presented herein is the best data that exists at this time, it can not be considered the final data. In consideration of the extremely dynamic nature of the program, it is intended to issue periodic revisions reporting new developments in the Westinghouse program.

This document is divided into five parts. Part 1 describes the reactor and certain of its mechanical and nuclear operating characteristics. Part 2 describes the reactor control system and gives the operating specifications for the control drum actuator. Part 3 discusses the poison systems for shipping, ground handling, and in-flight. Part 4 is devoted to destruct systems, and includes a discussion of reentry physics, chemical destruct system, nuclear destruct system, and fission product inventory. Part 5 is concerned with the toxicity of Beryllium, and is transmitted as a separate document.

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1.0 REACTOR DESCRIPTION

1.1 Introduction

The NERVA engine is adapted from the LASL KIWI B-4 design, suitably modified for its use as a flight propulsion unit. The environment which the NERVA engine must withstand is given in Table 1. The ground handling and powered flight operating modes require that the core and reflector be capable of carrying acceleration loads in all directions, as opposed to the KIWI design, which was designed only for static tests under gravity loads. This requirement led to the need for a system to offer radial support to the core structure.

In its flight application, the NERVA engine is mounted atop the second stage with its nozzle in down-firing position. A tank carrying the coolant-propellant, liquid hydrogen, is mounted above the reactor. In order to prevent excessive nuclear heating in this cryogenic tank, a shield structure must be provided between the tank and the reactor.

In operation, the liquid hydrogen is pumped to the reactor at a rate of 71.7 lb/sec., 706 psia, and 100°R. It enters the core at 180°R and 665 psia, and exits at 4090°R and 550 psia. Full power operation is 1120 MW. The principal parts of the reactor are the core, an inner graphite reflector, an outer beryllium reflector which contains the control drums and the shield. The reactor is installed in a pressure vessel, with the exhaust nozzle at its bottom end (Fig. 1). In operation, liquid hydrogen flows through the nozzle, cooling it regeneratively, and into the reactor (Fig. 2). Hydrogen flows through the inner and outer reflectors, through the annulus between them, and the annulus between the outer reflector and the pressure vessel, through the control drums, and the annulus between the drums and the outer reflector. A seal in the bottom of the inner reflector prevents the flow from short-circuiting the core, and is capable of absorbing the relative radial thermal expansion (3/16") between the core and reflector. At the top of the reactor, the flow is baffled to pass upward through the outer shield region, part of the flow being ported between the shield and pressure vessel. In the upper plenum region, the coolant is redirected downward, cooling the inner shield region and core, and exhausting through the nozzle.

1.1.1 Reactor Core

The reactor consists of 235 fuel clusters supported by steel tie rods from a top end

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SUMMARY OF REACTOR ENVIRONMENT

TABLE 1

	SHIPPING	GROUND HANDLING	STATIC TEST	BOOSTER OPERATION	NERVA ENGINE OPERATION
1.0 SUSTAINED LOADING					
1.1 Linear Acceleration	1g along any axis	1.5g along any axis	$\pm 1g$ parallel to thrust axis. Zero normal to thrust axis.	$\pm 8.0g$, $-1.2g$ parallel to thrust axis. $\pm 1.0g$ normal to thrust axis.	$\pm 1.0g$ both parallel and normal to thrust axis.
1.2 Spin acceleration and velocity	Negligible	.010 rad/sec ² and 1 rpm.	Not applicable	Negligible	Negligible
1.3 Angular acceleration and velocity in pitch and yaw modes	Negligible	Not applicable	Not applicable	1 rad/sec ² and 10 degrees per sec.	1 rad/sec ² and 25 degrees per sec.
2.0 DYNAMIC LOADING					
2.1 Shock	5g along any axis	6g along any axis maximum duration estimated to be 0.010 sec.	5g along any axis maximum duration estimated to be 0.010 sec.	1g along any axis maximum duration estimated to be 0.010 sec.	1g along any axis maximum duration estimated to be 0.010 sec.
2.2 Vibration	Negligible	See Figure 3	Negligible	See figures 3 & 4	See figure 3
3.0 AMBIENT CONDITIONS					
3.1 Pressure	30 inch Hg	30 inch Hg	30 inch Hg	30 inches Hg to 10 ⁻⁸ mm Hg	10 ⁻⁸ - 10 ⁻¹² mm Hg
3.2 Temperature	-80°F to +160°F	-80°F to +160°F	-65°F to +160°F	-130°F to +103°F	External heat supplied by radiation only.
3.3 Humidity	5 - 20%	Zero to 100%	Zero to 100%	Zero to 100%	Zero

FIG. 1

REACTOR ASSEMBLY

FIG. 2
REACTOR ELEVATION - FLOW SCHEMATIC

core support plate (Fig. 3). The core weight 2750 lbs., is nominally 35" in diameter and 52" long. To achieve power flattening, the core is divided into 5 fuel loading regions. The specification of this loading, and the assignment of the regions, are described in Section 1.3.2.

1.1.1.1 Fuel Cluster

A fuel cluster consists of a central hexagonal unloaded element, surrounded by 6 hexagonal loaded fuel elements. At the top or inlet end, each fuel element is fitted with an aluminum orifice jet, which also serves to locate the top end of the elements in an aluminum cluster plate (Fig. 4). The orifice accommodates variations in radial power, fuel loading and flow channel diameter. At the nozzle or outlet end, each of the 6 fueled elements is located in a niobium carbide coated graphite support block by a graphite alignment sleeve (Fig. 5).

At the top of the cluster, the tie rod is centered in the cluster plate by a centering bushing. This bushing also serves to locate the center unfueled element with regard to the cluster plate. The fuel cluster assembly is provided with a compression spring and top fitting which is bolted into the core support plate. The entire assembly is held together, in compression, by the tie rod. The tie rod locking device is so designed that all parts will be captive even should the rod nut fail.

1.1.1.2 Fuel Elements

All elements are extruded of a mixture of uranium carbide beads (coated with pyrolytic graphite) in a graphite matrix. They are 52" long, hexagonal, and nominally .753" across flats. The loaded elements have 19 flow holes, .097" diameter, running their length. The holes are coated internally with niobium carbide for corrosion resistance, as is the hot end surface of all elements.

The central element has a single hole down its center, .362" in diameter, to accept an .054" thick pyrographite insulating sleeve which is, in turn, lined with a .005 stainless steel liner to prevent graphite flakes from clogging the cooling passages.

Loaded elements are clearly marked with a seven digit number indicating the fuel loading (first digit), the size, for those elements which are cut to achieve a cylindrical periphery (second digit), and a five digit serial number.

FIG. 3
CORE ASSEMBLY

FIG. 4
FUEL CLUSTER - INLET END

The unloaded element is marked with a six digit number indicating its unloaded condition (first digit zero), 1 hole (second digit one) or 13 hole (second digit 2) and a four digit serial number.

Twelve .10" diameter holes are also provided around the outer edge of the unloaded elements in the central core region to accept a core poison or graphite if the need for reactivity adjustment is indicated during the proof test critical experiment.

1.1.1.3 Assembled Core

Each cluster is top supported from the core support plate by a .110" diameter Inconel X tensile tie rod. At the bottom end, the tie rod is formed into a knob and bears against a pierced molybdenum cone. This cone bears, in turn, against a pyrolytic graphite washer and cup in the supporting block (Fig. 5). Thus, the tie rod is insulated from heat generated in the surrounding loaded elements, and is cooled by coolant flowing from the inlet plenum down the pyrographite annulus, and out the cone.

The core support block design incorporates an interlocking feature so that neighboring clusters will retain a fuel cluster in the event its tie rod fails (Fig. 6).

The outer face of the core approximates a cylindrical shape by the shaping of the outer elements, and the addition of graphite filler pieces.

Volume Fractions of Core Material:

	<u>Void Fraction</u>
Fueled graphite	.584
Unfueled graphite	.119
Pyrographite	.011
Coolant channels - void	.264
Niobium coating	.011
Support rods	.0041

1.1.2 Core Support Plate

This member is an aluminum structure, 37.5" in diameter, varying in thickness from 4.12" to 6", bored on its bottom face to accept the upper ends of the fuel cluster assemblies and counterbored on its top face to accept the tie rods nuts and locking devices. The core support plate transfers the entire core load to the titanium dome end reflector

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FIG. 5

FUEL CLUSTER - OUTLET END

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FIG. 6

INTERLOCKING SUPPORT BLOCKS

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support ring through a conical titanium transition ring.

1.1.3 Inner Reflector (Fig. 7)

Immediately surrounding the core is the graphite inner reflector. It is 52" long and nominally 2" thick. The inner reflector ring is fabricated of a cylinder of type H4LM graphite. It is fitted with 612 equally spaced 3" square spring loaded graphite pressure pads faced with pyrographite, which bear against the core, offering radial, but not axial restraint (Fig. 8). The lateral supports act as two springs in series; a soft spring to bundle the core, absorb thermal expansion and small accelerations; the hard spring to limit deflections under larger lateral accelerations. One advantage of this system is that under normal operating conditions the core is lightly loaded; under severe load conditions lateral motion can be snubbed with only a small displacement. A second advantage is that the core is allowed free expansion within normal operation, without the excessive restraint of the hard spring. Flow holes run the length of the inner reflector, and it is fitted, at its upper end, with a continuous spring type seal, preventing the inlet flow from reaching the upper core face, but allowing flow access to the shield inlet region. This coolant flow is directed to cool structural parts. Springs at the dome end of the inner reflector load it, and the nozzle end seal, against the nozzle under all conditions of operation.

1.1.4 Outer Reflector (Fig. 9)

The outer reflector is 52" long and nominally 4.619" thick. It is made of beryllium and is fabricated in 12 segments which are bored longitudinally to accept the control drums, shim rods and flow channels. The beryllium segments are held between a circular dome end support barrel and a nozzle end support plate by 12.625" diameter Titanium through bolts. The reflectors have 216 holes to accept .5" diameter aluminum or beryllium shim rods. Inlet coolant flows between the outer reflector and the support barrel. There is no flow past the shim rods (Fig. 10).

The gravity and inertia loads on the core of the outer reflector, control drums, and drum shafts are transmitted through the dome end reflector support ring to the pressure vessel flange.

To support the control drum and its drive shaft, 12 titanium bearing housings,

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FIG. 7
INNER REFLECTOR ASSEMBLY

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FIG. 8
LATERAL SUPPORT

FIG. 9
OUTER REFLECTOR

equally spaced, are mounted on the top support ring. Each contains a ball bearing to provide thrust load capability and radial guidance. The bottom end of the control drum is provided with a ball bearing in the bottom support ring to assure proper positioning and free rotation of the control drum. A nominal clearance of .050" is maintained between the drum and the reflector.

At its top end, the control drum shaft is splined to accept a stop ring and the drum-to-actuator coupling (Fig. 10). The stop ring carries an extended lug whose purpose is to prevent travel of the drum past its "full in" position. During shipment, each housing is fitted from above with a removable locking device. Should the drum attempt to move from its "full in" position, the device will prevent drum motion. The stop ring, coupling, and both shafts are splined with one double tooth, to assure proper angular relation in assembly. The actuator shaft spline is crowned to accommodate for minor misalignment. The actuator shaft is prevented from being inadvertently withdrawn from the coupling by a circumferential split lock ring. A torsion spring is secured between the coupling and bearing housing provides a positive force to rotate the drum to its "in" position in the event of mechanical failure of the shafts.

1.1.4.1 Control Drums

The control drum is shown in Fig. 11. The drum is essentially a beryllium cylinder in an .050" thick aluminum casing. The control element is a .050" thick boral vane 120° in angular coverage. The beryllium core is recessed to accept the control vane. The vane is spaced between the core and the outer sheath by riveted spacer pins. Excepting those places where there is contact between the vane and the support pins, there is free coolant flow around the boral vane.

1.1.5 Reactor Shield

The reactor shield is made up of 682 shield modules which are held in place by upper and lower weldment assemblies. Each module consists of seven capsules, fused between stainless steel tie plates. The capsule itself consists of a borated steel plug, and a variable number of lithium hydride plugs and stainless steel washers in a stainless steel jacket, secured by end plugs (See Fig. 12). The shield plugs are spring loaded within the jacket to allow

FIG. 10
OUTER REFLECTOR (CUTAWAY)

FIG. 11
CONTROL DRUM

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SHIELD CLUSTER

FIG. 12
SHIELD MODULE

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thermal expansion. The upper and lower weldments are domed (See Fig. 13), and the module tie plates and capsule lengths are varied to suit the configuration. Beneath the lower weldment assembly is a perforated plate to which is brazed wire cloth, serving as a core inlet filter. The shield assembly rests on, and is bolted to, the core support plate.

The core is allowed to expand radially against the spring loaded lateral supports. Core longitudinal expansion is toward the dome end, compressing the tie rod springs. The nozzle end flow seal is designed to accept radial expansion with a segmented, polished, graphite structure on the stainless steel nozzle extension.

Flow through the inner reflector is baffled from the top of the core by a ring member, acting as a continuous finger spring to allow radial expansion. A two piece spring seal is used to allow relative motion between the shield and the core support plate.

1.2 OPERATING CHARACTERISTICS - (Mechanical)

1.2.1 Startup - the transient start program is in the process of being refined to provide a more representative flow model. Runs were made of a series single channel model for the heated bleed engine. The results, which are presented in Figs. 14 through 16, are presented to show the general parametric behavior of the plant. The single series model and a 20 channel model are in preparation to describe more accurately the hot bleed engine.

1.2.2 Full Power - in the full power run, a propellant flow of 71.7 lb/sec., leaves the reactor at mixed mean conditions of 4090°R and 550 psia. The total flow distribution is shown in Fig. 17. The core flow is divided between the coolant and tie rod channels. Fig. 18 shows the axial temperature of a nominal fueled element with 6% bypass through the tie rod coolant channel. Fig. 19 shows the temperature distribution in the cross section of a nominal fueled element at a station 46 inches from the inlet. The same element was studied for the condition where its center channel was plugged (Fig. 20). It can be seen that this plugging increases the maximum fuel temperature by 420°R.

1.2.3 Cooldown - the cooldown procedure proposed for the NERVA engine involves a three phase cooling operation.

- 1) Turbopump circulation to assure sufficient flow - approximately 0 - 50 seconds after shutdown.

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(FUSION WELDED) FLAT TIE PLATE

TOP VIEW 5X SIZE

SECTION VIEW 12X SIZE

FIG. 13
SHIELD ASSEMBLY

FIG. 14
POWER & PRESSURE INPUT SCHEDULES

FIG. 15
PRESSURE VS TIME (INLET AND OUTLET)

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FIG. 16
TEMPERATURE VS TIME - CORE INLET,
OUTLET AND REFLECTOR

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NRX-A FLOW CHART

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FIG. 18
AXIAL TEMPERATURE DISTRIBUTION --
NOMINAL ELEMENT

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FIG. 19
 NOMINAL FUEL ELEMENT - MAXIMUM
 FUEL TEMPERATURE

FIG. 20

NOMINAL FUEL ELEMENT (MAXIMUM FUEL TEMPERATURE)

CENTER CHANNEL PLUGGED

- 2) Tank pressure flow 50 - 100 seconds.
- 3) Pulse cooling - 100 seconds to restart.

In step 3, a pulse of coolant is admitted by the engine control system to the reactor whenever any component critical temperature is exceeded. The cooling pulse lasts until the average core temperature has decreased by 300°R . The propellant is assumed to enter the nozzle at 30 psia, 41.3°R , and .1 quality. The critical component temperatures are:

Core	2500°R
Tie Rod Steel Sleeve	2000°R
Tie Rod	1700°R
Al Support Plate	900°R
Pressure Vessel	900°R
Graphite Sleeve	2000°R
Graphite Reflector	2000°R
Beryllium Reflector	900°R
Nozzle	1500°R

A transient analysis was run to determine the requirements on a pulse cooling system following 20 minute operation at 1120 MW. It was assumed that 520 pounds of coolant were used in Step 1 (turbopump operations) and 200 pounds were used in Step 2 (pump head flow). Steps 1 and 2 provided dissipation for 21% of after shutdown energy. The pulse cooling step was carried out for 35 pulses (approximately 10^4 seconds after shutdown). The first pulse was initiated by high temperature in the tie rod sleeve (111 seconds after shutdown), lasting 20 seconds, the second in the sleeve (165 seconds after shutdown, lasting 16 seconds) and all subsequent pulses due to tie rod temperature. The summary of the cooldown calculations is given in Table 2. It can be seen that 66% of the total shutdown energy is removed by the first 35 pulses, using a flow of 2456 pounds. It is estimated that 1380 pounds additional will suffice to dissipate the remaining energy.

1.2.4 Partial Power Performance

The partial power performance of the reactor core and core support channels was investigated to determine partial load flow requirements for adequate tie rod cooling.

SUMMARY OF COOLDOWN CALCULATIONS

TABLE II

Pulse	Time after shutdown sec (flow is turned on)	Time off (before pulse)	Time on (length of pulse)	Average flow rate lb/sec	Propellant used per pulse lb	Average Propellant Exit temp. °R	Total Propellant usage lb	Average Thrust lb
Initial turbine cooldown (estimated)	0	0	50	10.4	520	3000	520	—
Initial cooldown using tank pressure	50	0	50	4.09	204.6	1923	725	2110
1	111.4	11.4	20	4.15	82.9	1937	808	2160
2	164.7	33.3	16	3.95	63.1	1980	871	2040
3	183.1	2.4	13	4.33	56.2	1897	927	2230
4	249.5	53.4	13	4.03	52.4	1888	980	2070
5	298.4	35.9	12	4.12	49.4	1874	1029	2100
6	345.9	35.4	12	4.20	50.3	1849	1079	2130
7	414.7	56.9	12	4.16	49.9	1836	1129	2100
8	487.8	61.0	11	4.15	45.7	1821	1175	2080
9	560.4	61.7	11	4.20	46.2	1801	1221	2100
10	645.3	73.9	11	4.20	46.3	1785	1267	2100
11	739.3	83.0	11	4.22	46.4	1769	1314	2100
12	853.7	103.5	11	4.18	46.0	1768	1360	2080
13	967.7	103.0	11	4.21	46.3	1762	1406	2080
14	1087.8	109.2	11	4.25	46.8	1744	1453	2100

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Pulse	Time after shutdown sec (flow is turned on)	Time off (before pulse)	Time on (length of pulse)	Average flow rate lb/sec	Propellant used per pulse lb	Average Propellant Exit Temp. °R	Total Propellant usage lb	Average Thrust lb
15	1232.6	133.8	11	4.25	46.7	1727	1500	2080
16	1414.5	170.9	11	4.18	46.0	1731	1546	2060
17	1588.0	162.4	11	4.20	46.2	1734	1592	2000
18	1741.5	142.6	10	4.27	42.7	1712	1635	2080
19	1963.2	211.7	11	4.23	46.5	1705	1681	2060
20	2224.6	250.3	11	4.18	46.0	1711	1727	2040
21	2455.3	219.7	11	4.21	46.3	1714	1773	2050
22	2693.1	226.8	11	4.22	46.4	1707	1820	2060
23	2959.7	255.6	11	4.23	46.6	1693	1866	2060
24	3288.5	317.8	11	4.18	46.0	1692	1912	2020
25	3602	302.5	11	4.20	46.2	1690	1959	2040
26	3957	343.8	11	4.18	46.0	1688	2005	2020
27	4348	380.6	11	4.17	45.8	1686	2050	2020
28	4770	410.2	11	4.16	45.8	1681	2096	2010
29	5262	481.5	12	4.15	49.8	1678	2146	2000
30	5883	609.1	12	4.09	49.0	1685	2195	1980
31	6453	557.7	12	4.09	49.1	1681	2244	1970
32	7206	741.3	12	4.03	52.4	1683	2296	1950
33	8064	844.4	13	4.00	52.0	1684	2349	1940

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Pulse	Time after shutdown sec (flow is turned on)	Time off (before pulse)	Time on (length of pulse)	Average flow rate lb/sec	Propellant used per pulse lb	Average Propellant Exit temp. °R	Total Propellant usage lb	Average Thrust lb
34	9005	928.1	13	3.98	51.7	1682	2400	1930
35	10118	1100.0	14	3.96	55.4	1678	2456	1910
36 - 58 Estimated	10 ⁶	—	—	—	60	1600	3836	—

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The problem exists at low power because the heat conduction from the fuel to the tie rod channel coolant is essentially independent of power, and dependent solely on core temperature. If flow is maintained proportional to power, core temperature will be essentially independent of power and heat input to the tie rod channels remains about the same. However, the tie rod channel coolant flow is also proportionally reduced with power, so that tie rod temperature must rise. The design goal for tie rod temperature in normal operation is 800°R , allowing temperature deviations of 200°R for transient conditions.

The reactor core was represented by two average channels, the fuel and tie rod flows. The calculation results are plotted in Figs 21 & 22 and show the calculated tie rod channel exit and mixed core outlet temperatures as a function of power level. It can be seen from Fig. 22, that operation at 50% power, (50% flow) maintaining mixed core outlet temperature at 4090°F , results in a tie rod channel exit temperature of 1240°R (extrapolated). However, it is possible to maintain tie rod temperature at the design goal of 800°R by reducing the flow to only 79%, resulting in a mixed core outlet temperature of 2750°R , with a consequent decrease in specific impulse of 18%. Thus, it is possible to keep the tie rod within its safe operating temperatures by programming flow, non-proportional with power, and accepting the accompanying loss of impulse.

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FIG. 21
TIE ROD CHANNEL EXIT TEMPERATURE VS CORE FLOW

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FIG. 22
 MIXED CORE EXIT TEMPERATURE VS CORE FLOW

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1.2.5 Hot Channel Factor

In order to evaluate the degree of conservatism in design of a reactor, it is necessary to examine the effects of all possible deviations from the nominal (or specified) values. Such deviations include manufacturing tolerances, experimental errors, and operational variation. A statistical treatment of these variations, or of their affect on the operational characteristics of the reactor, yields the "hot channel factor". When this factor is applied to an average temperature difference the worst or "hot channel" temperature difference is approximated.

A preliminary statistical treatment of the NERVA core is presented in WANL-TNR-057, "A Method of Statistical Analysis of the Effects of Manufacturing Tolerances and other Deviations on Operating Temperatures of the NERVA B-4 Core", April, 1962. The model chosen for this study considered a coolant flow rate of 70 lb/sec., with an inlet temperature of 130°R and a nozzle condition specified as 4090°R at 550 psia. A bypass flow of 6% was assumed to permit cooling of the fuel element assembly support structure. To simulate the power flattening, a binominal distribution was approximated. In order to achieve a nearly constant coolant outlet temperature, the coolant channels are orificed according to their power level. With this arrangement, the maximum fuel temperature does not differ appreciably between channels having different power densities. This is due to the fact that the coolant temperature rise represents 75% of the total temperature and maximum fuel temperature. Since the coolant temperature rise is controlled by orificing, the fuel temperatures are very similar across the core.

In order to evaluate the hot channel factor, the study determined errors and confidence levels) for horizontal and axial power distributions, hole diameter variation, clad thickness variation, channel orientation, outside dimensions, orifice coefficients, fuel mixture proportions, compacted density, fuel enrichment, plenum factor, irradiation factor, roughness factor, H-T conversion, fuel conductivity, clad conductivity, and heat transfer coefficient. The results of these possible errors were then multiplied cumulatively and also combined using the statistical approach.

It was observed that the cumulative multiplication approach yielded a hot spot of 7892°R at rated power, while the statistical approach showed that only 1 channel w

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exceed 5900°R , its most probable value lying below 6000°R . Other conclusions of interest were:

1. Those factors which affect coolant temperature rise strongly are the factors which also affect fuel temperatures.
2. A comparison of the factors which affected fuel temperature rise, listed in order of their affect:

Horizontal Power Distribution	41.8%
Channel Orientation	19.5
Orifice Coefficients	9.4
Roughness Change	8.8
Plenum Factor	6.3
Others	14.2
	<hr/>
	100%

3. The statistical approach provides a numerical evaluation of the risk involved in assuming all deviations do not occur simultaneously.

This approach, though preliminary, provides a useful statistical approach, resulting in a more practical design criterion. Further work will be performed on this method, particularly in testing the normality of distribution of each deviation.

1.3 OPERATING CHARASTERISTICS - (Nuclear)

1.3.1 Reactivity Characteristics

Initial Reactivity, Room Temperature	\$ 4.20
Control Drum Span, Room Temperature	7.50
Control Drum Span, Operating Temperature	7.30
Temperature Defect (Cold to Operating)	-1.40
Hydrogen Worth (Operating Condition)	+ 2.50
Net Reactivity Change (Ambient to Operating)	+1.1
Reactivity Shim Available at Proof Test Critical	± 4.0
Neutron Lifetime	3×10^{-5} sec.

Reactivity lost in 30 min. Operation	-1.50
Minimum Shutdown Reactivity, All Conditions	-2.00
Enriched U Reactivity Coefficient, core	+ .27%
Graphite Reactivity Coefficient, core	+ .58%
Niobium Coefficient	- .06%
Beta	.0078

1.3.2 Core Loading

The NERVA reactor fuel element concentrations and reactivity shims must be chosen to satisfy the following conditions:

1. The radial power ratio at operating conditions must be within 0.8 - 1.2 with the same angular setting for all drums, and the most adverse setting for all drums.
2. The cold critical drum position is based on providing a minimum holddown capability of \$2 under any operating conditions.
3. The reactor must be capable of restart at any time after 30 minutes of full power operation. It is assumed there will be a \$1.50 loss in reactivity during operation and that 30 ¢ reactivity will be necessary to assure restart.
4. Maximum fuel concentration is 440 mg/cc, so as to stay within the range of LASL experience with this material.
5. Fuel loading is based on a core with all shim holes open and drums set "full-out".

The preliminary loading specifications for NRX-A-1 are:

<u>Zone</u>	<u>Fuel Concentration mg/cc</u>	<u>Fuel Loading kg</u>
1	417	160.26
2	314	11.69
3	213	6.43
4	172	5.22
5	143	
Total		183.6 kg

Zone arrangement is shown in Fig. 23.

CORE LOADING SCHEME

FIG. 23

Region	Loading Designation	Loading, Mg U/cm^3	Number of Elements of this Loading
1	9	417	1296
2	8	314	126
3	7	213	102
4	6	172	66
5	5	143	18 Full 18 12 Hole Elemts. 24 16 Hole Elemts.

NOTE: All undesignated fueled elements are loading 9.

NOTE:
THIS DRAWING SHOWS CORE OUTER CLUSTER
CROSS SECTION LOADING IN THE DIRECTION
OF PROPELLANT FLOW

1.3.3 Median Fission Energy

Two curves, Fig. 24 and 25, show median fission energy as a function of core radius. The two "hot case" curves show the effect of introducing boron as the drum poison, resulting in a capture of thermal neutrons, with a consequent hardening of the spectrum and an increase in E_m .

2.0 REACTOR CONTROL SYSTEM

2.1 Introduction

A schematic of the proposed reference reactor control system is shown in Fig. 26. The proposed control system, in conjunction with the engine programmer, provides control during the startup (initial) phase, flow initiation and subsequent power rise to full power, shutdown, and restart. The functional operations performed by the system are described below.

2.2 Typical Operation (Initial Startup)

2.2.1 Initial Nuclear Startup

Prior to startup all relays shown on the diagram are de-energized and the contacts are shown for the de-energized state.

Nuclear startup is initiated in the engine programmer by supplying a negative D.C. voltage (SUS) of a specified level to the control system. This signal is fed to an integrating amplifier 2 which produces a 1 sec. period output signal corresponding to a linear change in the demanded logarithm of the neutron density (LnND_1). The demanded log neutron density signal is compared to the measured log neutron density forming the error signal $\epsilon_{\log}$.

The error signal is fed to a nonlinear network having the approximate input-output response shown on the diagram. The output signal of the nonlinear network is proportional to the demanded control drum velocity, $\dot{\theta}_D$. The demanded control drum velocity is integrated by amplifier 3 forming the demanded control drum position signal which is fed to the control drum actuator systems. The actuators will respond to establish the drum position at the demanded position. During the initial phase of the startup, the LnNM signal will lag the LnND_1 signal due to the effects of the reactor being subcritical, the control drums being in a low gain region of operation, the LnND_1 ramp signal demanding a relatively short reactor period, and the time constants that are associated with the nuclear startup instrumentation.

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FIG. 24

MEDIAN ENERGY OF FISSION VS CORE RADIUS -
NO DRUMS

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MEDIAN ENERGY OF FISSION (E_m) AS FUNCTION OF RADIAL CORE COORDINATE

FIG. 25.

MEDIAN ENERGY OF FISSION VS CORE RADIUS
WITH DRUMS AT 90°

REVISIONS			
REV	DESCRIPTION	DATE	APPROVED
A	ADDED COMPONENT VALUES & GENERAL CHANGES	3/28/54	
B	GENERAL CHANGES	3/28/54	

REV	DATE	DESCRIPTION	DATE	DESCRIPTION	DATE	DESCRIPTION
1	3/28/54	INITIAL DESIGN	3/28/54	REVISION 1	3/28/54	REVISION 2
2	3/28/54	REVISION 1	3/28/54	REVISION 2	3/28/54	REVISION 3
3	3/28/54	REVISION 2	3/28/54	REVISION 3	3/28/54	REVISION 4
4	3/28/54	REVISION 3	3/28/54	REVISION 4	3/28/54	REVISION 5
5	3/28/54	REVISION 4	3/28/54	REVISION 5	3/28/54	REVISION 6
6	3/28/54	REVISION 5	3/28/54	REVISION 6	3/28/54	REVISION 7
7	3/28/54	REVISION 6	3/28/54	REVISION 7	3/28/54	REVISION 8
8	3/28/54	REVISION 7	3/28/54	REVISION 8	3/28/54	REVISION 9
9	3/28/54	REVISION 8	3/28/54	REVISION 9	3/28/54	REVISION 10
10	3/28/54	REVISION 9	3/28/54	REVISION 10	3/28/54	REVISION 11
11	3/28/54	REVISION 10	3/28/54	REVISION 11	3/28/54	REVISION 12
12	3/28/54	REVISION 11	3/28/54	REVISION 12	3/28/54	REVISION 13
13	3/28/54	REVISION 12	3/28/54	REVISION 13	3/28/54	REVISION 14
14	3/28/54	REVISION 13	3/28/54	REVISION 14	3/28/54	REVISION 15
15	3/28/54	REVISION 14	3/28/54	REVISION 15	3/28/54	REVISION 16
16	3/28/54	REVISION 15	3/28/54	REVISION 16	3/28/54	REVISION 17
17	3/28/54	REVISION 16	3/28/54	REVISION 17	3/28/54	REVISION 18
18	3/28/54	REVISION 17	3/28/54	REVISION 18	3/28/54	REVISION 19
19	3/28/54	REVISION 18	3/28/54	REVISION 19	3/28/54	REVISION 20
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21	3/28/54	REVISION 20	3/28/54	REVISION 21	3/28/54	REVISION 22
22	3/28/54	REVISION 21	3/28/54	REVISION 22	3/28/54	REVISION 23
23	3/28/54	REVISION 22	3/28/54	REVISION 23	3/28/54	REVISION 24
24	3/28/54	REVISION 23	3/28/54	REVISION 24	3/28/54	REVISION 25
25	3/28/54	REVISION 24	3/28/54	REVISION 25	3/28/54	REVISION 26
26	3/28/54	REVISION 25	3/28/54	REVISION 26	3/28/54	REVISION 27
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37	3/28/54	REVISION 36	3/28/54	REVISION 37	3/28/54	REVISION 38
38	3/28/54	REVISION 37	3/28/54	REVISION 38	3/28/54	REVISION 39
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40	3/28/54	REVISION 39	3/28/54	REVISION 40	3/28/54	REVISION 41
41	3/28/54	REVISION 40	3/28/54	REVISION 41	3/28/54	REVISION 42
42	3/28/54	REVISION 41	3/28/54	REVISION 42	3/28/54	REVISION 43
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44	3/28/54	REVISION 43	3/28/54	REVISION 44	3/28/54	REVISION 45
45	3/28/54	REVISION 44	3/28/54	REVISION 45	3/28/54	REVISION 46
46	3/28/54	REVISION 45	3/28/54	REVISION 46	3/28/54	REVISION 47
47	3/28/54	REVISION 46	3/28/54	REVISION 47	3/28/54	REVISION 48
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53	3/28/54	REVISION 52	3/28/54	REVISION 53	3/28/54	REVISION 54
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55	3/28/54	REVISION 54	3/28/54	REVISION 55	3/28/54	REVISION 56
56	3/28/54	REVISION 55	3/28/54	REVISION 56	3/28/54	REVISION 57
57	3/28/54	REVISION 56	3/28/54	REVISION 57	3/28/54	REVISION 58
58	3/28/54	REVISION 57	3/28/54	REVISION 58	3/28/54	REVISION 59
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93	3/28/54	REVISION 92	3/28/54	REVISION 93	3/28/54	REVISION 94
94	3/28/54	REVISION 93	3/28/54	REVISION 94	3/28/54	REVISION 95
95	3/28/54	REVISION 94	3/28/54	REVISION 95	3/28/54	REVISION 96
96	3/28/54	REVISION 95	3/28/54	REVISION 96	3/28/54	REVISION 97
97	3/28/54	REVISION 96	3/28/54	REVISION 97	3/28/54	REVISION 98
98	3/28/54	REVISION 97	3/28/54	REVISION 98	3/28/54	REVISION 99
99	3/28/54	REVISION 98	3/28/54	REVISION 99	3/28/54	REVISION 100

FIGURE 26
REACTOR REFERENCE CONTROL SYSTEM

If the LnND_1 signal exceeds the LnNM signal by the amount of the bias signal added at the feedback loop of amplifier 2, the diode around this amplifier will conduct, limiting the LnND_1 signal. During the initial subcritical portion of the startup transient, the diode has the effect of maintaining the LnND_1 ramp signal close to the value that the subcritical reactor is capable of producing. After criticality is achieved, the reactor log neutron density will rise to the demanded level and follow the demand to a preset low (.1 to 1% of F.P.) power level, preparing the core for flow initiation.

When the control drums have rotated out beyond some minimum position, possibly 20° , and the measured log neutron density has exceeded the preset power, relay A is energized. A-1 opens holding the LnND_1 signal at the ready power. The coincidence trigger circuit also providing a flow permissive signal to the engine programmer. The reactor is controlled to the ready power until control is transferred from the nuclear startup control to the power range control.

2.2.2 Power Range Operation (Startup)

Control of the reactor from the ready power to full power is accomplished by a separate part of the controller. The input of the temperature comparing amplifier 5 is tied to the summing point of the LnNM and LnND_2 signal by means of contact B-5. This effectively presets the LnND_2 signal to differ from the LnNM signal by the temperature error. $T_C M - T_C D$ at the instant of transfer. Present analysis indicates that the error between $T_C M$ and $T_C D$ at the time of transfer will not introduce deleterious transients.

Coolant flow will be supplied by the tank pressure during the chilldown and while the reactor is holding at the ready power level. The turbine pump can now be started to initiate the bootstrap operations.

When the measured chamber pressure, $P_C M$, reaches a specified level, relay B is energized. The level is chosen to ensure that a bootstrap has been initiated. Contact B-4 closes, removing the LnND_1 signal, B-1 opens, removing the $\dot{\theta}_D$ signal from amplifier 3, B-3 opens, adding proportional gain to amplifier 3 and B-2 closes, connecting the power range controller to amplifier 3.

The $P_c M$ signal is used to produce the chamber temperature demand signal, $T_c D$, in accordance with the relationship $T_c D = f(P_c M)$. The particular function to be used will depend primarily upon the reactor core design limitation. A function was chosen which will protect the tie rods in the reactor core.

The function which has been used for the initial simulator runs consists of two straight line segments defined by the following equations:

$$(1) \quad T_c D = 750 = 7.80 P_c M \quad \text{When } 0 \leq P_c M \leq 218$$

and

$$(2) \quad T_c D = 2450 = 4.94 P_c M \quad \text{When } 218 \leq P_c M \leq 550$$

where

$T_c D$ = Chamber temperature demand, $^{\circ}R$

$P_c M$ = Measured chamber pressure, psi

During power ($T_c D$) increase the $T_c D$ signal is compared to the measured chamber temperature signal, $T M$, and the error is fed through the proportional and integral gain to amplifier 5 to form $LnND_2$. $LnND_2$ is compared to $LnNM$ and the resulting error is fed through the proportional and integral gain of amplifier 3 to form θ_D . The actuators will respond to this θ_D signal to control the power level in accordance with the $T_c D$ program.

Limiters are provided in the $f(P_c M)$ function circuit to limit the chamber temperature demand to $4100^{\circ}R$, and to limit any excess temperature rates that might occur from a rapid rate of change of $P_c M$. The temperature demand, $T_c D$, is permitted to increase at $150^{\circ}R/sec.$ and to decrease at $300^{\circ}R/sec.$ by means of the nonlinear circuitry associated with amplifier 4.

2.2.3 Power Range Operation (Shutdown)

Reactor shutdown is accomplished by the engine programmer reducing the chamber pressure demand. The $T_c D$ signal is programmed down in accordance with the $f(P_c M)$ function.

During the decrease in power and $T_c D$ the lead circuit of amplifier 5 is activated by the diode conduction. The total $T_c D$ signal is now composed of the sum of the $T_c D$ signal from amplifier 4 and the output of the lead circuit associated with amplifier 5. This total

$T_c D$ signal is compared to $T_c M$ in amplifier 5 to produce the $LnND_2$ signal. The lead circuit is necessary for tie rod protection due to the inherent lags in the system and the sensing elements.

The criteria for tie rod protection is that the temperature of the coolant flow from the tie rods, and thus the temperature of the tie rods, must remain below some critical value at all times.

Power is programmed down in the power range system until $P_c M$ drops below a specified level de-energizing relay B. It is now permissible to close the turbine-power control valve and the tank shutoff valve. Control is now transferred to the nuclear startup portion of the control system.

Since $LnND_1$ is now demanding the original source level, and $LnNM$ is relatively high, a large error signal is produced which causes the drums to rotate at $45^\circ/\text{sec.}$ to the "full-in" (minimum reactivity) position.

2.3 Restart

If a restart is initiated prior to control transfer to the nuclear startup system, the sequence of operations to full power are the same as the initial power range operation.

If a restart is initiated after the control transfer, the control drums will be at the "full-in" position and relay A will be de-energized.

When the engine programmer supplies a startup (restart) signal, the $LnND_1$ signal will increase at its maximum rate subjected to the limit on ϵ_{log} . The control drums will not rotate out until the $LnND_1$ signal reaches the decaying $LnNM$ signal.

If the measured neutron density is less than the ready power neutron density when restart is initiated, the drums will rotate out (when $LnND_1 > LnNM$) to bring the reactor to a critical condition the ready power. The flow permissive signal is now given to the engine programmer.

If the measured neutron density is greater than the ready power neutron density when restart is initiated, the drums will rotate out (when $LnND_1 > LnNM$) to bring the reactor to a critical condition at some level greater than 1%. The flow permissive will be given to the engine programmer when the drums reach the 20° position.

Control is again transferred to the power range when $P_c M$ reaches a specified level.

2.4 Cutback Operations

Three cutback circuits are included in the control system for safety. The control drums are run in when the measured neutron density, reactor period, or the measured chamber temperature exceed specified limits. The cutbacks exist as long as the limits are exceeded. The magnitude of the cutback signal is proportional to the excess of the variable above the limit, giving smooth control drum transients during cutback operation.

2.5 Nuclear Instrumentation Switching

Three ranges of nuclear instrumentation, each range having a redundancy of three channels, are shown on the diagram. In each range, the three channels are auctioneered for the highest signal. In order to avoid switching transients while changing ranges, a zener diode fires and holds the output of that range at the zener voltage. Each succeeding stage acts in the same manner and adds its output voltage to the lower stages.

2.6 Cooldown Flow

In order to maintain specific component temperatures within specified limits during decay heat operation, a decay heat cooling program is used. When any of the component temperature exceed specified limits, the trigger circuit is energized providing a cooldown flow initiation signal to the engine programmer. Since this circuit is effective during all phases of reactor operation, it is possible to operate the reactor for extended periods at the ready power level.

2.7 Component Requirements

The preliminary operational concept of the reactor control system is presented in the previous section. Detailed specifications for the components of the system have not yet been developed. The drum actuator, however, is capable of independent study, and has been specified.

Considerable work must still be done to evaluate the actuators considered, and to determine the compatibility of the unit specified with the reactor control system. This work is in progress, and will be reported in a later issue of this report.

2.7.1 Actuator Requirements

2.7.1.1 General

1. High reliability, fail safe where possible.
2. Light weight, compatible with requirements.
3. Zero maintenance through engine life.

Load Characteristics

1. Total moment of inertia
(drum, coupling, and extension shaft) .24 in lb sec²
2. Load friction (max.) torque
Static .32 in lb
Dynamic 25 in lb.
3. Torsional stiffness drum - actuator coupling 80,000 in lb per radian
4. Backlash drum-actuator (max.) $\pm .25^\circ$
5. Spring load (determined by scram requirements)

2.7.1.2 Performance Characteristics

1. Travel
Max. $0^\circ \pm 1^\circ$ to $180^\circ \pm 1^\circ$
Working $15^\circ \pm 1^\circ$ to $165^\circ \pm 1^\circ$

2. Scram Operation

A mechanism shall be provided to move the control drum in the direction of scram (toward the 0° position) in the event of electrical or pressure system failure, or upon receipt of a fast shutdown command. This may be accomplished with a spring on the front of drum, if desired.

Turn around time shall be .05 seconds. It is defined as the time from failure of electrical or pressure system or after receipt of a fast shutdown command to the time that the drum has reversed direction and achieved an initial velocity in the direction of the 0° position.

Soft stops shall be used to snub the drum in the $0^\circ - 15^\circ$ and $165^\circ - 180^\circ$ positions. Maximum rebound from 0° position shall be limited to one half of the angle from which scram was initiated, or 20° , whichever is greater.

Allowable drum deceleration (maximum) 2000 radians per sec²

3. Positioning Accuracy

Static position resolution at output shaft (maximum)	$\pm 0.2^{\circ}$
Linearity of position transducer	$\pm 4\%$ full travel

4. Dynamic Response

Transient response to an 18° step input	
Rise time to 62.5% of step command (maximum)	0.055 sec.
Settling time to within $\pm 0.9\%$ (maximum)	0.15 sec.
Allowable overshoot (maximum)	6°
Frequency response to $\pm 2^{\circ}$ input (without velocity limits)	
Phase shift at 8 cycles per second (maximum)	90°
Phase shift at 20 cycles per second (maximum)	100°
Position bobble during a 3° per second input command (maximum)	$\pm 0.5^{\circ}$

5. Velocity (maximum)

Direction of decreasing reactivity	
Normal Operation	45° per sec.
Scram	300° per sec.
Direction of increasing reactivity	

2.7.1.3 Interlocks

A positive acting, electrically initiated mechanical lock will be provided on the actuator which will prevent drum rotation until the proper signal is received, and will fail safe with the drum in the "full-in" position.

3.0 POISON SYSTEMS

3.1 Transportation Poison System

It is planned to assemble the NERVA reactor, which includes the fuel elements, the graphite reflector barrel, the beryllium reflector and control drums, and the top support plate at the Westinghouse Astronuclear Laboratory (WANL) at Large, Pennsylvania. The assembled reactor will be loaded into a shipping container and shipped by truck to the

Westinghouse Astronuclear Experimental Facility (WANEF) in Waltz Mills, Pennsylvania. After a set of proof test critical experiments, it will be returned by truck to WANL and subsequently loaded on a flat car for rail shipment to Cape Canaveral, Florida. Shipment by rail will make use of a special train. Shipments by truck will be escorted.

3.1.1 Shipping Container

A shipping container is provided for the core which is designed to minimize the possibility of water flooding. In the event that the core should be flooded, a poison system is installed in the core to maintain its sub-criticality. Hazards resulting from core shipment are the subject of a separate report, WANL-TME-202.

As shown, Figure 27 the shipping container is double walled, consisting of an inner container and an outer container. The reactor is rigidly mounted in the inner container in a manner which simulates its actual support in the reactor vessel of the completely assembled reactor. The core axis is vertical and the top support plate is at the upper end. In the event several cores must be stored in the same building, a 10 foot center to center distance will be observed.

3.1.1.1 Inner Shipping Container

The inner shipping container is mounted in the outer container through a suspension system designed to protect the reactor during normal transportation and handling, from shock and vibration. Design objectives are that the magnitude of the maximum shock load that the container permits the core to experience shall not exceed 25 g vertically and 1.8 g horizontally when subjected to 6" end, side and corner drop tests and a longitudinal impact test. The container is fabricated of 1/2 inch stainless steel. The flanged joint is provided with a seal which will withstand a test pressure of 20 psi, although during normal use pressure will be limited to 7.5 psi by a pressure relief valve. During shipment the container will be pressurized to 5 psi with nitrogen. The inner container is mounted in a manner which permits it to be easily removed from the outer container.

3.1.1.2 Outer Shipping Container

The outer container encloses the suspension system and provides additional containment for the reactor. It is supported by, and integrated with, a base structure which

FIG. 27
SHIPPING CONTAINER

is approximately 10 ft. by 10 ft. The container will be bolted to the transport vehicle through this base structure.

Excluding the base structure, the container is 74 inches in diameter and approximately 130 inches high. It is fabricated of 1/4 inch carbon steel. The flanged joint is gasketed and will withstand a test pressure of 20 psi although during normal use pressure will be limited to 7.5 psi by a pressure relief valve. During shipment, the container will be pressurized to 5 psi with nitrogen. Suitable lifting rings, eyes or lugs are provided for loading and unloading the container by crane. Each individual lifting ring or eye will be capable of supporting the entire weight of the loaded container.

Provisions have been included in the design of the base structure for use of fork lift trucks, which may be used only when the container is empty.

3.1.2 Shipping Poison System

3.1.2.1 Requirements:

- a. The system must be installed or removed without damage to the core.
- b. The system must be capable of maintaining the reactor subcritical:
 1. when flooded, regardless of control drum position.
 2. when all drums are rotated to their full out position, no flooding.
- c. The poison employed to provide drum override capability 2.1.2.1.b.1 must be capable of being removed from the core through the reactor nozzle.
- d. The poison must be retained within the core under the design ground environment (shock and vibration) and fire.
- e. There must be no trace of the poison left in the core after withdrawal.

3.1.2.2 The Poison System

It is presently planned to use poison tubes which are installed in the reactor from the nozzle end, retained by a positive mechanical means. The poison tubes are 0.05" I.D. stainless steel tubes filled with natural boron carbide to an effective density of 80% theoretical. The tubes have a 0.0075 inch wall thickness and are covered with a 0.0075 inch sheath of polyethylene, and are 53.5 inches long. A small flange, at the bottom of the tube, prevents it from passing through the element. It is presently conceived that a

two part retention scheme will be employed. The inner disk section, approximately 8 inches in diameter, will allow the drum override poison to remain in place when the outer retention ring and poison is removed.

3.1.2.3 Nuclear Worth of Poison

The multiplication constants for a water flooded NERVA reactor were determined for the unpoisoned condition ($k = 1.5$) and for various amounts of B-10 homogeneously distributed through the core. A four group diffusion theory model was used for the calculation with a self-shielding factor of 2 applied to the thermal cross section of Boron to account for the heterogeneity of the tubes. In order to reduce the multiplication constant to .8 required the addition of 3000 grams of B-10. The present design introduced the poison in the form of 20.4 Kg of natural B_4C in 6360 tubes. The reactivity value of the poison in the wet reactor is \$75.

The value of the poison in the dry reactor may be inferred from LASL experiments in KIWI-B-4, which showed boron in a 1/8" tube to be worth about \$2/Kg near the core center, and hydrogen to be worth \$7.5/Kg. The combination of the Boron carbide and the polyethylene sheath yield a net poison reactivity of \$28. This value will be confirmed in the proof test critical experiments. Calculations indicated that 950 poison tubes, in the central 8 inches of the reactor, would provide drum override (\$9.50) capability.

3.2 In-Flight Poison System

The purpose of the in-flight system is to prevent the reactor from going critical in the event of a malfunction prior to that time when flight conditions dictate that nuclear operation can safely be started.

3.2.1 Requirements

- a. The installed poison must be capable of maintaining the reactor subcritical in a submerged condition.
- b. The system must be able to retain the poison under the conditions of powered flight (yet to be determined), and ground handling.
- c. The system must be able to eject or otherwise remove all the poison from

the core under the conditions of powered flight, within * seconds of receipt of the "poison-out signal".

- d. Following removal of the poison system, no obstacle must remain in the path of the coolant channels, or the main nozzle. Also, the poison removal equipment must be so mounted as to be separated with the second stage, or to clear the (2nd. stage - N/S) adapter on separation.
- e. The poison must stay with the core in the event of fire or mechanical damage.
- f. The removal of the poison must not compromise the mechanical integrity of the reactor core.
- g. The system must be ground safe, and must be interlocked against operation by at least two unrelated events.

3.2.2 System Design

The most promising concept for an in-flight-poison system, currently being pursued on a half scale planar mock-up, is similar to the shipping poison system. It employs 5200 poison wires in fuel coolant channels. These wires are secured to a pull head in the inlet of the nozzle. It is envisioned that the system would be armed on closure of a vacuum sensitive baroswitch (to achieve the ground safe protection), and activated on receipt of a signal from the engine programmer. At this time, the pull head would be withdrawn, at a predetermined rate, toward the second stage, pulling the poison wires out of the core.

3.2.3 Alternate Designs

In order to eliminate the need for a device to withdraw the wires (possibly a five foot distance) other concepts were considered, and will continue to be examined. A trapped gas was discarded on the basis that a fire or mechanical damage would certainly rupture or melt any diaphragm or seals containing the material. The same was true of liquid poison. Pellets were considered, held in place by shear plugs, fusible plugs, and pyrotechnic blowout devices. The shear plug was discarded because of its poor reproducibility and because of the low shock environment available. Fusible plugs were felt to be prone to melting in the fire environment. Pyrotechnic plugs have not been explored but would

* Not yet specified.

certainly present a development problem, and would require a considerable harness. Pellets are subject to packing, which might jam them in place, even should the "stopper" be removed. However, the flight vibrations would probably aid in shaking them loose. Other schemes which were devised were discarded because they could not be withdrawn through the 8" nozzle throat.

3.3 Ground Handling Sequence - (Atlantic Missile Range)

A tentative ground handling sequence for the operations of the Atlantic Missile Range is as follows, starting with arrival of the reactor, in its shipping container, at the Nuclear Assembly Building:

1. Remove reactor from shipping container.
- ✓ 2. Check drum locks and shipping poison system.
3. Place in trunnion cradle, inspect for shipping damage.
4. Lock in upright condition, remove drum lock no. 1, test (by hand) drum for free rotation and return spring action. Replace lock and repeat for all drums.
5. Perform any preliminary assembly operations on reactor.
6. Install reactor in pressure vessel. Remove drum locks, install head and actuators.
7. Invert reactor. Remove outer retention ring.
8. Remove poison wires from outer region.
9. Immediately before performing critical experiment, remove inner retention plate and all wires.
10. Perform critical experiment.
11. If any modifications to reactor are required, during or after critical, cadmium poison strips must be installed between the inner and outer reflector.
12. When reactor is in satisfactory condition, reactor is positioned bottom-up, in-flight poison system is installed, and nozzle is installed.
13. Reactor is attached to hydrogen tank, gimbal actuators, and other equipment installed and all connections made.
14. Nuclear stage is moved to vertical assembly building and mated to chemical stage.

4.0 DESTRUCT SYSTEMS

4.1 Re-Entry Physics

4.1.1 Introduction

In considering the NERVA operation one must recognize the reactor may be destroyed, by accident or intent, over some area where ground deposition of the reactor debris can create a hazard. The evaluation of this hazard must involve an integrated study of the many factors involved, including:

1. Trajectory
2. Reactor operating history
3. Time, altitude and vehicle orientation at destruct event
4. Nature of destruct mechanism
 - a. Are particles created? If so, what are sizes, space orientation, and velocities?
 - b. If fission products are removed by thermal action or by reactor excursion, what percentage leave the core as gases? To what extent is the core weakened?
 - c. If the core is weakened, at what altitudes of re-entry does the core fragment, and what size particles are created?
5. Particle Consideration
 - a. It has been tentatively determined that particles 25 microns or smaller will be retained in the atmosphere for appreciable lengths of time. What percentage of particles are within this size range?
 - b. What are the velocities and re-entry angles of the re-entering particles?
 - c. What is the effect of particle shape and surface condition?
 - d. What is the rate of burnup of particles on re-entry?
 - e. From the fission products history, what is the particle source strength?

f. What is the rate of loss of fission product during re-entry?

6. Meteorological Conditions

Finally, it is possible to postulate the pattern of deposition of these particles, and to calculate the doses from the distributed source, ecological uptake and reconcentration. Of the total problem, only a part of the particle dynamic problem has been studied and is reported herein. (The fission product inventory work is reported elsewhere.)

4.1.2 Analytical Program - Particle Dynamics

In undertaking the analytical portion of this program, it was necessary to make some assumption regarding the shape of the particles re-entering the earth's atmosphere. It was assumed that these particles would be spherical to simplify the mathematical treatment. Actually, this assumption leads to conservative results since the surface area-to-volume ratio is smaller for a sphere than for any other shape. It may, therefore, be expected that actual particles, having the same maximum dimension as the sphere, will burn more rapidly upon re-entry because of the presence of greater surface areas. It was further assumed that this spherical particle was spinning because of residual angular momentum. As a consequence, heat transfer and mass loss is calculated to be uniform over the entire surface, and the particle remains spherical throughout its re-entry history.

A particle re-entering the earth's atmosphere passes through a number of different aerodynamic flow regimes - free molecule, near free molecule, low Reynolds number continuum, and fully developed boundary layer continuum.* In each flow regime the accompanying mass and heat transfer phenomena can be described by a unique set of equations. Therefore, in order to describe the history of re-entering particles it is necessary to establish appropriate equations for each of these regimes.

4.1.2.1 Continuum Flow Analysis

During the initial development of an analytical program, it was found that only in the case of the fully developed boundary layer continuum flow regime were equations

* L. Goldberg and S. M. Scala, "Mass Transfer in the Hypersonic Low Reynolds Number Viscous Layer", General Electric Missile and Space Vehicle Dept., Document No. R 62 SD07 (January 1962).

developed to predict the mass loss rate of graphite during re-entry. It was decided to use these equations throughout the entire re-entry flight path to obtain some idea of the extent of graphite burn-up which occurs during re-entry. Since the assumption of a continuum flow regime throughout the flight trajectory should tend to result in lower burn-up rates, it was believed that such an analysis would give a conservative result or a lower order of minimum particle sizes which could be consumed during re-entry.

The details and results of this initial program are reported in a Westinghouse topical report.*

A group of equations were assembled by means of a Fortran code into an overall computer program. It was assumed in this program that the particles re-entered from an initial altitude of 1.8×10^6 ft. and an initial velocity of 25,000 ft/sec. Arbitrary re-entry angles ranging from 5° to 90° were assumed. It was further assumed that oxidation did not occur when the temperature of the re-entering particle decreased below 1000°R . The following conclusions were drawn as a result of this program:

1. The larger particle sizes (1" and 1/2" diameter) exhibit a negligible size change on re-entry, which is independent of re-entry angle.
2. Particles of less than 1/4" diameter show a larger relative change in size. Also, the total decrease in radius is greater for the smaller re-entry angles.
3. A 1/32" particle will have its radius reduced to 150-250 microns. The exact value depends on re-entry angle.
4. It appears that a 1/64" diameter particle will be reduced to less than 25 micron diameter during re-entry.
5. Initial temperature of the particle has no effect on radius change during re-entry.

4.1.2.2 Free Molecule Flow Regime

The analytical program just described is obviously invalid at high altitudes

* WANL-TNR-039

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WANL-TME-203

(above 300,000 ft.) where a re-entering particle encounters the free molecule flow regime. Therefore, a program was undertaken to describe the aerodynamic and mass loss behavior of a re-entering particle in this regime.

By definition, free molecule flow exists whenever the mean free path of the ambient atmosphere is large compared with the characteristic dimension of the body moving through the gas. It is generally considered that as long as the mean free path is ten times as large as the characteristic body dimension (in the case of spherical bodies this dimension is the radius) free molecule flow exists. Under these conditions molecules may approach, collide and rebound from the body surface without colliding with other incident or re-emitted molecules in the neighborhood of the body. Therefore, each molecule behaves as a discrete particle. Since such a flow field rules out the formation of a definable shock wave and the presence of a gaseous boundary layer at the body surface the transfer of energy to the body by collisions with incident gas molecules is the most efficient mechanism possible. Moreover, the accompanying oxidation and mass transfer phenomena are reaction rate controlled in contrast to the partial or even completely diffusion controlled mechanisms occurring in shock wave type continuum flow. For spherical bodies 1/16 inch and greater in diameter this flow regime exists at altitudes greater than 300,000 ft.

A set of equations have been developed for fuel re-entry in the free molecule flow regime. These equations have been programmed for use on an IBM 7090 digital computer. The full development of these expressions is given in a WANL Topical Report ("Aerodynamic Heating and Oxidation of Graphite in Free Molecule Flow", Westinghouse Electric Corporation, WANL-TNR-067, October 1962.)

The results of the computer program for free molecule flow using the same initial re-entry conditions as for the continuum flow program indicate that trajectory, heating, and mass loss plots are somewhat different from those obtained using a continuum flow analysis for high altitude re-entry. However, the essential conclusions reached in the earlier study remains the same. Only particles 1/64" or smaller will be reduced to less than a 25 micron diameter during re-entry.

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4.1.2.3 Fallout Times

The analytical programs undertaken in this study program have considered re-entry only in the case of hypersonic flow conditions. Some considerations have been given to the amount of settling time required for a particle to reach earth once it has reached a terminal velocity. These values are of importance if one wishes to consider the radiological hazards associated with a particle of fuel once it reaches the earth's surface. In figure 28 is shown the length of time required for various size particles to reach the earth's surface from an altitude of 100,000 ft.

4.1.3 Experimental Program

The mass loss rate equation for graphite used in the case of the analytical program for the continuum flow regime was that of S. M. Scala.* Although this equation has an excellent theoretical foundation it has not been verified experimentally. Therefore, an experimental program was undertaken. The details of this investigation are reported elsewhere.** In this program a number of graphite samples of NERVA fuel were investigated in a plasmajet facility where hypersonic continuum flow conditions may be simulated. A range of altitude-velocity points, typical of those found on a re-entry trajectory curve, was studied. The results of this study are shown in Figure 29 as a function of altitude for the various velocities studied in this program. The solid lines represent the theoretically predicted values, the symbols the experimentally determined ones. It is obvious from these curves that major deviations from theory occur only at low velocities and high altitudes. Since these points are not in a hypersonic continuum flow regime, these deviations are to be expected. As a result of these studies, it appears that Scala's equation may be used in the appropriate altitude-velocity region which may be discerned from an examination of Figure 29.

Although Scala's theory adequately predicts mass loss rates for graphite oxidation, the material of actual interest during re-entry is fueled graphite. Therefore, it was of

* S. M. Scala, "Hypersonic Boundary Layer Materials Erosion", General Electric Company, MSVD, TIS Document No. R59SD438 (Sept. 1959).

** "Experimental Study of the Re-Entry Oxidation", Westinghouse Electric Corporation, WANL-TNR-072, November 1962.

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FIG. 28
PARTICLE SIZE VS TIME TO FALL 100,000 ft.

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FIG. 29
 MASS LOSS RATE OF GRAPHITE AS A FUNCTION OF ALTITUDE

considerable importance to compare oxidation rates for both fueled and unfueled materials. A comparison of oxidation rates was made at several test points, and the results of this study are shown in Table III. It is apparent that no significant difference exists between the rates of oxidation of both materials. The conclusion may then be made that the oxidation of both fueled and unfueled graphite should follow the same mechanism and that the same rate laws should apply for the oxidation of both materials.

The determination of the destruct system requirements must represent a composite of all the elements of the problem; the fission product inventory and associated initial source strength, destruct system dynamics, particle re-entry dynamics, meteorological considerations, and the relation between dose rate, particle size, and particle concentration. Until all these data are available it may be necessary to study the destruct system from a totally conservative viewpoint, relaxing requirements as justification is developed. Such an approach at this time would lead to a requirement of 1/64" diameter or smaller particles.

4.2 Anti-Criticality Destruct System (A.C.D.S.)

The minimum critical mass of U-235 graphite-water systems has been studied by W. Stratton, LASL. Although minimum critical mass was found to be relatively insensitive to carbon-uranium ratio, it is extremely sensitive to hydrogen-uranium ratio, with a peak at H/U of 550, resulting in a minimum critical mass of 890 grams. Applying a safety factor, it can be stated that no more than 750 gm of uranium should land within any 27" diameter circle on the surface of the water. It is desirable, of course, to design the A.C.D.S. so that the largest pieces of fuel having uranium content less than 27" apart. For the higher fuel concentration presently considered, this results in a desired particle diameter of 15 cm.

4.3 Chemical Destruct System

In the event of a post-operational flight failure of the reactor, or a deliberate action to disassemble the reactor, it is desirable to minimize the deposition of fission products in the impact area. One means of accomplishing this objective is to attack chemically the reactor fuel so that this fission products may be leached and dispersed as a gas, or so that the fuel may be weakened structurally, fragment and burnup on re-entry. This system is being explored as an alternate or supplement to an explosive destruct system.

TABLE III

COMPARISON OF MASS LOSS RATES AND OXIDATION
RATES FOR FUELED AND UNFUELED GRAPHITE SAMPLES

<u>Test Point</u>	<u>Sample Diameter (in.)</u>	<u>Oxidation Rate $R \times 10^4$ (g/cm²-sec.)</u>		<u>Per Cent Difference *</u>
		<u>Fueled</u>	<u>Unfueled</u>	
1	1/2	7.0	7.8	-10
1	1/4	11.1	17.5	-36
1	1/8	12.8	13.8	- 7
2	1/2	7.6	6.1	+25
2	1/4	14.3	11.9	+20
2	1/8	13.9	12.0	+16
3	1/2	6.1	6.4	- 5
3	1/4	7.4	8.0	- 8
3	1/8	12.4	8.6	+42

* The Percent Difference is defined as being equal to:

$$\frac{R(\text{Fueled}) - R(\text{Unfueled})}{R(\text{Fueled})} \times 100\%$$

4.3.1 Requirements

1. The reacting material must be capable of being stored in a cryogenic environment, and released from this environment.
2. The system must withstand flight environment, including radiation effects, as well as powered flight heating, shock and vibration.
3. The reactant must be capable of penetrating the NbC coating on the fuel channel and the pyrographite coating on the fuel beads by reaction or diffusion.
4. The system must be capable of leaching * % of the fission products from the fuel, or of so weakening the fuel structure that it will form fragments 1/64" maximum on re-entry.
5. The reaction must occur within * seconds after receipt of the destruct signal.

In order to explore the choice of suitable reagents a twofold program was developed; a theoretical approach utilizing analytical techniques (free energy calculations) to determine the most favorable reagents, and a series of experiment to test these selections. The results presented herein represent a preliminary screening. Further work must be done before definitive conclusions can be drawn.

4.3.2 Experimental Program

The experimental program tested a number of reagents, under varying environments. The methods employed and a full discussion are included in WANL-TNR-075, "Preliminary Investigation of NERVA Core Destruct Chemicals". A summary of these results is given in the following pages.

4.3.2.1 Results and Conclusions (See Tables

1. The most promising reagents tested to date have been Br_2 , Cl_2 , BrF_3 and BrF_5 , resulting in structural degradation of the samples tested.
2. Chlorine may be able to leach the uranium from the UC_2 .
Further work on activated samples must be done to

*Not yet determined.

demonstrate reliability of this technique.

3. The reaction between these reagents and the UC_2 is favored over the reaction with the carbon matrix. This preference is advantageous in that the stoichiometric requirements for the reaction may be met with far less material for the UC_2 which only constitutes a small fraction of the core.
4. O_2 and H_2O are not satisfactory reagents, based on the large volumes required and the decrease over $1000^\circ C$, of their selectivity for the UC_2 reaction.
5. Other materials which were screened and did not show to advantage were HNO_3 , HF , HCl , and N_2 .
6. Future work must include effect on reaction efficiency of time, temperature, and optimum fuel reagent proportions.

4.4 Nuclear Destruct - TREAT Experiments

4.4.1 Introduction

One possible means to minimize the hazards of an airborne reactor falling to the ground intact, is to destroy it while in flight. One technique proposed for this destruction is to force the reactor to a high speed transient, relying on the energy created therein to destroy the core, or to weaken it sufficiently that it would fragment on re-entry. In order to simulate this excursion mechanism, a test series was planned for the Argonne TREAT (Transient Reactor Test Facility) reactor, utilizing NERVA fuel samples.

The experiments in the TREAT reactor are intended to explore the following:

1. Material degradation in the fuel, possibly resulting in fuel failure and burnup during re-entry.
2. Behavior of the fuel under varying environments in a reactor transient, aiding understanding of the shutdown mechanism.
3. Release of fission product gases, particularly Iodine, from fuel element material subjected to a reactor transient.

RESULTS OF FREE ENERGY CALCULATIONS

	<u>Reactants</u>	<u>Products</u>	<u>$\Delta^{\circ} F$ 298 (K-cal.)</u>
1)	$2UF_6 + UC_2$	$3UF_4 + 2C$	-249
	$2UF_6 + C$	$2UF_4 + CF$	-21.8
2)	$4BF_3 + 3UC_2$	$4B + 6C + 3UF_4$	-92.0
	$4BF_3 + 3C$	$3CF_4 + 4B$	+591.0
3)	$4BCl_3 + 3UC_2$	$4B + 6C + 3UCl_4$	-200.4
	$4BCl_3 + 3C$	$3CCl_4 + 4B$	+317.5
4)	$4BBr_3 + 3UC_2$	$4B + 6C + 3UBr_4$	-286.5
	$4BBr_3 + 3C$	Data not available	--
5)	$2F_2 + UC_2$	$UF_4 + 2C$	-379.0
	$2F_2 + C$	CF_4	-151.8
6)	$2Cl_2 + UC_2$	$UCl_4 + 2C$	-188.0
	$2Cl_2 + C$	CCl_4	-16.4
7)	$2Br_2 + UC_2$	$UBr_4 + 2C$	-146.5
	$2Br_2 + C$	Data not available	--
8)	$2F_2O + UC_2$	$UF_4 + CO_2 + C$	-492.7
	$2F_2O + C$	$CF_4 + O_2$	-171.5
9)	$4HF + UC_2$	$UF_4 + 2C + 2H_2$	-120.0
	$4HF + C$	$CF_4 + 2H_2$	-162.2
10)	$2O_2 + UC_2$	$UO_2 + 2CO$	-281.0
	$1/2O_2 + C$	CO	-32.8
11)	$4H_2O + UC_2$	$UO_2 + 2CO + 4H_2$	-62.2
	$H_2O + C$	$CO + H_2$	+21.8
12)	$CrO_2Cl_2 + UC_2$	$UO_2 + CrCl_2 + 2C$	-174.7
	$CrO_2Cl_2 + C$	$CO_2 + CrCl_2$	-54.0

TABLE IV
 RESULTS OF FREE ENERGY CALCULATIONS
 (Room Temperature)

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OXYGEN REACTIONS IN THE STATIC SYSTEM

TABLE V

OXYGEN REACTIONS IN THE STATIC SYSTEM							
Run Number	Sample Type	Temp. Range °C	Reaction Time (min)	Stoichiometric %	Wt. Change %	Initial Cond. of Sample	Remarks (Resulting Sample Cond.)
1	B-4 Fuel	1650 (max.)	10	9.2 x 100	-79	Carbided	Powdered completely with slight shock temp. dripped with time
2	B-4 Fuel	1650 to 1200	10	2.6 x 100	-66.8	Carbided	~17% of orig. was hard core, balance powder
3	B-4 Fuel	2450 to 1000	10	4.25 x 100	-78.5	Carbided	~4% of orig. was hard core, balance powder
4	B-4 Fuel	2300	2	--	-15.9	Carbided	Done in open air powdered surface but hard core
5	Carbon	2300	2	--	-10.7	Off the shelf	Done in open air powdered surface but hard core
6	95% Carbon 5%	2300	2	--	-11.6	Off the shelf	Done in open air yellow film, hard core
7	B-4 Fuel	2250 to 1050	10	2.9 x 100	-69.9	Carbided	Reagent = 15% H ₂ O, 85% O ₂ ~12% of orig. was hard core, balance powder
8	B-4 Fuel	2300	10	70	-47.2	Carbided	Reagent = 52% H ₂ O, 48% O ₂ hard core no powder metallic surface appearance
9	B-4 Fuel	1000	10	2.7 x 100	-42.0	Off the shelf	Powdered surface hard core
10	B-4 Fuel	1000	10	2.3 x 100	-40.8	Carbided	Powdered surface hard core

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OXYGEN REACTIONS IN THE DYNAMIC SYSTEM

TABLE VI

OXYGEN REACTIONS IN THE DYNAMIC SYSTEM

<u>Run Number</u>	<u>Sample Type</u>	<u>Temp. Range °C</u>	<u>Reaction Time (min)</u>	<u>Stoichiometric %</u>	<u>Wt. Change %</u>	<u>Flow Rate cc/min</u>	<u>Initial Cond. of Sample</u>	<u>Remarks (Resulting Sample Cond.)</u>
O ₂ -1A	B-4 Fuel	1500 to 1700	15	51	-38.6	160	Carbided	~61% of original was hard core; balance powder
O ₂ -2A	B-4 Fuel	1460 to 1570	3	26	-15.0	450	Carbided	~86% of original was hard core; very little powder
O ₂ -3A	B-4 Fuel	1400 to 1700	30	95	-53.4	165	Carbided	~32% of original was hard core; red powder formed (possibly UO ₃)
O ₂ -4A	B-4 Fuel	850 to 1100	13	50	-24.7	200	Off the shelf	~73% of original was hard core; balance powder
O ₂ -1B	B-1 Fuel	1040 to 1080	13	52	-48.0	200	Off the shelf	~60% of original was hard core; balance powder
O ₂ -2B	B-1 Fuel	795 to 1000	5.8	50	-28.3	400	Off the shelf	~65% of original was hard core; balance powder
O ₂ -3B	B-1 Fuel	830 to 1200	3.4	50	-24.5	800	Off the shelf	~63% of original was hard core; balance powder
O ₂ -4B	B-1 Fuel	815 to 1200	1.6	50	-24.2	1600	Off the shelf	~72% of original was hard core; balance powder
O ₂ -5B	B-1 Fuel	760 to 815	18.4	50	+2.0	100	Off the shelf	~96% of original was hard core; balance powder

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HALIDE REACTIONS

Run Number	Sample Type	Temp. Range °C	Reaction Time (min)	Stoichiometric %	Wt. Change %	Initial Condition of Sample	Reagent Used	Remarks (Resulting Sample Condition)
1	B-4 Fuel	2160	10	~ 6 x 100	-10.1	Carbided	HCl Vapor	Surface powder but hard core (>80% orig. sample)
2	B-4 Fuel	20 to 100	~ 60	Excess	-.05	Carbided	10% HF 90% H ₂ O	Sample cracked but remained hard
3	B-4 Fuel	20 to 2100	30	Excess	-19.8	Carbided	Cl ₂ Gas	Intact specimen but probably no uranium left
4	B-4 Fuel	20	30	Excess	-45	Off the shelf	BrF ₃ + air	Surface powder but intact
5	B-4 Fuel	20 to 90	160	Excess	-45	Off the shelf	BrF ₃ + N ₂	120 min. at 20°C - intact 40 min at 90°C - powder and fragments
6	B-1 Fuel	80	40	Excess	+16.2	Carbided	BrF ₃ vapor	Powder and fragments
7	B-4 Fuel	30 to 110	120	Excess	-40	Off the shelf	BrF ₃ vapor	Coarse powder
8	B-1 Fuel	100	15	Excess	+~35	Off the shelf	BrF ₃ vapor and liquid	Fine powder
9	B-1 Fuel	100 (400)	15 (5)	Excess	+21	Off the shelf	BrF ₃ vapor (air)	Fine powder
10	B-1 Fuel	100	30	Excess	+1.0	Off the shelf	Cl ₂ Gas	Surface powder, intact core
11	B-1 Fuel	20	90	Excess	+2.4	Off the shelf	BrF ₅ vapo	Surface powder, intact core
12	B-1 Fuel	456	30	Excess	+5.4	Off the shelf	Cl ₂ Gas	Powder and weak fragments
13	B-1 Fuel	810 to 860	90	Excess	-18	Off the shelf	Cl ₂ Gas	60 min - hard and intact 30 min - hard fragments
14	B-1 Fuel	840	90	Excess	-17.1	Off the shelf	Cl ₂ Gas	Hard and intact
15	B-1 Fuel	20	1.0	Excess	+0.3	Off the shelf	Br ₂ (Liq.)	Swelled and weakend but did not fragment
16	B-4 Fuel	20	1.25	Excess	+13	Off the shelf	Br ₂ (Liq.)	Weak fragments

NOTE:

1. Runs 1, 2 and 3 - Static System. Balance of runs - Flow System.
2. Runs 3, 8, 9, 12 and 14 analyzed for Uranium.
3. Runs 4, 7 and 11 were performed in the nickel crucible.
4. Runs 5, 6, 8 and 9 were performed in nickel cylinders.
5. Runs 10, 12, 13, 14 were performed in the quartz tube.
6. Runs 15 and 16 were performed in a beaker in a hood.

TABLE VII
HALIDE REACTIONS

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TABLE VIII
ANALYSIS OF SAMPLES FOR REACTIVITY OF REACTION WITH UC₂

B-4 Fuel	<u>Untreated</u>	<u>Cl₂-3</u>	<u>Br₂-3</u>
Specific Activity, cpm/gm	3,949	67	2,880
Uranium Conc., Wt. %	18.8	0.5	14.5

B-1 Fuel	<u>Untreated</u>	<u>Cl₂-1</u>	<u>Cl₂-2</u>	<u>Br₂-1</u>	<u>BrF₃</u>	<u>BrF₅</u>
Specific Activity, cpm/gm	2,969	3023	635	3350	1926	3033
Uranium Conc., Wt. %	19.7	12.7	0.5	19.6	?	14.4

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Run Number	Sample Type	Temp. Range °C	Reaction Time (Min.)	Stoichiometric %	Wt. Change %	Initial Cond. of Sample	Reagent Used	Remarks (Resulting Sample Cond.)
1	B-4 Fuel	2170-1240	10	30	-18.5	Carbided	H ₂ O (Vapor)	Surface Powder But Hard Core (>70% Original Sample)
2	B-4 Fuel	2030-1750	10	30	-23.8	Off the Shelf	H ₂ O (Vapor)	Surface Powder But Hard Core (>70% Original Sample)
3	B-4 Fuel	2100-1100	30	50	-33.5	Carbided	30% H ₂ O ₂ 70% H ₂ O (Vapor)	Intact
4	B-4 Fuel	1350-870	90	Excess	-36.7	Carbided	30% H ₂ O ₂ 70% H ₂ O (Vapor)	Continuous Delivery by Boiling-Intact
5	B-4 Fuel	20-100	3	Excess	+ 2.5	Carbided	30% H ₂ O ₂ 70% H ₂ O (Liquid)	Intact
6	B-4 Fuel	100	30	Excess	+ 0.9	Carbided	H ₂ O (Liquid)	Sample in Boiling Water-Cracked But Hard
7	B-4 Fuel	20-230	90	Excess	-12.4	Off the Shelf	HNO ₃ (Liquid) H ₂ O ₂ (Liquid)	Intact
8	B-4 Fuel	1000	13.0	Excess	-	Off the Shelf	H ₂ O (Liquid)	Outer Walls Almost Eaten Away
9	B-4 Fuel	1050	0.25	Excess	-	Off the Shelf	H ₂ O (Liquid)	Outer Wall Cracked
10	B-4 Fuel	825-1000	0.25-1.00	Excess	-	Off the Shelf	H ₂ O (Vapor)	No Apparent Attack
11	B-4 Fuel	1400	60	Excess	0	Off the Shelf	N ₂ (Dry)	No Apparent Attack

TABLE IX
MISCELLANEOUS REACTIONS

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4. Irradiation effects on reactor materials and components.

4.4.2 Results and Conclusions

The results of a number of tests in TREAT were discussed in a paper presented by Dr. A. Boltax at the Nuclear Propulsion Conference, Monterey, California, August, 1962. In these tests, transient irradiations were performed in helium, vacuum, and water environments, on UO_2 fuel samples, and on samples made of UC_2 particles, coated with pyrolytic graphite. The results of these tests indicate: (Table X)

1. In TREAT, it was possible to simulate NERVA excursions of up to 10,000 MW/sec. These excursions were capable of causing the 20-30% UO_2 fuel to fragment with a TREAT energy threshold of 350 MW/sec., corresponding to 3300 MW/sec. in NERVA.
2. The UC_2 fuel, containing 4-5% UO_2 , is more resistant to this radiation damage than fuel containing 20-30% UO_2 with the threshold for fragmentation about 770 MW sec. Coated particle fuel, containing 1-2% UO_2 seems to be the most resistant, with transients equivalent to 11,000 MW sec necessary to cause fragmentation.
3. Samples pulsed in a water environment had the lowest fragmentation threshold, about 100 MW sec. At 800 MW sec, a violent reaction occurred, producing a pressure pulse of 5000 psi, for under 0.1 sec.
4. The mean particle size varies inversely as the excursion energy.
5. Constraint of the fuel (such as might occur in an actual core environment) appears to inhibit fragmentation.

Work is continuing on these studies. There are indications that heat inputs on the order of 2000 calories per gram, equivalent to 10,000 MW/sec in NERVA, (about 3400°C , Maximum Temperature), may result in driving out material with a somewhat higher U/C ratio than that of the original sample. When this phenomenon occurs, there does not appear to be a marked tendency toward fragmentation. However, work is

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SUMMARY OF SELECTED TRANSIENT EXPERIMENTS

TABLE X

	Reactor Energy (MWS)	Peak Power (MW)	Reactor Period (ms)	Maximum Sample Temp. (°C)	Xenon-133, microcuries			Particle Size d _{sv} (mils)
					Produced	Released	% Released	
Water Tests	105	300	140	900	735	109	15	86
Air Stored Fuel	360	1120	75	1710	2480	1680	68	28
	800	4470	40	> 2600	4470	--	--	18
Helium Tests	255	700	95	1700	1300	30	2	No fragmentation
Air Stored Fuel	380	2430	54	2150	1940	1400	72	32
	645	3850	40	~ 2900	3300	2100	64	23
	890	3850	40	> 3000	4550	2300	51	22
Helium Test	325	1120	81	1950	--	--	--	No fragmentation*
Oxygen Stored Fuel								
	270	--	54	1740	1590	60	4	No fragmentation
	770	3860	42	> 3000	3860	2150	56	330
	790	4730	40	> 3000	4340	320	7	No fragmentation**
Vacuum Test	765	4140	40	> 3000	3930	1070	27	66
Completely Carbided Fuel								

* The results given were the last obtained in a series of transients used in a calibration experiment.

** The fuel sample holes were lined with NbC. Also, this sample was constrained by multiple layers of graphite cloth. Further experiments on the effects of constraint on fuel sample fragmentation are in preparation.

continuing to determine if this mechanism may provide a suitable means of dissipating fission products and removing uranium from the core prior to re-entry.

A series of experiments which have been run subsequent to those reported (op. cit.) will be described in a forthcoming WANL-ANL joint issue document. A new series of experiments, including water immersion, weight loss, fragmentation, iodine release and thermal stress, will be conducted in the period November, 1962 through February, 1963.

4.5.1 Fission Product Inventory

The WANL fission product inventory code is currently being checked against the NRDL code. In the interim, three conditions have been analyzed, and presented herein.

1. Neutron burst, duration 0.1 minutes, 10^{10} fissions
2. Operation for 10 minutes at 1120 MW
3. Operation for 30 minutes at 1120 MW.

4.5.1.1 ACTIVITY (μc) FROM 0.1 MIN. BURST OF 10^{10} THERMAL FISSIONS

Isotope	0.1	1	10	100	1000
Zn ⁷²	1.772 - 7	1.771 - 7	1.768 - 7	1.729 - 7	1.385 - 7
Ga ⁷²	0	0	0	0	0
Ga ⁷³	1.168 - 5	1.166 - 5	1.142 - 5	9.232 - 6	1.103 - 6
Ga ⁷⁴	1.395 - 3	1.288 - 3	5.788 - 4	1.947 - 7	0
Ge ⁷⁷	1.427 - 4	1.426 - 4	1.413 - 4	1.289 - 4	5.132 - 5
Ge ⁷⁸	4.955 - 3	4.931 - 3	4.693 - 3	2.860 - 3	2.013 - 5
As ⁷⁷	6.993 - 5	6.995 - 5	7.015 - 5	7.186 - 5	7.438 - 5
As ⁷⁸	1.887 - 6	3.564 - 5	3.519 - 4	1.963 - 3	6.375 - 5
As ⁷⁹	1.936 - 1	1.8-6 - 1	9.030 - 2	8.816 - 5	6.375 - 5
Se ^{79m}	1.717 - 3	2.915 - 2	1.009 - 1	1.552 - 4	0
Se ⁷⁹	0	0	0	0	0
Se ^{81m}	4.599 - 3	4.549 - 3	4.078 - 3	1.365 - 3	2.382 - 8
Se ⁸¹	2.229 - 1	2.156 - 1	1.549 - 1	7.032 - 3	3.517 - 8
Se ⁸³	2.745 - 1	2.677 - 1	2.086 - 1	1.720 - 2	0
Se ⁸⁴	8.614	7.130	1.077	6.670 - 9	0
Se ⁸⁵	5.011 + 1	1.920 + 1	1.308 - 3	0	0
Br ⁸³	4.126 - 1	4.226 - 2	5.050 - 2	5.754 - 2	7.993 - 4
Br ⁸⁴	1.077 - 1	2.503 - 1	7.182 - 1	1.141 - 1	3.365 - 10
Br ⁸⁵	5.845 - 1	6.424	1.467	1.373 - 9	0

Isotope	0.1	1	10	100	1000
Kr ^{83m}	9.323 - 3	9.500 - 3	1, 148 - 2	3.216 - 2	2.742 - 3
Kr ^{85m}	2.395 - 2	3.364 - 2	1.350 - 1	1.199 - 1	1.125 - 2
Kr ⁸⁵	0	0	0	0	0
Kr ⁸⁷	9.965 - 1	9.885 - 1	9.125 - 1	4.101 - 1	1.367 - 4
Kr ⁸⁸	6.635 - 1	6.610 - 1	6.369 - 1	4.394 - 1	1.067 - 2
Kr ⁸⁹	4.431 + 1	3.645 + 1	5.191	1.774 - 8	0
Rb ⁸⁷	0	0	0	0	0
Rb ⁸⁸	1.290 - 3	2.405 - 2	2.086 - 1	4.763 - 1	0
Rb ⁸⁹	9.993 - 2	1.693 + 1	6.146	1.307 - 1	0
Rb ⁹⁰	6.588 + 1	5.229 + 1	5.189	4.804 - 1	0
Sr ⁸⁹	2.057 - 3	2.057 - 3	2.057 - 3	2.055 - 3	2.038 - 3
Sr ⁹⁰	1.557 - 7	2.648 - 6	1.128 - 5	1.224 - 5	1.224 - 5
Sr ⁹¹	3.117 - 1	3.114 - 1	3.081 - 1	2.768 - 1	9.464 - 2
Sr ⁹²	1.022	1.018	9.792 - 1	6.662 - 1	1.410 - 2
Sr ⁹³	2.401 + 1	2.218 + 1	1.007 + 1	3.747 - 3	0
Sr ⁹⁴	1.263 + 2	7.818 + 1	6.442 - 1	0	0
Y ⁹⁰	0	0	0	0	0
Y ^{91m}	1.270 - 4	2.398 - 3	2.352 - 2	1.293 - 1	6.224 - 2
Y ⁹¹	5.181 - 8	9.826 - 7	1.024 - 5	9.748 - 5	6.022 - 4
Y ⁹²	1.639 - 4	3.104 - 3	3.143 - 2	2.255 - 1	
Y ⁹³	1.349 - 3	2.465 - 2	1.784 - 1	2.792 - 1	1.016 - 1
Y ⁹⁴	2.205 - 1	3.290	6.342	2.824 - 1	0

Isotope	0.1	1	10	100	1000
Y ⁹⁵	1.930 + 1	1.813 + 1	9.715	1.896 - 2	0
Zr ⁹³	0	0	0	0	0
Zr ⁹⁵	7.153 - 6	1.318 - 4	1.031 - 3	2.065 - 3	2.053 - 3
Zr ⁹⁷	1.806 - 1	1.805 - 1	1.794 - 1	1.688 - 1	9.150 - 2
Nb ^{93m}	0	0	0	0	0
Nb ^{95m}	4.049 - 4	4.049 - 4	4.044 - 4	3.998 - 4	3.561 - 4
Nb ⁹⁵	3.560 - 10	5.289 - 9	5.536 - 8	5.526 - 7	5.193 - 6
Nb ^{97m}	5.739	3.156	1.782 - 1	1.622 - 1	8.792 - 2
Nb ⁹⁷	2.758 - 3	3.949 - 2	8.895 - 2	1.375 - 1	9.864 - 2
Nb ⁹⁸	3.841 - 2	3.795 - 2	3.366 - 2	1.014 - 2	6.167 - 8
Nb ¹⁰⁰	6.482 - 1	5.265 + 1	6.584	6.159 - 9	0
Nb ¹⁰¹	1.690 + 2	9.054 + 1	1.768 - 1	0	0
Mo ⁹⁹	4.742 - 2	4.742 - 2	4.734 - 2	4.661 - 2	3.986 - 2
Mo ¹⁰¹	4.051 - 1	5.635	8.005	1.118 - 1	0
Mo ¹⁰²	1.164 + 1	1.103 + 1	6.410	2.826 - 2	0
Tc ^{99m}	3.970 - 6	7.538 - 5	7.820 - 4	7.157 - 3	3.151 - 2
Tc ⁹⁹	0	0	0	0	0
Tc ¹⁰¹	6.690 - 4	1.460 - 1	3.307	4.935 - 1	0
Tc ¹⁰²	3.741	1.110 + 1	6.457	2.846 - 2	0
Tc ¹⁰⁴	4.285 - 2	7.095 - 1	2.242	7.725 - 2	0
Tc ¹⁰⁵	3.111	2.903	1,451	3.264 - 3	3.229 - 3
Ru ¹⁰³	3.268 - 3	3.268 - 3	3.268 - 3	3.264 - 3	3.229 - 3

Isotope	0.1	1	10	100	1000
Ru ¹⁰⁵	4.043 - 4	7.418 - 3	5.553 - 2	8.399 - 2	8.124 - 3
Ru ¹⁰⁶	2.234 - 5	2.234 - 6	2.234 - 5	2.234 - 5	2.231 - 5
Ru ¹⁰⁷	1.280	1.118	2.880	3.707 - 7	0
Rh ^{103m}	9.907 - 1	1.872 - 5	1.857 - 4	1.146 - 3	1.612 - 3
Rh ^{105m}	1.435 - 5	2.836 - 3	5.195 - 2	8.419 - 2	8.144 - 3
Rh ¹⁰⁵	3.569 - 10	3.148 - 7	8.509	2.561 - 3	9.671 - 3
Rh ¹⁰⁶	1.528 - 6	1.662 - 5	2.234 - 5	2.234 - 5	2.231 - 5
Rh ¹⁰⁷	2.020 - 3	3.543 - 2	1.731 - 1	1.462 - 2	0
Pd ¹⁰⁷	0	0	0	0	0
Pd ¹⁰⁹	1.165 - 3	1.164 - 3	1.155 - 3	1.069 - 3	4.921 - 4
Pd ^{111m}	1.793 - 4	1.789 - 4	1.756 - 4	1.453 - 4	2.196 - 5
Pd ¹¹¹	2.549 - 2	2.481 - 2	1.895 - 2	1.365 - 3	1.769 - 5
Pd ¹¹²	2.478 - 4	2.477 - 4	2.465 - 4	2.346 - 4	1.43 - 0 4
Ag ^{109m}	5.996 - 5	7.414 - 4	1.156 - 3	1.070 - 3	4.925 - 4
Ag ¹¹¹	8.091 - 8	1.517 - 6	1.302 - 5	5.104 - 5	5.160 - 5
Ag ¹¹²	4.473 - 8	8.483 - 7	8.720 - 6	7.292 - 5	1.608 - 4
Ag ¹¹⁵	1.143 - 2	1.110 - 2	8.246 - 3	4.226 - 4	0
Cd ^{115m}	5.818 - 10	1.090 - 8	9.879 - 8	3.396 - 7	3.491 - 7
Cd ^{117m}	1.908 - 3	1.902 - 3	1.837 - 3	1.299 - 3	4.058 - 5
Cd ¹¹⁷	7.931 - 7	1.495 - 5	1.447 - 4	6.823 - 4	3.371 - 5
In ^{115m}	3.409 - 9	6.446 - 8	6.667 - 7	6.923 - 6	2.107 - 5
In ¹¹⁵	0	0	0	0	0

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Isotope	0.1	1	10	100	1000
In ^{117m}	5.538 - 8	1.084 - 6	1.466 - 5	2.824 - 4	9.329 - 5
In ¹¹⁷	2.834 - 7	5.351 - 6	5.269 - 5	2.894 - 4	1.988 - 5
Sn ^{117m}	0	0	0	0	0
Sn ¹²¹	2.839 - 4	2.838 - 4	2.827 - 4	2.722 - 4	1.865 - 4
Sn ¹²³	2.073 - 7	2.073 - 7	2.073 - 7	2.072 - 7	2.065 - 7
Sn ¹²⁵	2.623 - 2	2.456 - 2	1.275 - 2	4.712 - 5	2.793 - 5
Sn ¹²⁷	3.559 - 2	3.540 - 2	3.351 - 2	1.939 - 2	8.149 - 5
Sn ¹²⁸	2.026 - 1	2.004 - 1	1.817 - 1	6.014 - 2	1.062 - 6
Sn ¹³⁰	2.370 + 1	1.865 + 1	1.693	0	0
Sn ¹³²	6.568 + 1	4.946 + 1	2.902	0	0
Te ^{125m}	0	0	0	0	0
Te ^{127m}	1.321 - 6	1.321 - 6	1.321 - 6	1.321 - 6	1.315 - 6
Te ¹²⁷	0	0	0	0	0
Te ^{129m}	5.682 - 5	5.715 - 5	6.045 - 5	8.962 - 5	1.911 - 4
Te ¹²⁹	2.452 - 5	4.634 - 4	4.597 - 3	2.731 - 2	5.644 - 3
Te ^{131m}	8.777 - 4	1.058 - 3	2.614 - 3	7.094 - 3	5.254 - 3
Te ¹³¹	4.325 - 3	8.006 - 2	6.430 - 1	4.552 - 1	1.065 - 3
Te ¹³²	5.396 - 6	1.2192 - 3	2.634 - 2	3.132 - 2	2.736 - 2
Te ^{133m}	2.941	2.906	2.577	7.764 - 1	4.785 - 6
Te ¹³⁴	5.007	4.935	5.007	1.000	5.006 - 7
Sb ¹²⁵	8.652 - 10	4.771 - 9	3.677 - 8	2.373 - 7	2.373 - 7
Sb ^{126m}	0	1.590 - 8	1.226 - 7	0	0

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Isotope	0.1	1	10	100	1000
Sb ¹²⁶	0	0	0	0	0
Sb ¹²⁷	2.260 - 7	4.283 - 6	4.344 - 5	3.363 - 4	6.672 - 4
Sb ¹²⁸	2.727 - 3	1.418 - 2	9.192 - 2	7.314 - 2	8.377 - 4
Sb ¹²⁹	7.952 - 2	7.934 - 2	7.757 - 2	6.188 - 2	6.458 - 3
Sb ¹³⁰	1.158 - 1	1.876	4.276	8.026 - 4	0
Sb ¹³¹	3.525	3.431	2.616	1.736 - 1	0
Sb ¹³²	1.078	1.540 + 1	8.864	0	0
I ¹²⁹	0	0	0	0	0
I ¹³¹	1.222 - 8	2.337 - 6	2.598 - 4	5.763 - 3	5.631 - 3
I ¹³²	1.799 - 9	2.043 - 6	6.905 - 4	1.183 - 2	2.798 - 2
I ¹³³	5.013 - 2	5.156 - 2	6.496 - 2	1.345 - 1	1.021 - 1
I ¹³⁴	5.382 - 1	5.905 - 1	1.038	1.676	4.068 - 5
I ¹³⁵	4.737 - 1	4.730 - 1	4.657 - 1	3.988 - 1	8.451 - 2
I ¹³⁶	6.594 + 1	4.267 + 1	5.494 - 1	0	0
I ¹³⁷	4.235 + 2	9.138 + 1	1.998 - 5	0	0
Xe ^{133*}	1.260 - 8	2.387 - 7	2.492 - 6	2.418 - 5	1.724 - 4
Xe ¹³³	2.235 - 7	4.303 - 6	5.114 - 5	9.092 - 4	1.080 - 2
Xe ^{135*}	4.076 - 1	3.970 - 1	3.112 - 1	1.272 - 1	2.636 - 2
Xe ¹³⁵	4.642 - 5	8.752 - 4	8.506 - 3	5.833 - 2	1.391 - 1
Xe ¹³⁷	3.847	3.455 - 1	9.152	0	0
Xe ¹³⁸	1.006	9.701	6.722	1.714 - 1	2.403 - 10
Xe ¹³⁹	2.346	9.419 - 1	1.025 - 2	0	0

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Isotope	0.1	1	10	100	1000
Cs ¹³⁵	0	0	0	0	0
Cs ¹³⁷	3.026 - 7	1.248 - 6	9.910 - 6	1.217 - 5	1.217 - 5
Cs ¹³⁸	2.53 - 1	4.378 - 1	1.783	1.148	5.150 - 0
Cs ¹³⁹	4.373	1.382 - 1	1.096 - 1	1.541 - 2	0
Ba ^{137*}	0	0	0	0	0
Ba ¹³⁹	3.180 - 2	1.058 - 1	1.159 - 1	1.212	6.611 - 4
Ba ¹⁴⁰	1.076 - 2	1.076 - 2	1.075 - 2	1.072 - 2	1.036 - 2
Ba ¹⁴¹	1.091 - 1	1.054 - 1	7.451	2.328 - 1	0
Ba ¹⁴²	1.870 - 1	1.757 - 1	9.417	1.838 - 2	0
La ¹⁴⁰	1.546 - 7	2.937 - 6	3.071 - 5	3.040 - 4	2.635 - 3
La ¹⁴¹	1.535 - 2	4.461 - 2	2.836 - 1	6.815 - 1	4.548 - 2
La ¹⁴²	8.010 - 3	1.470 - 1	1.101	1.121	5.082 - 4
La ¹⁴³	1.039 - 1	1.004 - 1	7.086	2.217 - 1	0
Ce ¹⁴¹	1.089 - 8	4.058 - 7	2.294 - 5	8.474 - 4	3.948 - 3
Ce ¹⁴³	1.820 - 4	3.399 - 3	3.077 - 2	9.019 - 2	6.730 - 2
Ce ¹⁴⁴	4.647 - 4	4.647 - 4	4.647 - 4	4.646 - 4	4.639 - 4
Ce ¹⁴⁵	4.095	3.326 - 1	4.159	3.891 - 9	0
Ce ¹⁴⁶	6.881	6.579	4.200	4.720 - 2	0
Ce ¹⁴⁸	7.610	2.984	2.570 - 3	0	0
Pr ¹⁴³	4.750 - 5	4.755 - 5	5.307 - 5	2.913 - 4	2.741 - 3
Pr ¹⁴⁴	9.244 - 7	1.726 - 5	1.521 - 4	4.560 - 4	4.639 - 4
Pr ¹⁴⁵	3.983 - 3	6.840 - 2	3.086 - 1	2.877 - 1	5.047 - 2

Isotope	0.1	1	10	100	1000
Pr ¹⁴⁶	9.773 - 1	1.794 - 1	1.323	4.713 - 1	0
Pr ¹⁴⁷	7.006	6.641	3.955	2.185 - 2	0
Pr ¹⁴⁸	1.360	1.419 - 1	1.209	0	0
Nd ¹⁴⁴	0	0	0	0	0
Nd ¹⁴⁷	1.520 - 5	2.816 - 4	2.305 - 3	5.239 - 3	5.054 - 3
Nd ¹⁴⁹	2.940 - 1	2.924 - 1	2.776 - 1	1.651 - 1	9.122 - 4
Nd ¹⁵¹	1.054	1.005	6.218 - 1	5.124 - 3	0
Pm ¹⁴⁷	0	0	0	0	0
Pm ¹⁴⁹	3.199 - 6	6.-62 - 5	6.180 - 4	4.801 - 3	9.225 - 3
Pm ¹⁵¹	2.146 - 5	3.982 - 4	3.313 - 3	7.763 - 3	5.410 - 3
Sm ¹⁴⁷	0	0	0	0	0
Sm ¹⁵¹	0	0	0	0	0
Sm ¹⁵³	1.661 - 3	1.661 - 3	1.657 - 3	1.621 - 3	1.299 - 3
Sm ¹⁵⁵	4.288 - 2	4.178 - 2	3.221 - 2	2.394 - 3	0
Sm ¹⁵⁶					
	7.520 - 4	7.511 - 4	7.425 - 4	6.615 - 4	2.083 - 4
Eu ¹⁵⁵	7.067 - 10	1.326 - 8	1.224 - 7	4.626 - 7	4.897 - 7
Eu ¹⁵⁶	1.410 - 6	1.431 - 6	1.640 - 6	3.606 - 6	1.436 - 5
Eu ¹⁵⁷	2.636 - 4	2.634 - 4	2.617 - 4	2.446 - 4	1.245 - 4
Eu ¹⁵⁸	1.040 - 3	1.029 - 3	9.278 - 4	3.281 - 4	1.245 - 4
Gd ¹⁵⁹	3.094 - 5	3.092 - 5	3.074 - 5	2.902 - 5	1.628 - 5
Gd ¹⁶¹	6.354 - 4	5.368 - 4	9.948 - 5	0	0

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Isotope	0.1	1	10	100	1000
Tb ¹⁶¹	2.223 - 9	3.893 - 8	2.017 - 7	2.376 - 7	2.228 - 7
Sn ¹²⁶	0	0	0	0	0

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4.5.1.2

Activity from Fission Product Inventory

10 Min. Operating Time - 1120 MW Power

Time After Shutdown:	0	10	20	30
<u>Nuclide</u>				
Zn ⁷²	3.745×10^3	3.736×10^5	3.727×10^5	3.718×10^5
Ga ⁷²	1.530×10^3	4.570×10^3	7.577×10^3	1.055×10^4
Ga ⁷³	2.444×10^7	2.387×10^7	2.332×10^7	2.277×10^7
Ga ⁷⁴	1.965×10^9	8.080×10^8	3.323×10^8	1.366×10^8
Ge ⁷⁷	3.005×10^8	2.975×10^8	2.945×10^8	2.915×10^8
As ⁷²	4.489×10^5	1.339×10^6	2.217×10^6	3.084×10^6
As ⁷⁷	1.478×10^8	1.473×10^8	1.469×10^8	1.465×10^8
Ge ⁷³	1.021×10^{10}	9.661×10^9	9.144×10^9	8.654×10^9
As ⁷³	3.825×10^8	1.083×10^9	1.692×10^9	2.220×10^9
As ⁷⁹	2.868×10^{11}	1.328×10^{11}	6.146×10^{10}	2.845×10^{10}
Se ^{79m}	1.670×10^{11}	1.769×10^{11}	9.866×10^{10}	4.849×10^{10}
Se ⁷⁹	14.908	57.604	87.577	103.107
Se ^{81m}	9.171×10^9	8.121×10^9	7.191×10^9	6.637×10^9
Se ⁸¹	3.955×10^{11}	2.740×10^{11}	1.904×10^{11}	1.327×10^{11}
Se ⁸³	5.080×10^{11}	3.850×10^{11}	2.917×10^{11}	2.211×10^{11}
Br ⁸³	9.773×10^{10}	1.140×10^{11}	1.244×10^{11}	1.305×10^{11}
Kr ^{83m}	2.194×10^{10}	3.593×10^{10}	3.239×10^{10}	3.803×10^{10}
Se ⁸⁴	7.698×10^{12}	9.427×10^{11}	1.154×10^{11}	1.414×10^{10}

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Time After Shutdown:	0	10	20	30
<u>Nuclide</u>				
Br ⁸⁴	1.150×10^{12}	1.437×10^{12}	1.238×10^{12}	9.992×10^{11}
Kr ⁸⁵	2.128×10^4	9.623×10^4	1.823×10^5	2.674×10^5
Se ⁸⁵	1.049×10^{13}	2.461×10^8	5.775×10^3	1.355×10^{-1}
Br ⁸⁵	9.159×10^{12}	1.197×10^{12}	1.188×10^{11}	1.180×10^{10}
Kr ^{85m}	1.940×10^{11}	3.033×10^{11}	3.076×10^{11}	3.007×10^{11}
Kr ⁸⁷	2.019×10^{12}	1.874×10^{12}	1.690×10^{12}	1.546×10^{12}
Rb ⁸⁷	2.700×10^4	7.792×10^{-4}	1.245×10^{-3}	1.671×10^{-3}
Kr ⁸⁸	1.376×10^{12}	1.320×10^{12}	1.267×10^{12}	1.216×10^{12}
Rb ⁸⁸	2.378×10^{11}	5.953×10^{11}	8.199×10^{11}	9.552×10^{11}
Kr ⁸⁹	3.874×10^{13}	4.442×10^{12}	5.092×10^{11}	5.837×10^{10}
Rb ⁸⁹	9.859×10^{12}	1.160×10^{13}	8.006×10^{12}	5.174×10^{12}
Sr ⁸⁹	4.353×10^9	4.353×10^9	4.352×10^9	4.352×10^9
Rb ⁹⁰	5.079×10^{13}	3.899×10^{12}	2.993×10^{11}	2.298×10^{10}
Sr ⁹⁰	1.658×10^7	2.518×10^7	2.584×10^7	2.589×10^7
Y ⁹⁰	1.165×10^4	5.211×10^4	9.805×10^4	1.443×10^5
Sr ⁹¹	6.559×10^{11}	6.481×10^{11}	6.405×10^{11}	6.328×10^{11}
Y ^{91m}	2.562×10^{10}	7.207×10^{10}	1.120×10^{11}	1.463×10^{11}
Y ⁹¹	1.163×10^7	3.737×10^7	6.643×10^7	9.831×10^7
Sr ⁹²	2.117×10^{12}	2.028×10^{12}	1.943×10^{12}	1.862×10^{12}
Y ⁹²	3.334×10^{10}	9.821×10^{10}	1.578×10^{11}	2.129×10^{11}
Sr ⁹³	3.398×10^{13}	1.413×10^{13}	5.876×10^{12}	2.444×10^{12}

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Time After Shutdown:	0	10	20	30
<u>Nuclide</u>				
Y ⁹³	2.173×10^{11}	4.670×10^{11}	5.667×10^{11}	6.039×10^{11}
Zr ⁹³	40.908	45.229	51.513	58.563
Nb ^{93m}	7.817×10^{-6}	8.465×10^{-5}	1.287×10^{-4}	1.910×10^{-4}
Sr ⁹⁴	5.124×10^{13}	2.477×10^{11}	1.198×10^9	5.789×10^6
Y ⁹⁴	1.256×10^{13}	1.139×10^{13}	8.064×10^{12}	5.703×10^{12}
Y ⁹⁵	2.955×10^{13}	1.451×10^{13}	7.389×10^{12}	3.704×10^{12}
Zr ⁹⁵	1.220×10^9	2.749×10^9	3.588×10^9	3.982×10^9
Nb ^{95m}	8.565×10^8	8.554×10^8	8.544×10^8	8.534×10^8
Nb ⁹⁵	6.469×10^4	5.171×10^5	1.071×10^6	1.704×10^6
Zr ⁹⁷	3.810×10^{11}	3.784×10^{11}	3.759×10^{11}	3.733×10^{11}
Nb ^{97m}	2.123×10^{12}	3.638×10^{11}	3.612×10^{11}	3.488×10^{11}
Nb ⁹⁷	1.541×10^{11}	1.973×10^{11}	2.136×10^{11}	2.282×10^{11}
Nb ⁹⁸	7.616×10^{10}	6.665×10^{10}	5.833×10^{10}	5.105×10^{10}
Mo ⁹⁹	1.003×10^{11}	1.001×10^{11}	9.993×10^{10}	9.976×10^{10}
Tc ^{99m}	8.346×10^8	2.481×10^9	4.092×10^9	5.670×10^9
Tc ⁹⁹	1.751×10^2	1.350	2.371	3.493
Nb ¹⁰⁰	6.511×10^{13}	5.371×10^{12}	5.331×10^{11}	5.292×10^{10}
Nb ¹⁰¹	5.334×10^{13}	5.207×10^{10}	5.682×10^7	4.960×10^4
Mo ¹⁰¹	1.774×10^{13}	1.347×10^{13}	8.383×10^{12}	5.214×10^{12}
Tc ¹⁰¹	3.560×10^{12}	3.591×10^{12}	9.345×10^{12}	8.251×10^{15}
Mo ¹⁰²	1.356×10^{13}	1.016×10^{13}	5.560×10^{12}	3.043×10^{12}

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Time After Shutdown:	0	10	20	30
<u>Nuclide</u>				
Tc ¹⁰²	1.840×10^{13}	1.023×10^{13}	5.601×10^{12}	3.065×10^{12}
Ru ¹⁰³	6.916×10^9	6.915×10^9	6.915×10^9	6.913×10^9
Rh ^{103m}	2.015×10^8	5.733×10^8	9.026×10^8	1.194×10^9
Mo ¹⁰⁴	1.609×10^{13}	1.005×10^{12}	6.280×10^{10}	3.924×10^9
Tc ¹⁰⁴	3.774×10^{12}	4.172×10^{12}	2.939×10^{12}	2.006×10^{12}
Tc ¹⁰⁵	4.609×10^{12}	2.134×10^{12}	9.877×10^{11}	4.572×10^{11}
Ru ¹⁰⁵	6.631×10^{10}	1.473×10^{11}	1.816×10^{11}	1.946×10^{11}
Rh ^{105m}	5.673×10^{10}	1.422×10^{11}	1.795×10^{11}	1.939×10^{11}
Rh ¹⁰⁵	5.931×10^7	3.947×10^8	9.176×10^8	1.517×10^9
Ru ¹⁰⁶	4.728×10^7	4.728×10^7	4.728×10^7	4.728×10^7
Rh ¹⁰⁶	4.399×10^7	4.728×10^7	4.728×10^7	4.728×10^7
Ru ¹⁰⁷	1.410×10^{12}	3.125×10^{11}	6.924×10^{10}	1.534×10^{10}
Rh ¹⁰⁷	2.463×10^{11}	3.692×10^{11}	3.114×10^{11}	2.365×10^{11}
Pd ¹⁰⁷	1.774×10^{-1}	8.143×10^{-1}	1.464×10^0	1.979×10^0
Pd ¹⁰⁹	2.455×10^9	2.434×10^9	2.413×10^9	2.939×10^9
Ag ^{109m}	2.226×10^9	2.436×10^9	2.415×10^9	2.395×10^9
Pd ^{111m}	3.755×10^8	3.677×10^8	3.600×10^8	3.526×10^7
Ag ¹¹¹	1.555×10^7	4.100×10^7	6.012×10^7	7.420×10^7
Pd ¹¹¹	4.669×10^{10}	3.461×10^{10}	2.568×10^{10}	1.906×10^{10}
Pd ¹¹²	5.231×10^8	5.202×10^8	5.174×10^8	5.145×10^8
Ag ¹¹²	9.338×10^6	2.750×10^7	4.492×10^7	6.162×10^7

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Time After Shutdown: 0

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Nuclide

Ag ¹¹⁵	2.064×10^{10}	1.484×10^{10}	1.067×10^{10}	7.668×10^9
Cd ^{115m}	1.108×10^5	2.896×10^5	4.181×10^5	5.105×10^5
In ¹¹⁵	8.337×10^{-6}	0.0	0.0	0.0
Cd ¹¹⁵	7.762×10^7	1.123×10^7	1.370×10^8	1.547×10^8
In ^{115m}	9.002×10^5	3.305×10^6	6.394×10^6	9.934×10^6
Cd ^{117m}	3.962×10^9	3.813×10^9	3.669×10^9	3.530×10^9
In ¹¹⁷	5.799×10^7	1.624×10^8	2.542×10^8	3.349×10^8
Sn ^{117m}	1.574	12.677	38.934	83.726
Cd ¹¹⁷	1.584×10^8	4.397×10^8	6.730×10^8	8.653×10^8
In ^{117m}	1.435×10^7	3.447×10^8	1.020×10^8	1.599×10^8
Sn ¹²¹	5.996×10^8	5.971×10^8	5.946×10^8	5.921×10^8
Sn ¹²³	4.387×10^7	4.387×10^5	4.386×10^5	4.386×10^5
Te ^{125m}	1.286×10^0	7.51×10^0	16.968×10^0	28.087×10^0
Sn ¹²⁵	3.957×10^{10}	1.910×10^{10}	9.244×10^9	4.489×10^9
Sb ¹²⁵	1.460×10^5	3.312×10^5	1.207×10^5	4.640×10^5
Sb ^{126m}	2.027×10^1	5.229×10^1	9.452×10^1	8.996×10^1
Sb ¹²⁶	1.333×10^{-2}	8.468×10^{-3}	2.074×10^{-2}	3.657×10^{-2}
Sn ¹²⁶	6.283×10^1	6.283×10^1	6.283×10^1	6.283×10^1
Sn ¹²⁷	7.281×10^{10}	6.852×10^{10}	6.448×10^{10}	6.091×10^{10}
Sb ¹²⁷	4.669×10^7	1.363×10^8	2.214×10^8	3.008×10^8
Te ^{127m}	2.747×10^6	2.747×10^6	5.653×10^6	2.796×10^6

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Time After Shutdown:	0	10	20	30
<u>Nuclide</u>				
Te ¹²⁷	1.686×10^5	1.044×10^6	2.841×10^6	5.358×10^6
Sn ¹²⁸	4.040×10^{11}	3.578×10^{11}	3.230×10^{11}	2.805×10^{11}
Sb ¹²³	1.090×10^{11}	2.353×10^{11}	2.792×10^{11}	2.876×10^{11}
Sb ¹²⁹	1.662×10^{11}	1.621×10^{11}	1.581×10^{11}	1.542×10^{11}
Te ^{129m}	1.205×10^8	1.318×10^8	1.393×10^8	1.465×10^8
Te ¹²⁹	4.981×10^9	1.418×10^{10}	2.230×10^{10}	2.942×10^{10}
I ¹²⁹	1.301×10^{-3}	8.805×10^{-3}	2.300×10^{-2}	4.307×10^{-2}
Sn ¹³⁰	1.774×10^{13}	1.234×10^{12}	8.578×10^{10}	5.964×10^9
Sb ¹³⁰	3.502×10^{12}	6.352×10^{12}	2.612×10^{12}	9.991×10^{11}
Sb ¹³¹	6.451×10^{12}	4.772×10^{12}	3.539×10^{12}	2.602×10^{12}
Te ^{131m}	3.786×10^9	6.982×10^9	8.882×10^9	1.105×10^{10}
Te ¹³¹	7.552×10^{11}	1.745×10^{12}	2.179×10^{12}	2.270×10^{12}
I ¹³¹	1.581×10^8	9.433×10^8	2.143×10^9	3.478×10^9
Sn ¹³²	4.290×10^{13}	1.836×10^{12}	7.861×10^{10}	3.365×10^9
Sb ¹³²	3.728×10^{13}	7.003×10^{12}	4.990×10^{11}	2.870×10^{10}
Te ¹³²	2.985×10^{10}	6.307×10^{10}	6.677×10^{10}	6.692×10^{10}
I ¹³²	4.819×10^8	2.958×10^9	6.030×10^9	9.012×10^9
Te ^{133m}	5.830×10^{12}	5.103×10^{12}	4.466×10^{12}	3.908×10^{12}
I ¹³³	1.221×10^{11}	1.517×10^{11}	1.773×10^{11}	1.995×10^{11}
Xe ^{133m}	5.451×10^5	9.815×10^6	1.807×10^7	2.750×10^7
Xe ¹³³	5.207×10^7	1.743×10^8	2.209×10^8	4.988×10^8

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Time After Shutdown:	0	10	20	30
<u>Nuclide</u>				
Te ¹³⁴	9.795×10^{12}	8.337×10^{12}	7.096×10^{12}	6.039×10^{12}
I ¹³⁴	1.696×10^{12}	2.703×10^{12}	3.232×10^{12}	3.641×10^{12}
I ¹³⁵	9.941×10^{11}	9.771×10^{11}	9.604×10^{11}	9.440×10^{11}
Xe ^{135m}	7.533×10^{11}	5.865×10^{11}	4.786×10^{11}	4.083×10^{11}
Xe ¹³⁵	9.288×10^9	2.607×10^{10}	4.080×10^{10}	5.412×10^{10}
Cs ¹³⁵	5.333×10^0	5.424×10^0	5.593×10^0	5.835×10^0
I ¹³⁶	2.932×10^{13}	2.328×10^{11}	1.848×10^9	1.467×10^7
Ba ^{137m}	6.057×10^6	1.837×10^7	2.219×10^7	2.296×10^7
I ¹³⁷	5.721×10^{13}	2.276×10^6	9.051×10^{-2}	3.600×10^{-9}
Xe ¹³⁷	4.641×10^{13}	8.976×10^{12}	1.518×10^{12}	2.568×10^{11}
Cs ¹³⁷	1.225×10^7	2.351×10^7	2.539×10^7	2.570×10^7
Xe ¹³⁸	1.753×10^{13}	1.166×10^{13}	7.755×10^{12}	5.138×10^{12}
Cs ¹³⁸	2.335×10^{12}	4.650×10^{12}	5.591×10^{12}	5.732×10^{12}
Xe ¹³⁹	5.149×10^{13}	2.032×10^9	8.021×10^4	3.166×10^0
Cs ¹³⁹	3.002×10^{13}	1.640×10^{13}	7.906×10^{12}	3.812×10^{12}
Ba ¹³⁹	1.262×10^{12}	3.050×10^{12}	3.785×10^{12}	3.330×10^{12}
Ba ¹⁴⁰	2.276×10^{10}	2.275×10^{10}	2.274×10^{10}	2.274×10^{10}
La ¹⁴⁰	3.268×10^7	9.790×10^7	1.629×10^8	2.277×10^8
Ba ¹⁴¹	1.920×10^{13}	1.306×10^{13}	8.888×10^{12}	6.047×10^{12}
La ¹⁴¹	3.359×10^{11}	8.025×10^{11}	1.102×10^{12}	1.290×10^{12}
Ce ¹⁴¹	1.754×10^7	1.030×10^8	2.436×10^8	4.193×10^8

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Time After Shutdown:	0	10	20	30
<u>Nuclide</u>				
Ba ¹⁴²	2.865×10^{14}	1.433×10^{13}	7.163×10^{12}	3.581×10^{12}
La ¹⁴²	1.326×10^{12}	2.904×10^{12}	3.509×10^{12}	3.643×10^{12}
La ¹⁴³	1.829×10^{13}	1.244×10^{13}	8.465×10^{12}	3.759×10^{12}
Ce ¹⁴³	3.402×10^{10}	8.693×10^{10}	1.227×10^{11}	1.468×10^{11}
Pr ¹⁴³	1.009×10^8	1.264×10^8	1.636×10^8	2.112×10^8
Ce ¹⁴⁴	9.834×10^8	9.834×10^8	9.834×10^8	9.834×10^8
Pr ¹⁴⁴	1.723×10^8	4.388×10^8	6.178×10^8	7.379×10^8
Nd ¹⁴⁴	1.597×10^{-13}	0.0	0.0	0.0
Ce ¹⁴⁵	3.418×10^{13}	3.393×10^{12}	3.368×10^{12}	3.343×10^{10}
Pr ¹⁴⁵	4.463×10^{11}	6.917×10^{11}	7.038×10^{11}	6.928×10^{11}
Ce ¹⁴⁶	1.149×10^{13}	6.981×10^{12}	4.240×10^{12}	2.575×10^{12}
Pr ¹⁴⁶	1.606×10^{12}	3.421×10^{12}	3.918×10^{12}	3.765×10^{12}
Pr ¹⁴⁷	1.130×10^{13}	6.340×10^{12}	3.558×10^{12}	1.997×10^{12}
Nd ¹⁴⁷	2.683×10^9	6.401×10^9	8.486×10^9	9.654×10^9
Pm ¹⁴⁷	4.744×10^3	2.867×10^4	6.691×10^4	1.132×10^5
Sm ¹⁴⁷	0.0	0.0	0.0	0.0
Ce ¹⁴⁸	1.630×10^{13}	4.962×10^8	1.510×10^4	4.595×10^{-1}
Pr ¹⁴⁸	1.560×10^3	6.875×10^{11}	1.966×10^{10}	5.619×10^8
Nd ¹⁴⁹	6.047×10^{11}	5.708×10^{11}	5.387×10^{11}	5.085×10^{11}
Pm ¹⁴⁹	6.638×10^8	1.939×10^9	3.141×10^9	4.272×10^9
Nd ¹⁵¹	1.734×10^{12}	1.017×10^{12}	5.969×10^{11}	3.502×10^{11}

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Time After Shutdown:	0	10	20	30
<u>Nuclide</u>				
Pm ¹⁵¹	3.833×10^9	9.272×10^9	1.244×10^{10}	1.426×10^{10}
Sm ¹⁵¹	2.190×10^2	1.339×10^3	3.150×10^3	5.362×10^3
Sm ¹⁵³	3.511×10^9	3.503×10^9	3.494×10^9	3.485×10^9
Sm ¹⁵⁵	7.893×10^{10}	5.913×10^{10}	4.430×10^{10}	3.319×10^{10}
Eu ¹⁵⁵	1.363×10^5	3.622×10^5	5.314×10^5	6.582×10^5
Sm ¹⁵⁶	1.581×10^9	1.561×10^9	1.541×10^9	1.522×10^9
Eu ¹⁵⁶	3.228×10^6	3.718×10^6	3.21×10^6	4.679×10^6
Eu ¹⁵⁷	5.559×10^8	5.517×10^8	5.476×10^8	5.435×10^8
Eu ¹⁵⁸	2.080×10^9	1.853×10^9	1.651×10^9	1.471×10^9
Gd ¹⁵⁹	6.527×10^7	6.485×10^7	6.444×10^7	6.402×10^7
Gd ¹⁶¹	6.133×10^8	9.424×10^7	1.448×10^7	2.225×10^6
Tb ¹⁶¹	2.770×10^5	4.701×10^5	4.994×10^5	5.036×10^5
Total	2.921×10^{14}	1.730×10^{14}	1.218×10^{14}	9.189×10^{13}

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4.1.1.3

Activity from Fission Product Inventory

30 Min. Operating Time - 1120 MW Power

Time After Shutdown:	30	40	50	60
<u>Nuclide</u>				
Zn ⁷²	1.121×10^6	1.118×10^6	1.115×10^6	1.113×10^6
Ga ⁷²	1.368×10^4	2.270×10^4	3.163×10^4	4.046×10^4
Ga ⁷³	7.163×10^7	6.996×10^7	6.833×10^7	6.674×10^7
Ga ⁷⁴	3.105×10^9	1.278×10^9	5.251×10^8	2.159×10^8
Ge ⁷⁷	8.925×10^8	8.834×10^8	8.745×10^8	8.656×10^8
As ⁷²	4.005×10^6	6.639×10^6	9.239×10^6	1.180×10^7
As ⁷⁷	4.420×10^8	4.407×10^8	4.394×10^8	4.381×10^8
Ge ⁷⁸	2.901×10^{10}	2.746×10^{10}	2.599×10^{10}	2.469×10^{10}
As ⁷⁸	3.157×10^9	4.996×10^9	6.588×10^9	7.959×10^9
As ⁷⁹	4.810×10^{11}	2.227×10^{11}	1.031×10^{11}	4.772×10^{10}
Se ^{79m}	4.426×10^{11}	3.241×10^{11}	1.701×10^{11}	8.211×10^{10}
Se ⁷⁹	1.600×10^2	2.484×10^2	3.014×10^2	3.280×10^2
Se ^{81m}	2.448×10^{10}	2.168×10^{10}	1.920×10^{10}	1.700×10^{10}
Se ⁸¹	8.598×10^{11}	5.971×10^{11}	4.151×10^{11}	2.911×10^{11}
Se ⁸³	1.185×10^{12}	8.978×10^{11}	6.804×10^{11}	5.156×10^{11}
Br ⁸³	3.351×10^{11}	3.689×10^{11}	3.884×10^{11}	3.980×10^{11}
Kr ^{83m}	8.126×10^{10}	9.734×10^{11}	1.140×10^{11}	1.245×10^{11}
Se ⁸⁴	8.756×10^{12}	1.072×10^{12}	1.313×10^{11}	1.608×10^{10}

Time After Shutdown: 30

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60

Nuclide

Br ⁸⁴	3.860×10^{12}	3.709×10^{12}	3.030×10^{12}	2.446×10^{12}
Kr ⁸⁵	2.998×10^5	5.460×10^5	8.003×10^5	1.049×10^6
Se ⁸⁵	1.049×10^{13}	2.461×10^8	5.775×10^3	1.355×10^{-1}
Br ⁸⁵	1.048×10^{13}	1.328×10^{12}	1.318×10^{11}	1.308×10^{10}
Kr ^{85m}	8.049×10^{11}	9.116×10^{11}	9.014×10^{11}	8.794×10^{11}
Kr ⁸⁷	5.556×10^{12}	5.083×10^{12}	4.651×10^{12}	4.256×10^{12}
Rb ⁸⁷	2.294×10^{-3}	3.696×10^{-3}	4.978×10^{-3}	6.151×10^{-3}
Kr ⁸³	3.963×10^{12}	3.803×10^{12}	3.649×10^{12}	3.502×10^{12}
Rb ⁸⁸	1.653×10^{12}	2.370×10^{12}	3.806×10^{12}	3.052×10^{12}
Kr ⁸⁹	4.370×10^{13}	5.019×10^{12}	5.743×10^{11}	6.583×10^{10}
Rb ⁸⁹	2.947×10^{13}	2.478×10^{13}	1.649×10^{13}	1.059×10^{13}
Sr ⁸⁹	1.306×10^{10}	1.306×10^{10}	1.306×10^{10}	1.306×10^{10}
Rb ⁹⁰	5.499×10^{13}	4.221×10^{12}	3.241×10^{11}	2.488×10^{10}
Sr ⁹⁰	6.761×10^7	7.692×10^7	7.763×10^7	7.768×10^7
Y ⁹⁰	1.618×10^5	2.945×10^5	4.329×10^5	5.716×10^5
Sr ⁹¹	1.945×10^{12}	1.922×10^{12}	1.198×10^{12}	1.877×10^{12}
Y ^{91m}	2.097×10^{11}	3.304×10^{11}	4.340×10^{11}	5.228×10^{11}
Y ⁹¹	1.154×10^8	1.021×10^8	2.973×10^8	3.998×10^8
Sr ⁹²	6.088×10^{12}	5.833×10^{12}	5.589×10^{12}	5.355×10^{12}
Y ⁹²	2.898×10^{11}	4.689×10^{11}	6.344×10^{11}	7.871×10^{11}
Sr ⁹³	5.398×10^{13}	2.245×10^{13}	9.335×10^{12}	3.882×10^{12}

Time After Shutdown:	30	40	50	60
Nuclide				
Y ⁹³	1.251×10^{12}	1.638×10^{12}	1.786×10^{12}	1.835×10^{12}
Zr ⁹³	1.377×10^2	1.552×10^2	1.760×10^2	1.977×10^2
Nb ^{93m}	2.582×10^{-4}	4.201×10^{-4}	5.884×10^{-4}	9.129×10^{-4}
Sr ⁹⁴	5.149×10^{13}	2.489×10^{11}	1.203×10^9	5.818×10^6
Y ⁹⁴	3.202×10^{13}	2.515×10^{13}	1.780×10^{13}	1.259×10^{13}
Y ⁹⁵	5.173×10^{13}	2.586×10^{13}	1.293×10^{13}	6.465×10^{12}
Zr ⁹⁵	7.606×10^9	1.037×10^{10}	1.175×10^{10}	1.244×10^{10}
Nb ^{95m}	2.566×10^9	2.563×10^9	2.559×10^9	2.556×10^9
Nb ⁹⁵	1.706×10^6	3.290×10^6	5.143×10^6	7.129×10^6
Zr ⁹⁷	1.135×10^{12}	1.128×10^{12}	1.120×10^{12}	1.112×10^{12}
Nb ^{97m}	2.850×10^{12}	1.086×10^{12}	1.076×10^{12}	1.069×10^{12}
Nb ⁹⁷	5.655×10^{11}	6.391×10^{11}	6.831×10^{11}	7.224×10^{11}
Nb ⁹⁸	2.011×10^{11}	1.760×10^{11}	1.541×10^{11}	1.384×10^{11}
Mo ⁹⁹	3.004×10^{11}	3.998×10^{11}	2.993×10^{11}	2.998×10^{11}
Tc ^{99m}	7.407×10^9	1.224×10^{10}	1.698×10^{10}	2.161×10^{10}
Tc ⁹⁹	4.147×10^0	7.212×10^0	1.058×10^1	1.423×10^1
Nb ¹⁰⁰	6.001×10^{13}	5.957×10^{12}	5.913×10^{11}	5.869×10^{10}
Nb ¹⁰¹	5.340×10^{13}	5.212×10^{10}	5.087×10^7	4.965×10^4
Mo ¹⁰¹	3.960×10^{13}	2.707×10^{13}	1.684×10^{13}	1.048×10^{13}
Tc ¹⁰¹	2.150×10^{13}	2.619×10^{13}	2.422×10^{13}	1.989×10^{13}
Mo ¹⁰²	3.428×10^{13}	1.876×10^{13}	1.027×10^{13}	5.620×10^{12}

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Time After Shutdown: 30

40

50

60

Nuclide

Tc ¹⁰²	3.423×10^{13}	1.890×10^{13}	1.034×10^{13}	5.661×10^{12}
Ru ¹⁰³	2.075×10^{10}	1.075×10^{10}	2.074×10^{10}	2.074×10^{10}
Rh ^{103m}	1.678×10^9	2.670×10^9	3.549×10^9	4.327×10^9
Mo ¹⁰⁴	1.716×10^{13}	1.072×10^{12}	6.697×10^{10}	4.184×10^9
Tc ¹⁰⁴	1.089×10^{13}	9.117×10^{12}	6.310×10^{12}	4.300×10^{12}
Tc ¹⁰⁵	7.730×10^{12}	3.578×10^{12}	1.657×10^{12}	7.668×10^{11}
Ru ¹⁰⁵	3.948×10^{11}	5.235×10^{11}	5.739×10^{11}	5.888×10^{11}
Rh ^{105m}	3.785×10^{11}	5.156×10^{11}	5.711×10^{11}	5.882×10^{11}
Rh ¹⁰⁵	1.372×10^9	2.829×10^9	4.576×10^9	6.425×10^9
Ru ¹⁰⁶	1.418×10^8	1.418×10^8	1.418×10^8	1.418×10^8
Rh ¹⁰⁶	1.386×10^8	1.419×10^8	1.418×10^9	1.418×10^8
Ru ¹⁰⁷	1.702×10^{12}	3.970×10^{11}	8.797×10^{10}	1.949×10^{10}
Rh ¹⁰⁷	9.268×10^{11}	9.170×10^{11}	7.225×10^{11}	5.390×10^{11}
Pd ¹⁰⁷	2.455×10^0	4.258×10^0	5.806×10^0	6.988×10^0
Pd ¹⁰⁹	7.303×10^9	7.241×10^9	7.178×10^9	7.117×10^9
Ag ^{109m}	7.078×10^9	7.246×10^9	7.184×10^9	7.123×10^9
Pd ^{111m}	1.103×10^9	1.080×10^9	1.058×10^9	1.036×10^9
Ag ¹¹¹	1.168×10^8	1.754×10^8	2.189×10^8	2.513×10^8
Pd ¹¹¹	1.070×10^{11}	7.934×10^{10}	5.891×10^{10}	3.478×10^{10}
Pd ¹¹²	1.561×10^9	1.552×10^9	1.544×10^9	1.535×10^9
Ag ¹¹²	8.176×10^7	1.340×10^8	1.842×10^8	2.322×10^8

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Time After Shutdown:	30	40	50	60
Nuclide				
Ag ¹¹⁵	4.615×10^{10}	3.317×10^{10}	2.385×10^{19}	1.714×10^{10}
Cd ^{115m}	8.184×10^5	1.218×10^6	1.505×10^6	1.712×10^6
In ¹¹⁵	2.222×10^{-13}	0.0	0.0	0.0
Cd ¹¹⁵	3.269×10^8	4.040×10^8	4.391×10^8	4.982×10^8
In ^{115m}	1.060×10^7	1.964×10^7	3.009×10^7	4.147×10^7
Cd ^{117m}	1.144×10^{10}	1.101×10^{10}	1.060×10^{10}	1.020×10^{10}
In ¹¹⁷	4.849×10^8	7.517×10^8	9.943×10^8	1.206×10^9
Sn ^{117m}	5.517×10^1	1.353×10^2	2.721×10^2	4.708×10^2
Cd ¹¹⁷	1.271×10^9	1.987×10^9	2.560×10^9	3.036×10^9
In ^{117m}	1.698×10^8	3.142×10^8	5.853×10^8	6.740×10^8
Sn ¹²¹	1.791×10^9	1.784×10^9	1.776×10^9	1.769×10^9
Sn ¹²³	1.316×10^6	1.316×10^6	1.316×10^6	1.316×10^6
Te ^{125m}	2.579×10^1	5.256×10^1	8.499×10^1	1.201×10^2
Sn ¹²⁵	6.792×10^{10}	3.284×10^{10}	1.593×10^{10}	7.776×10^9
Sb ¹²⁵	8.980×10^5	1.216×10^6	1.370×10^6	1.444×10^6
Sb ^{126m}	$2.6-5 \times 10^2$	2.167×10^2	2.652×10^2	3.987×10^2
Sb ¹²⁶	7.317×10^1	6.578×10^{-1}	1.120×10^{-1}	1.664×10^{-1}
Sn ¹²⁶	1.885×10^2	1.885×10^2	1.885×10^2	1.885×10^2
Sn ¹²⁷	2.006×10^{11}	1.945×10^{11}	1.830×10^{11}	1.722×10^{11}
Sb ¹²⁷	4.061×10^8	6.590×10^8	8.977×10^8	1.122×10^9
Te ^{127m}	9.940×10^5	8.389×10^6	8.387×10^6	8.390×10^6

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Time After Shutdown:	30	40	50	60
Nuclide				
Te ¹²⁷	4.099×10^6	9.289×10^6	1.679×10^7	2.642×10^7
Sn ¹²⁸	1.078×10^{12}	9.551×10^{11}	8.458×10^{11}	7.490×10^{11}
Sb ¹²⁸	6.359×10^{11}	8.104×10^{11}	8.449×10^{11}	8.137×10^{11}
Sb ¹²⁹	4.363×10^{11}	4.743×10^{11}	4.626×10^{11}	4.511×10^{11}
Te ^{129m}	3.951×10^8	4.176×10^8	4.395×10^8	4.608×10^8
Te ¹²⁹	4.146×10^{10}	6.591×10^{10}	8.741×10^{10}	1.062×10^{11}
I ¹²⁹	4.298×10^{-5}	7.485×10^{-2}	1.344×10^{-1}	2.095×10^{-1}
Sn ¹³⁰	1.906×10^{13}	1.325×10^{12}	9.216×10^{10}	6.408×10^9
Sb ¹³⁰	1.747×10^{13}	9.963×10^{12}	3.988×10^{12}	1.519×10^{12}
Te ^{131m}	2.001×10^{10}	2.737×10^{10}	3.269×10^{10}	3.659×10^{10}
Te ¹³¹	4.680×10^{12}	6.204×10^{12}	6.643×10^{12}	6.451×10^{12}
I ¹³¹	3.244×10^9	6.574×10^9	1.047×10^{10}	1.441×10^{10}
Sn ¹³²	4.481×10^{13}	1.918×10^{12}	8.212×10^{10}	3.516×10^9
Sb ¹³²	4.478×10^{13}	7.530×10^{12}	5.292×10^{11}	3.028×10^{10}
Te ¹³²	1.597×10^{11}	1.986×10^{11}	2.005×10^{11}	2.005×10^{11}
I ¹³²	9.470×10^9	1.800×10^{10}	2.689×10^{10}	3.540×10^{10}
Te ^{133m}	1.540×10^{13}	1.348×10^{13}	1.180×10^{13}	1.032×10^{13}
I ¹³³	4.511×10^{11}	5.285×10^{11}	5.954×10^{11}	6.533×10^{11}
Xe ^{133m}	3.081×10^7	5.538×10^7	8.351×10^7	1.147×10^8
Xe ¹³³	5.475×10^8	9.843×10^8	1.484×10^9	2.040×10^9
Te ¹³⁴	2.523×10^{13}	2.147×10^{13}	1.827×10^{13}	1.555×10^{13}

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Time After Shutdown:	30	40	50	60
Nuclide				
I ¹³⁴	7.521×10^{12}	9.475×10^{12}	1.075×10^{13}	1.151×10^{13}
I ¹³⁵	2.932×10^{12}	2.882×10^{12}	2.832×10^{12}	2.784×10^{12}
Xe ^{135m}	1.818×10^{12}	1.423×10^{12}	1.249×10^{12}	1.101×10^{12}
Xe ¹³⁵	7.616×10^{10}	1.210×10^{11}	1.513×10^{11}	1.984×10^{11}
Cs ¹³⁵	1.635×10^1	1.685×10^1	1.747×10^1	1.848×10^1
I ¹³⁶	2.956×10^{13}	2.347×10^{11}	1.863×10^9	1.479×10^7
Ba ^{137m}	4.662×10^7	6.352×10^7	6.824×10^7	6.917×10^7
I ¹³⁷	5.721×10^{13}	2.276×10^6	9.051×10^{-2}	3.600×10^{-9}
Xe ¹³⁷	5.690×10^{13}	1.075×10^{13}	1.818×10^{12}	3.076×10^{11}
Cs ¹³⁷	6.174×10^7	7.462×10^7	7.683×10^7	7.720×10^7
Xe ¹³⁸	3.694×10^{13}	2.457×10^{13}	1.634×10^{13}	1.087×10^{13}
Cs ¹³⁸	1.257×10^{13}	1.597×10^{13}	1.676×10^{13}	1.609×10^{13}
Xe ¹³⁹	5.149×10^{13}	2.032×10^9	8.021×10^4	3.166×10^0
Cs ¹³⁹	5.433×10^{13}	2.812×10^{13}	1.356×10^{13}	6.535×10^{12}
Ba ¹³⁹	8.209×10^{12}	1.082×10^{13}	1.154×10^{13}	1.139×10^{13}
Ba ¹⁴⁰	6.826×10^{10}	6.823×10^{10}	6.821×10^{10}	6.818×10^{10}
La ¹⁴⁰	2.935×10^8	4.885×10^8	6.829×10^8	8.766×10^8
Ba ¹⁴¹	4.116×10^{13}	2.800×10^{13}	1.905×10^{13}	1.296×10^{13}
La ¹⁴¹	2.241×10^{12}	3.196×10^{12}	3.795×10^{12}	4.154×10^{12}
Ce ¹⁴¹	3.642×10^8	7.659×10^8	1.270×10^9	1.861×10^9
Ba ¹⁴²	5.014×10^{13}	2.507×10^{13}	1.253×10^{13}	6.267×10^{12}

~~CONFIDENTIAL~~
~~RESTRICTED DATA~~

Time After Shutdown: 30

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50

60

Nuclide

La ¹⁴²	7.740×10^{12}	1.006×10^{13}	1.071×10^{13}	1.057×10^{13}
La ¹⁴³	3.919×10^{13}	2.666×10^{13}	1.814×10^{13}	1.234×10^{13}
Ce ¹⁴³	2.437×10^{11}	3.645×10^{11}	4.326×10^{11}	4.837×10^{11}
Pr ¹⁴³	3.947×10^8	5.012×10^8	6.406×10^8	8.019×10^8
Ce ¹⁴⁴	2.950×10^9	2.950×10^9	2.950×10^9	2.950×10^9
Pr ¹⁴⁴	1.229×10^9	1.795×10^9	2.174×10^9	2.429×10^9
Nd ¹⁴⁴	3.642×10^{-12}	0.0	0.0	0.0
Ce ¹⁴⁵	3.791×10^{13}	3.763×10^{12}	3.735×10^{11}	3.708×10^{10}
Pr ¹⁴⁵	1.841×10^{12}	2.088×10^{12}	2.076×10^{12}	2.039×10^{12}
Ce ¹⁴⁶	2.272×10^{13}	1.380×10^{13}	8.378×10^{12}	5.088×10^{12}
Pr ¹⁴⁶	8.945×10^{12}	1.110×10^{13}	1.191×10^{13}	9.902×10^{12}
Pr ¹⁴⁷	2.119×10^{13}	1.189×10^{13}	6.676×10^{12}	3.747×10^{12}
Nd ¹⁴⁷	1.757×10^{10}	2.454×10^{10}	2.845×10^{10}	3.963×10^{10}
Pm ¹⁴⁷	1.003×10^5	2.088×10^5	3.440×10^5	4.943×10^5
Sm ¹⁴⁷	0.0	0.0	0.0	0.0
Ce ¹⁴⁸	1.630×10^{13}	4.962×10^8	1.510×10^4	4.595×10^{-1}
Pr ¹⁴⁸	1.630×10^{13}	7.078×10^{11}	2.024×10^{10}	5.784×10^8
Nd ¹⁴⁹	1.714×10^{12}	1.618×10^{12}	1.527×10^{12}	1.441×10^{12}
Pm ¹⁴⁹	5.744×10^9	9.352×10^9	1.275×10^{10}	1.595×10^{10}
Nd ¹⁵¹	3.348×10^{12}	1.964×10^{12}	1.153×10^{11}	6.762×10^{11}
Pm ¹⁵¹	2.554×10^{10}	3.597×10^{10}	4.200×10^{10}	4.546×10^{10}

Time After Shutdown:	30	40	50	60
<u>Nuclide</u>				
Sm ¹⁵¹	4.708×10^3	9.851×10^3	1.632×10^4	2.355×10^4
Sm ¹⁵³	1.051×10^{10}	1.048×10^{10}	1.046×10^{10}	1.043×10^{10}
Sm ¹⁵⁵	1.824×10^{11}	1.366×10^{11}	1.023×10^{11}	7.667×10^{10}
Eu ¹⁵⁵	1.030×10^6	1.552×10^6	1.943×10^6	2.236×10^6
Sm ¹⁵⁶	4.684×10^9	4.624×10^9	4.565×10^9	4.507×10^9
Eu ¹⁵⁶	1.115×10^7	1.260×10^7	1.403×10^7	1.543×10^7
Eu ¹⁵⁷	1.655×10^9	1.643×10^9	1.630×10^9	1.618×10^9
Eu ¹⁵⁸	5.584×10^9	4.975×10^9	4.433×10^9	3.949×10^9
Gd ¹⁵⁹	1.946×10^8	1.933×10^8	1.921×10^8	1.908×10^8
Gd ¹⁶¹	7.220×10^8	1.109×10^8	1.705×10^7	2.620×10^6
Tb ¹⁶¹	1.247×10^6	1.473×10^6	1.507×10^6	1.511×10^6

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