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SAFETY REPORT FOR THE
ASTRONUCLEAR WESTINGHOUSE
EXPERIMENTAL FACILITY
(PAX AND FCX)

(Title Unclassified)

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CHAPTER 1

INTRODUCTION

1.1 GENERAL

This document is a complete revision and consolidation of the safety analysis of the Permanent Assembly Experiment (PAX)*, Reference (1), and the safety analysis of the Flexible Critical Experiment (FCX), which was issued as Supplement 1 to Reference (1).

The present revision incorporates:

All significant changes to the facility, reactors, procedures, rules, and scope of experimental activities that have occurred since the original documents were issued.

Updating of calculations and other data that have been obtained in over two years of critical operations.

New changes that are necessary to meet the requirements of a broadening critical experimental program that has developed in the NERVA program.

Elimination of material which is not pertinent to the safety analysis, or which is now contained in related documents and would be redundant herein.** The intent is to make this safety report a more useful, manageable, and authoritative document.

(1) WANL-TNR-099 Safety Report for the Westinghouse Astronuclear Experimental Facility, May 1963.

* Formerly called the Pre-Assembly Experiment (PAX/PTX)

** Examples of deleted material now located in other documents include:

<u>Material</u>	<u>Document</u>
1) Certain Operational Instructions	WANEF-Operating Limits
2) Certain detailed procedures in original Chapters 6 and 9:	
a) Emergency procedures	WANEF-Reactor Operations Manual
b) Radiation safety procedures	
c) Maintenance procedures	
d) Fuel handling procedures	
3) Nuclear Safety Committee organization and procedures (formerly in Chapter 8)	WANEF-Criticality Hazards Control Procedure
	WANEF-Operating Limits

Rather than issue another Supplement of the Safety Report, WANL believes that a complete revision of the document would result in a clear and consistent system description and safety analysis. This document, therefore, constitutes the Safety Report for the performance of critical experiments in support of the NERVA program at the Westinghouse Astronuclear Experimental Facility (WANEF), by the Westinghouse Astronuclear Laboratory (WANL), under subcontract to the Aerojet-General Corporation and supersedes the original issue, addenda, and supplements. It is shown that a combination of design and procedural safeguards limits the magnitude of an accidental excursion by the PAX and FCX critical assemblies below the level where fission product release can occur. The experimental programs for PAX and FCX can, therefore, be conducted without risk to site personnel and to the general public.

1.2 SITE

WANEF is located at the Waltz Mill Site which is owned by Westinghouse. The site covers 850 acres and is located near the town of Waltz Mill, Pa., approximately twenty miles southeast of Pittsburgh. Besides critical experiment facilities, the site contains the Westinghouse Astronuclear Hydrogen Experimental Site, analytical chemistry laboratories, and extensive hot laboratory facilities.

The surrounding area is of relatively low population density, there being 350 people within a one-mile radius and 10,800 within five miles. The site, its surroundings, and its meteorology are discussed in Chapter 2.

1.3 FACILITY

The experiments will be performed in an existing test cell, which has concrete shielding walls. This facility, described in Chapter 3, was originally used for the performance of heavy water-low enrichment uranium oxide critical experiments by the Westinghouse Atomic Power Division. It has been used for PAX operations for more than two years.

Immediately adjoining the WANEF test cell is the CRX test cell in which light water-low enrichment uranium oxide critical experiments are performed under license CX-6. Also close by is the CES facility in which pool type critical experiments are performed on a WTR

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Astronuclear
Laboratory

WANL-TNR-099

Revision A

(Westinghouse Testing Reactor) type core under license CX-11. The work under CX-6 and CX-11 is under the direction of the Westinghouse Atomic Power Divisions (WAPD), whose organization with respect to WANL is discussed in Chapter 7. The WAPD facility is called the Westinghouse Reactor Evaluation Center (WREC).

1.4 REACTORS

The PAX and FCX reactor assemblies, on which low power critical experiments are performed, are contained in the WANEF test cell and are operable from a common console.

The PAX, a nuclear and mechanical mockup of the current NRX reactor, has been in operation since February 1964. It is described in detail in Chapter 4. The reactor core is a cylinder approximately 36 inches in diameter and 52 inches in height. It is fueled with highly enriched uranium (in the form of pyro-carbon coated uranium dicarbide beads) distributed in a graphite matrix, and is surrounded by a graphite reflector region and a beryllium reflector region. Control is exercised by means of poison plates attached to the beryllium drums in the beryllium reflector.

Distinguishing features of the PAX are its use of highly enriched uranium, its solid moderator, and its intermediate energy neutron spectrum.

Section 4.5 presents the nuclear characteristics of this reactor, while Chapter 8 contains the safety analysis and a discussion of its response to reactivity insertions.

The FCX reactor attained initial criticality in April 1966. Although similar to the PAX reactor in that the fuel materials, fuel concentrations and overall element shape are similar, it has a different mechanical design than the PAX and NRX assemblies so as to provide the flexibility required for variation of core materials, C/U ratios, reflector materials, and reflector thicknesses. Vertically moving control rods with fuel followers are used in the reactor. The design permits clean experiments to be performed, unperturbed by the varying quantities of extraneous material present in the actual reactors, thus mocking up rather simple analytical models of both NRX and NR reactors. The detailed description of the FCX is contained in Chapter 5, and its safety analysis in Chapter 9.

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1.5 EXPERIMENTAL PROGRAM

The following general types of experiments will be performed at WANEF:

PAX - Several general types of critical experiments can be performed with the PAX facility. The NRX reactors can be (1) mocked up using the reject components and fuel elements permanently assigned to WANEF, (2) assembled at WANEF using all reactor grade components, or (3) mocked up using both reject and reactor grade components. The PAX is assembled by loading poisoned fuel clusters into an assembled reflector and a critical approach is performed by the removal of the shipping poison wires. Low power experiments are then performed to assess the reactivity shimming required, measure the power distribution and control drum worths, obtain incremental and integral worths of various materials and structures in and around the reactor, determine the radiation environment, verify the worth of the shipping poison system, and determine the kinetics of the reactor. To date, such experiments have been conducted successfully on six different assemblies. In addition, a Proof Test Experiment (PTX) can be performed in the PAX test stand on a previously assembled NERVA reactor prior to its shipment to the Nuclear Rocket Development Station (NRDS) in Nevada. In a PTX the reactivity, drum worths, and power distribution experiments are performed.

FCX - Two general types of critical experiments can be performed on the FCX assembly. Bare and reflected cores will be operated to provide basic physics data for NERVA long range development objectives. The FCX can also be utilized for nuclear mockups of advanced NERVA reactor designs. Measurements similar to those conducted with the PAX will be performed.

The detailed experimental program and schedule, of course, depend upon the current requirements of the NERVA program.

1.6 POSTULATED ACCIDENTS

One can conceive of nuclear excursions, and the possible consequent release of fission products, from the following general causes:

- 1) Introduction of hydrogenous fluids into the cores.
- 2) Control malfunction (control element runout).
- 3) Excess reflection of a near-critical system by personnel or other massive objects.
- 4) Manual addition of excess reactivity during the course of an experimental procedure.
- 5) Core compaction.
- 6) Failure of in-reactor or out-of-reactor experiments.
- 7) Sabotage and acts of God.

In each of the situations analyzed either an inconceivable event is involved or multiple unlikely and unrelated failures of equipment and/or procedural safeguards are required to produce a nuclear excursion of any consequence.

1.7 SAFEGUARDS

To assure the freedom of the critical experiments from excursions caused by the items enumerated in Section 1.6, a number of design features and procedural safeguards have been developed over a period of time. Some have been utilized in PAX operations from the beginning (Items 1-3 following). Additional safety design features and procedural safeguards (Items 4-6) were introduced in 1965 to cover the installation of the FCX. The PAX auxiliary shutdown system (Item 7) was installed in 1966, and Items 8 to 10 are new.

- 1) Strict procedural control of liquids admitted into the test cell.
- 2) A control system allowing only slow reactivity insertions by one control element at a time.
- 3) Reactor scram interlocked with opening of the test cell. Shutdown reactivity requirements include an allowance for reflection by personnel.
- 4) A minimum control rod shutdown of 10\$ and a maximum excess reactivity of 50 ¢ for the FCX.

- 5) Use of scram interlock in the program patch panels to ensure that the proper connections are made from the console to the reactor to be operated.
- 6) Requirement that the reactor not connected to the control console have a minimum shutdown of 10\$.
- 7) An auxiliary shutdown system on PAX that can override a control drum rollout failure.
- 8) Use of a scram interlock on certain experimental devices to detect the onset of experiment failure.
- 9) Requirements that the PAX scram delay times from the neutronics and the experimental device interlocks (Item 8) be sufficiently short that any inadvertent power ramp is terminated before core heatup occurs.
- 10) The structural design of experimental supports have a minimum load safety factor specified.

1.8 SAFETY ANALYSES

A safety evaluation of the PAX assembly is presented in which several credible accidents are analyzed together with the administrative and engineered safeguards incorporated into the WANEF operating procedures to prevent such accidents. (An incredible accident is defined as one which requires the coincidence of two or more independent and unlikely events.) Also, the reduction in reactor shutdown margin due to control vane melting under excursion conditions is presented along with information on the threshold for fission product diffusion from the fuel. It is shown that the consequences of those accidents which have been judged to be credible, fall within the limitations imposed by control vane melting and the threshold for fission product diffusion. Thus there is no undue radiological hazard to the operating personnel or to the general public. The maximum credible accident now considered for PAX is one in which a control system failure allows a single drum runout which is terminated by a reactor scram initiated by the neutronics safety system, thus resulting in an energy release of 5×10^4 watt-seconds, a peak core temperature rise of about 0.05°C , and no release of fission products to the atmosphere.

By virtue of the operating limitation on the excess reactivity of the FCX assembly (0.50\$) even during the course of a credible accident, the FCX is not vulnerable to a reactivity excursion of any consequence. A safety analysis is presented to show that inherent design features as well as procedural safeguards ensure compliance with this restriction.

CHAPTER 2

SITE

2.1 SITE DESCRIPTION

2.1.1 Location

The Westinghouse Astronuclear Experimental Facility (WANEF) is located on the Westinghouse Waltz Mill Site in Westmoreland County, Pennsylvania near the town of Waltz Mill, approximately 20 air miles southeast of the suburbs of Pittsburgh. The position of the Waltz Mill site relative to the general Pittsburgh area is shown in Figure 2-1.

The 850 acre Westinghouse tract, shown in Figure 2-2, is bounded on the south by Sewickley Creek. It is divided by State Route 64117, a secondary road, which connects the towns of Waltz Mill and Madison while State Route 64164 (a secondary road) and U. S. Interstate Highway 70 traverse the southern sector of the site. The Pennsylvania Turnpike passes within two miles of the site boundary. Land usage in the vicinity of the site is devoted primarily to coal mining and farming which accounts for the low population density in this area.

As shown in Figure 2-3, WANEF is located within an eight acre exclusion area at the center of the tract. This area also contains the WREC (Westinghouse Reactor Evaluation Center) and CES (Critical Experiment Station) critical facilities, offices, hot cells, services buildings, and the Testing Reactor, now dismantled.

2.1.2 Population

The Waltz Mill Site is situated in an area of low population density. Population data presented herein is based primarily on the 1960 census. Comparison of these figures with those of the 1950 census shows a slight decrease in population due to curtailment of mining activities in the area.

The population distribution within one mile of the site is presented in Table 2-1. Communities within six miles of WANEF are listed in Table 2-2 and Figure 2-4; large cities and towns

Figure 2-1. Map of Pittsburgh Area

Figure 2-2. Topographical Map of Waltz Mill Tract

Figure 2-3. Waltz Mill Site Layout

beyond a six mile radius are tabulated in Table 2-3.

TABLE 2-1
POPULATION WITHIN ONE MILE OF WANEF

<u>Location</u>	<u>Population</u>
0-2,000 Ft. Circle	80 ⁽¹⁾
2,000 - 4,000 Ft. Annulus	100
4,000 Ft. - One Mile Annulus	250 ⁽²⁾

(1) Site Personnel

(2) These people are located primarily in the S-SW section.

2.1.3 Site Control

The Westinghouse facilities on the Waltz Mill site are enclosed by an eight foot high security fence. Admittance to this exclusion area is controlled by a security guard at the entrance gate on a 24 hour basis. All facilities are patrolled hourly after normal working hours by security personnel.

2.1.4 Site Activities

In addition to approximately 18 WANEF permanent operating staff personnel (refer to Chapter 7), the Waltz Mill site is currently (Summer 1966) staffed by 60 Westinghouse employees who are not associated with the NERVA program at the present time. A small permanent staff is engaged in criticality and exponential studies at the Westinghouse Reactor Evaluation Center (WREC) as experimental support for commercial power reactor design and development. A technical group engaged in hot laboratory and chemistry laboratory operations, in addition to service and maintenance personnel, comprise the remainder of this staff.

TABLE 2-2

POPULATION WITHIN SIX MILES OF WANEF

<u>City or Town</u>	<u>Population</u> [*]	<u>Distance from WANEF (Miles)</u>	<u>Direction</u>
Yukon	1062	1.1	W
Wyano	500**	2.0	SW
Madison	399	2.2	NNW
Hunker	380	2.5	ESE
Hutchinson	500**	2.8	W
New Stanton	1500	3.0	E
Ruffsedale	500**	4.0	SE
Smithton	649	4.5	SW
Fitz Henry	900**	4.5	SW
Herminie	1571	5.0	NW
Youngwood	2813	5.0	ENE
Alverton	600**	6.0	SE
West Newton	3982	6.0	W

* 1960 Census
 ** 1950 Census

<u>RADIUS</u>	<u>TOTAL POPULATION IN AREA</u>
2000 FEET	80 (WESTINGHOUSE SITE PERSONNEL)
4000 FEET	180
1 MILE	430
2 MILES	2,000
3 MILES	4,770
4 MILES	5,270
5 MILES	11,200
6 MILES	15,790

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Figure 2-4. Population Distribution of Waltz Mill Area

TABLE 2-3

LARGE POPULATION CENTERS NEAR WANEF

<u>City or Town</u>	<u>Population*</u>	<u>Distance from WANEF (Miles)</u>	<u>Direction</u>
Mount Pleasant	6, 107	7.5	SE
Greensburg	17, 383	8.0	NE
Jeannette	15, 565	8.0	N
Irwin	4, 270	8.5	NW
Scottdale	6, 244	9.5	SW
McKeesport	45, 489	12.0	NW
Clairton	13, 389	13.0	NW
Elizabeth	2, 597	13.5	WNW
Connellsville	12, 814	14.0	S
Latrobe	11, 932	16.5	ENE
Pittsburgh ⁺	604, 332	20.0	NW

* 1960 Census

⁺ Population of Pittsburgh including suburbs (Allegheny County and contiguous thickly settled portions of Butler, Washington, and Westmoreland Counties) is approximately 2, 000, 000.

In addition, a hydrogen flow test facility is located approximately 3000 feet from WANEF on the northeast sector of the Westinghouse tract and has a permanent staff of approximately 11 engineers and support personnel.

2.2 METEOROLOGY

2.2.1 General Information

The regional weather conditions at Waltz Mill are determined by its location in the foothills of the Allegheny Mountains. Waltz Mill has a humid, continental type of climate modified slightly by its nearness to the Atlantic Seaboard and the Great Lakes.

The predominant air which influences the climate of the Waltz Mill area has a polar continental source in Canada and moves in upon the region by way of tracks which vary from almost due north, from the Hudson Bay region, to a long westerly trajectory resulting from polar out-breaks into the Rockies which progress eastward.

During the summer season, there are frequent invasions of air from the Gulf of Mexico with resulting spill of warm, humid weather. During winter, air from the Gulf reaches this far north only occasionally.

The meteorological data presented in this section is mainly collected at the Allegheny County Airport which is an aerial distance of approximately 17 miles from the Waltz Mill site. This airport has been a first order Weather Bureau Airport Station (WBAS) since 1935.

The use of the WBAS data is justifiable from the following considerations:*

- 1) The aerial distance between the Waltz Mill Site and WBAS is only 17 miles; both are located within a uniformly irregular terrain of low hills of nonprominent contours. The nearest coherent mountain ridge of the Appalachian Mountain Range, the Chestnut Ridge, is about 15 miles to the east of the Site and oriented from SW to NE. Thus, there exists no major topographic feature between the Site and WBAS that would divide the general climatic regime of the area.

*

Personal Communication, H. H. Neuberger, Meteorological Consultant, Pennsylvania State University, 12 March 1963.

- 2) An analysis of simultaneous hourly wind observations at the site and at WBAS over a period of eight months between 1956 and 1958 (Tables 2-4 through 2-7 and Figure 2-5) demonstrates that the general distribution of wind speed for various wind directions is quite similar at both places, except that the average wind speeds at the site with winds from the sector between NW and E are somewhat less than those at WBAS. This, however, seems attributable to the location of the wind instruments at the site, about 350 feet to the NNW of the actual reactor location and between two hills, one to the NNW, the other one to the ENE of the wind tower. These hills prevent the free access of wind from the sector mentioned. Considering the fact that the actual site is in a less obstructed environment, it can safely be assumed that the average wind speeds there would not be significantly different from those at WBAS.
- 3) To further establish the uniformity of the climate over the general area, including both the site and WBAS, a comparison can be made of the precipitation pattern observed at several cooperative U. S. Weather Bureau stations in the immediate vicinity. The uniformity of the precipitation climate is suggested by the data in the Rainfall Intensity-Frequency Regime, Part 3 - Middle Atlantic Region (USWB Technical Paper No. 29, July 1958), in which the isopluvial charts show that there is no significant precipitation gradient in the region under consideration. From this publication the 2-year, 24-hour precipitation maximal for Pittsburgh and several co-operative stations in the vicinity are listed together with normal annual precipitation totals (from Climatic Data for Pennsylvania, USWB 1958) in Table 2-8. Considering the local variability of precipitation at each of the stations, the difference between the stations is not significant. Since the precipitation extremes and normals for these stations are so similar, the annual distribution of precipitation is expected to show similar uniformity for this region, and thereby supply further evidence of the climatic coherence of the area.

TABLE 2-4

PERCENTAGE FREQUENCY AND AVERAGE VELOCITY OF SURFACE HOURLY
WIND OBSERVATIONS IN VARIOUS SPEED GROUPS BY SEASONS

PITTSBURGH, PA., WBAS 1942-1950

Velocity:	Percentage Frequency (%)					Average Velocity (mph)
	Calm	1-3 mph	4-12 mph	13-24 mph	25 and above	
Season:						
Winter D-J-F	0.5	3.7	59.5	33.1	3.1	11.5
Spring M-A-M	0.4	4.4	61.3	31.0	2.8	11.2
Summer J-J-A	0.7	6.6	79.8	12.7	0.2	8.3
Fall S-O-N	0.5	4.7	73.0	21.2	0.6	9.6
Average	0.5	4.8	68.4	24.5	1.7	10.1

TABLE 2-5

PERCENTAGE FREQUENCY AND AVERAGE VELOCITY OF SURFACE HOURLY
WIND OBSERVATION IN VARIOUS SPEED GROUPS BY MONTHS

WALTZ MILL, PA., 1956-1957

Velocity:	Percentage Frequency (%)					Average Velocity (mph)
	Calm	1-3 mph	4-12 mph	13-24 mph	25 and above	
September	40.96	9.04	46.93	3.07	0	7.9
October	35.48	5.43	57.92	.36	0	8.2
November	22.08	6.71	47.3	19.97	3.93	10.4
December	23.27	3.46	42.33	30.94	0	11.8
January	11.6	4.51	28.11	39.5	16.37	13.3
February	24.9	10.95	47.64	11.79	4.71	8.9
March	8.81	5.51	33.06	28.8	23.79	11.6
Average	23.87	6.52	43.33	19.2	6.97	10.3

TABLE 2-6

PERCENTAGE FREQUENCY OF SURFACE HOURLY WIND OBSERVATION
 BY DIRECTION AND SEASONS AND AVERAGE WIND SPEED BY DIRECTION
 AT WBAS, PITTSBURGH, PA., 1942-1950

	Percentage Frequency (%)															
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
Winter D-J-F	2.6	2.7	2.7	2.4	3.2	3.5	2.8	5.4	8.5	13.4	7.8	11.7	9.3	11.5	8.2	3.4
Spring M-A-M	4.7	3.4	2.9	2.2	3.1	4.2	3.7	6.6	9.2	11.2	7.6	7.9	7.7	9.3	9.8	6.3
Summer J-J-A	5.8	4.0	3.4	2.6	3.2	4.0	4.0	6.4	11.7	15.4	8.4	5.8	5.5	5.9	7.0	5.9
Fall S-O-N	4.1	3.3	3.5	3.5	4.1	5.4	4.7	7.6	11.1	12.8	7.2	7.3	6.0	8.0	6.7	4.3
Average (%)	4.3	3.2	4.1	2.9	3.4	4.2	3.8	6.5	10.1	13.1	7.7	8.2	7.1	8.7	7.9	5.0
Average Annual Velocity (mph)	7.98	7.93	7.92	8.34	7.74	8.69	8.45	8.79	8.77	10.75	11.38	12.38	11.57	12.29	10.55	9.63

TABLE 2-7

PERCENTAGE FREQUENCY OF SURFACE HOURLY WIND OBSERVATION
 BY DIRECTION AND MONTH AND AVERAGE WIND SPEED BY DIRECTION
 AT WALTZ MILL, PA., 1956-1957

	Percentage Frequency (%)							
	N	NE	E	SE	S	SW	W	NW
September	21.54	18.41	6.14	1.6	3.82	12.73	23.38	9.37
October	20.77	14.51	14.93	13.75	6.59	4.86	12.22	7.29
November	8.26	7.54	6.17	4.17	5.37	17.8	40.0	7.21
December	11.68	5.42	4.17	2.08	5.21	28.76	27.31	11.25
January	16.88	3.16	1.86	0.53	1.05	19.52	42.92	9.49
February	20.39	11.59	4.62	2.31	9.72	4.17	33.84	13.42
March	18.15	12.31	13.52	5.86	12.3	2.22	23.19	1.82
Average (%)	16.81	10.42	7.34	5.86	6.29	12.87	28.98	8.55
Average Velocity (mph)	3.42	4.27	4.3	7.07	7.6	11.23	10.76	3.35

TABLE 2-8

PITTSBURGH AREA PRECIPITATION - (INCHES)

<u>Station</u>	<u>24-Hour Maximum</u>	<u>Normal Annual</u>
Pittsburgh City	2.80	36.26
Pittsburgh Airport	2.35	36.92
New Stanton	3.41	37.67
McKeesport	2.26	36.80
Bruceton	2.22	38.32
Newell	2.82	38.60
Donora	2.71	37.94

- 4) The WBAS upper air data should be applicable to the site in view of the low topographic contours of the region and the absence of any major mountain range between WBAS and the site. Information obtained from the analysis of winds aloft and vertical temperature gradients at WBAS should therefore, be applicable to the site.

2.2.2 Precipitation

Based on a study of 80 years of precipitation data for the Pittsburgh area (15 years at the Airport and 65 years in downtown Pittsburgh) approximately 36 inches of precipitation may be expected each year.

Precipitation is generally uniform throughout the year with the following seasonal variations: Winter 8.2 inches, Spring 9.4 inches, Summer 11.1 inches, and Fall 7.5 inches. During this period, 10.8 inches was experienced during one month and 4.08 inches was measured during one 24-hour period.

During the winter months, about 25 percent of the precipitation occurs as snow and there is about a 50 percent chance of measurable precipitation on any given day. The first appreciable snowfall is generally late in November and usually the last occurs early in April. Snow

BASED ON AVERAGE SURFACE WIND AT ALLEGHENY COUNTY AIRPORT FROM
WEATHER BUREAU AIRPORT STATION RECORDS 1942 TO 1950

- ALLEGHENY COUNTY AIRPORT 1942-1950
- - - - - ALLEGHENY COUNTY AIRPORT (8 MO. PERIOD 1956-1957)
- . - . - WALTZ MILL (8 MO. PERIOD 1956 AND 1957)

Figure 2-5. Waltz Mill-County Airport Wind Rose
(Direction and Distance Effluent Could
be Expected to Travel in 1 Hour)

lies on the ground an average of 33 days during the year. Snowfalls of 36.3 inches in one month and 17.5 inches in a single 24-hour period have been experienced.

Seven months of the year, from April through October, sunshine is possible more than 50 percent of the time. During the remaining five months, cloudiness is heavier because the track of migratory storms from west to east is closer to the area and because the northwest winds tend to blow humid air from across the Great Lakes. Thunderstorms occur during all months excepting mid-winter, and have a maximum frequency in mid-summer.

For use in studying possible rainout of airborne contamination, a wind rose was prepared for all hourly wind data collected at the Pittsburgh Airport during the period 1943-1950, when precipitation was actually falling at the time of observation. These data are presented in Table 2-9. As can be noted, precipitation occurs most frequently with winds from the WNW except during the summer when the most frequent direction is SSW. On the average, approximately 48 percent of the wind directions during rainy weather are recorded as WSW through NW, with a secondary maximum of approximately 20 percent in the zone S through SSW. Very few winds (approximately 9 percent) will occur from the directions NE through E when precipitation is falling.

2.2.3 Surface Winds

Percentage frequencies of hourly wind directions and speeds at the Airport for the period 1942-50, summarized by seasons, are presented in Tables 2-4 and 2-6*. Based on these data, less than 4 mph winds at Waltz Mill should be expected only about 5 percent of the time over a period of a year, while nearly 70 percent of the time, wind speeds are to be expected (Table 2-4) in the range between 4-12 mph. The prevailing wind flow can be expected to be from the south through southwest direction (approximately 30 percent of the time), based on the data contained in Tables 2-6 and 2-7. An additional 30 percent should be made up with winds from the west-southwest through northwest. Very few winds occur in the zones from northeast through southeast and there does not appear to be any serious shift in average direction from

* Wind data is tabulated as wind coming from the listed direction.

TABLE 2-9
PERCENTAGE FREQUENCY OF HOURLY WIND OBSERVATIONS
WHEN PRECIPITATION IS OCCURRING BY DIRECTIONS AND SEASONS
WBAS, PITTSBURGH, PA., 1943-1950

	Percentage Frequency (%)															
	<u>N</u>	<u>NNE</u>	<u>NE</u>	<u>ENE</u>	<u>E</u>	<u>ESE</u>	<u>SE</u>	<u>SSE</u>	<u>S</u>	<u>SSW</u>	<u>SW</u>	<u>WSW</u>	<u>W</u>	<u>WNW</u>	<u>NW</u>	<u>NNW</u>
Winter D-J-F	2.1	2.1	1.7	2.1	3.0	3.8	2.4	5.2	5.1	9.9	6.3	15.2	11.9	15.9	9.7	3.8
Spring M-A-M	3.6	2.3	2.1	2.0	3.5	5.0	3.8	5.8	6.6	9.1	6.0	9.0	9.8	13.5	12.0	5.8
Summer J-J-A	5.9	5.2	3.8	3.2	3.5	3.1	4.0	6.7	12.2	15.2	7.0	5.1	5.9	6.7	6.9	5.8
Fall S-O-N	2.9	2.4	3.2	4.7	2.3	3.6	3.2	5.4	8.1	12.1	6.2	9.2	8.0	14.4	9.5	4.9
Average (%)	3.8	3.0	2.7	3.0	3.1	3.9	3.4	5.8	8.0	11.6	6.4	9.6	8.9	12.6	9.5	5.0

season to season of the year. It is interesting to note that the average velocity is slightly stronger in the westerly quadrant.

2.2.4 Upper Air Winds

Wind roses were prepared for the 1000 and 3000 meter mean sea level surfaces and are presented in Table 2-10. These clearly indicate that most frequent upper air winds over Pittsburgh are from the westerly quadrant. At 1000 meters above sea level, approximately 65 percent of the winds should be expected from the SW through WNW directions in the winter and approximately 60 percent in the summer.

Radiosonde observations at the County Airport (WBAS), which are taken twice daily (10 PM and 10 AM) respectively and which obtain data on the temperature, pressure, and relative humidity vertically through the atmosphere are also tabulated. For the region below 3000 feet, thermal gradients were subdivided into three categories:

- Lapse - Thermal gradient $\leq -2^{\circ}\text{C/Kilometer}$
- Inversion - Thermal gradient $\geq +2^{\circ}\text{C/Kilometer}$
- Isothermal - Thermal gradient between -2°C/Km. and $+2^{\circ}\text{C/Km.}$

From Table 2-11, lapse conditions should be expected approximately 69 percent of the time directly above the earth's surface at observational time, and isothermal or inversion conditions approximately 31 percent of the time. Measurements indicate that many more inversions occur at 10 PM than at 10 AM; although it is interesting to note that at the 10 PM observations the highest frequency of inversion comes in the fall and summer months while at the 10 AM observations, the highest frequency of inversions is during the cold, or winter months. The inversions during the winter are likely to be persistent while those in the summer and fall are only nocturnal except during prolonged periods of anticyclonic stagnation. Periods of such stagnation have been discussed by Wexler.*

It is interesting to note that most of the isothermal and inversion conditions occurred at night. The frequency of these conditions is more than a factor of 10 greater at night, which

* Wexler, H., Air Pollution in Donora, Pa., Public Health Bulletin 306, pp.171-172 (1949)

TABLE 2-10

PERCENTAGE FREQUENCY OF WINDS ALOFT OBSERVATIONS AT 1000 AND 3000 METERS,
MEAN SEA LEVEL, FOR WINTER (D-J-F) AND SUMMER (J-J-A) BY DIRECTION
AT WBAS, PITTSBURGH, PA., 1934-1939

	1000 Meters		3000 Meters	
	Winter	Summer	Winter	Summer
N	2.3	4.8	1.6	5.2
NNE	1.5	3.7	1.3	2.8
NE	1.3	3.7	0.3	2.2
ENE	1.3	2.6	0.4	1.6
E	1.3	2.5	0.1	2.1
ESE	1.4	2.5	0.1	1.2
SE	1.6	3.7	0.4	1.4
SSE	2.4	3.6	0.5	1.4
S	4.3	4.0	1.6	2.1
SSW	6.3	5.5	2.9	3.9
SW	13.7	10.7	7.2	7.4
WSW	18.3	15.1	17.8	12.2
W	20.4	13.8	29.4	19.0
WNW	12.7	9.5	22.1	15.8
NW	7.3	8.1	10.8	13.7
NNW	3.9	6.1	3.6	7.8
Calm	0.1	0.4	0.0	0.2

is significant in that operations of the critical facility are more likely during the day.

Table 2-11 presents a further breakdown of inversions and isothermal conditions for the following height intervals in the atmosphere: surface to 499 feet, 500 to 1499 feet, and 1500 to 3000 feet above the surface. Inversions or isothermal conditions may actually exist in more than one of these levels at the same time, so that they cannot be compared directly. For the

TABLE 2-11

COMPARISON OF VARIOUS CLASSES OF LAPSE RATES BY
SEASONS AND OBSERVATIONAL TIMES
WBAS, PITTSBURGH, PA., MAY 1943 THROUGH DECEMBER 1949

		SEASONS					% Total** Observations
		Time of Reading*	D-J-F	M-A-M	J-J-A	S-O-N	
Lapse							
Surface to first significant level	03Z	369	328	265	281	1243	33.0
	15Z	305	351	348	338	1342	35.7
Isothermal							
Surface to first significant level	03Z	63	82	117	98	360	9.5
	15Z	12	2	1	5	20	0.5
Inversions							
Surface to first significant level	03Z	141	177	203	215	736	19.6
	15Z	35	9	1	16	61	1.6
Inversion and/or							
Isothermal surface to 499 feet	03Z	204	245	306	319	1071	28.5
	15Z	74	19	6	51	150	4.0
500 to 1499 feet	03Z	37	39	16	23	115	3.0
	15Z	52	29	19	42	142	3.8
1500 to 3000 feet	03Z	108	36	10	33	187	5.0
	15Z	82	53	25	58	218	5.8
<hr/>		<hr/>	<hr/>	<hr/>	<hr/>	<hr/>	<hr/>
Total Observations Available	03Z	573	587	585	594	2339	62.2
	15Z	352	362	350	359	1423	37.8
						<hr/>	<hr/>
						3762	100.00

* Z refers to Greenwich time. 03Z and 15Z equal, respectively, 10 PM and 10 AM local time.

** "Total" column is the frequency of occurrence of a given class of observations over the period considered. "% of Total" column is the percentage of all observations in which a given class of observations occurred during the period considered.

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TABLE 2-12

FREQUENCY OF THICKNESS (METERS) OF INVERSION AND ISOTHERMAL
LAYERS WHEN THEY ARE BASED AT THE SURFACE OR BELOW 500 FEET
WBAS, PITTSBURGH, PA., MAY 1943 THROUGH DECEMBER 1949

Layer Thickness	0-99	100-199	200-349	350-499	500-699	700-999	1000-1299	1300-1799	1800-2599	Total
Time of Readings	Number of Observations									
03Z	96	365	361	147	63	24	9	4	2	1071
15Z	14	38	53	23	20	7	1	3	1	150

same time period as that of Table 2-10, the thickness of the inversions or isothermal layers which had their base in the region between the surface and 500 feet are tabulated in Table 2-12. It should be noted that 75 percent of the stable layers which occurred in this 6 1/2 year period were less than 350 meters thick.

Care must be taken in using these data since they were taken at the same time as the radiosonde observations and do not necessarily indicate the maximum thickness that may have occurred at other times.

2.3 HYDROLOGY

2.3.1 Ground Water

The general drainage of the area is westward from the mountain ridges located to the east. Ground water occurs mostly in the massive sandstone, and also in the alluvium adjacent to the streams. The shales offer low water quantities in joints and fractures. An examination made of the site for evidence of ground water elevation showed that in the Sewickley Creek and adjacent tributaries ground-water level is at least equal to the elevation of the water in the creek beds. Two types of formations exist at the site. One is the alluvium adjacent to Sewickley Creek forming the flood plain, and the other is the Conemaugh sedimentary rocks with a thin mantle of residual soil formed by the weathering of these rocks. The alluvium near the creek with an approximate thickness of 0 to 15 feet consists mainly of clay, silt and fine-grained sand with stone fragments. This zone is not consistently water-bearing and frequently lies above the saturated zone. Where saturated, the yields are small because the strata are too fine-grained. The sedimentary rocks at this location consist of clayey and sandy shales, with the Saltsburg sandstone occurring in the area adjacent to Sewickley Creek. The shale yields only a small supply of water, while the massive Saltsburg sandstone can be considered as a reliable bedrock aquifer.

In general, it would be expected that the sub-surface ground water movements would be toward Sewickley Creek and its tributaries. These movements would be expected to be slow, in view of the impervious character of the shaly bedrock.

2.3.2 Surface Water

The water-shed area extending upstream from Waltz Mill is approximately 100 square miles in extent. The residual silty clay overburden found over most of this area has a high retention capacity. This is probably also true for the bedrock shales because of their low permeability and slow-draining features. The silt and sand in the alluvium of the flood plain adjacent to Sewickley Creek are more free-draining.

The Sewickley Creek has a flow at its discharge into the Youghiogheny River of 30,000,000 gallons per day. The flow rate is much lower at the site, which is approximately twelve miles above the confluence of the two streams.

2.3.3 Floods

The highest known flood of Sewickley Creek occurred in October 1954 when it reached a level approximately 50 feet below the elevation of the Waltz Mill site. The tributary to Sewickley Creek which traverses the site is sufficiently small that it does not present a flood threat, even at high flows.

Storms with several inches of rainfall in a few hours have been experienced in this general area. In order to divert drainage from neighboring hills during such occurrences a three foot interceptor trench is located near the exclusion area, as shown in Figure 2-3, and discharges into the tributary. Furthermore, the area in the immediate vicinity of WANEF has been contoured to ensure good drainage.

The possibility of a flood in the WANEF facility is therefore considered very remote.

2.3.4 Public Water Supplies

The residences within a five mile radius of the reactor site are served from three general water sources: The Westmoreland Water Authority, Herminie community water system, and individual sources in the form of wells or cisterns. The Westmoreland Water Authority supplies water to those communities requesting it and to individual residences that might be along the route of the water line. The Herminie community water serves only Herminie. Table 2-13 lists the towns within a five mile radius of the Waltz Mill site serviced by the Westmoreland

TABLE 2-13

TOWNS AND ACCOUNTS SERVICED BY THE WESTMORELAND
WATER AUTHORITY (Within a Five Mile Radius of the Site)

<u>Towns</u>	<u>No. of Accounts</u>
New Stanton	130
Old Stanton	25
Hunker	90
Youngwood	780
Ruffsedale	155
Yukon	215
Madison	120
Arona	20
Foxtown	105
Wendel -Herminie II	100

Water Authority. Wyano, Smithton, Middleton, Hutchinson and Waltz Mill are the principal communities which do not have central water systems and rely on private wells and cisterns for water supply.

Four major reservoirs serve the area. These are located beyond a five mile radius of the site in Herminie (Howell's Dam), Slickville (Beaver Run Dam), Mt. Pleasant (Bridgeport), and Scottdale (Green Lick). The latter three are part of the Westmoreland Water Authority system.

It should be noted that the hydrological and geological characteristics of the site are favorable toward minimizing any contamination of public water sources. For example, no springs have been observed in the area, even in locations where sizable excavations have been made for purposes of highway or railroad construction.

The subsurface ground water movements are toward Sewickley Creek and its tributaries which are now being used as waste streams by the nearby communities and mining installations. An additional advantage is the fact that the residual silty clay overburden found over all the area of the site has a high retention capacity permitting radioactivity decay during storage.

2.4 GEOLOGY

2.4.1 Geology of the Site

The site is located near Sewickley Creek, which is one of the major creek tributaries of the Youghiogheny River. The site is located on a small plateau; elevation 1000 feet above sea level; bounded on all sides by low lying hills. Peak elevation of these hills in the vicinity of the site is about 1300 feet.

Drainage characteristics of the site are good, as evidenced by the topographical relief map shown in Figure 2-2. The thickness of the soil mantle above the weathered bedrock is relatively thin throughout the site. Spot checks performed during various construction projects on the site revealed that the residual soil overburden varies from zero to four feet in thickness over much of the area. The alluvial deposits in the Sewickley Creek flood plain vary from about 10 to 20 feet in depth and from 0 to 500 feet in width. The residual overburden is primarily a silty clay formed as a result of weathering of the underlying shale. The clay-shale bedrock has a low water content due to the relatively low fracture and joint occurrence.

2.4.2 Mining

Drill surveys of the site area have shown that the typical underlying ground is composed of shale and sandstone, occurring in successive layers, to a depth of about 200 feet. Thereafter, the shale and sandstone beds alternate with shallow coal beds. A mining survey of the site has indicated that the first coal bed occurs at a depth of 190 feet. This particular bed is known as the Freeport Seam. A lower Freeport coal bed is located 50 to 60 feet below the upper bed. Below this, other coal beds occur intermingled with the shale and sandstone beds. It has been definitely established that there has been no coal mining under the area upon which construction has taken place.* Furthermore, mineral rights to this area have been purchased in order to prevent undermining at some future date.

*Mining survey performed by Mr. N. G. Alford, Consulting Mining Engineer

2.4.3 Seismology

The western portion of Pennsylvania has been free of both major earthquakes and minor tremors of local origin.

Shocks of strong intensity (VII to VIII on the Rosse-Jerel Scale) which caused moderate damage to buildings in Utica, New York (about 300 miles to the NE) and in Lima, Ohio (about 250 miles to the W) were recorded in August 1929 and September 1884, respectively.

Earthquakes of this intensity would result in accelerations of approximately 0.1 g. However, the accelerations are greatly reduced if the structure is located on rock or close to hilly country. The site, located in hilly country, and the site buildings, located virtually over bedrock, are therefore ideally situated against the hazard of earthquakes. In light of this information, it is felt that the possibility of seismic tremors in this area, of an intensity to cause concern, is extremely small and too remote to be considered a hazard.

CHAPTER 3

FACILITIES

3.1 GENERAL

Existing Westinghouse experimental facilities have been modified, resulting in the divided facility discussed in Paragraph 3.2.1. This separation provides for the independent operation of critical facilities under the cognizance of the Westinghouse Atomic Power Divisions (WAPD) and the presently described critical experimentation under the cognizance of WANL. The WANL facility contains a test cell, a vault, a fuel assembly room, a counting room, storage space, and offices which are described in Section 3.2.

General building services, shared with the Westinghouse Reactor Evaluation Center (WREC), are detailed in Section 3.3. The reactor control circuitry for each facility is completely independent. However, there is a common building alarm system for emergencies and the two facilities are connected by an intercommunications system. Chapter 7 discusses the relationship between WREC and WANEF in more detail.

A unique aspect of the NERVA critical experimentation is the requirement for handling fully assembled NERVA reactors (PTX), (refer to section 4.4) major components thereof (PAX), and flexible critical experiments (FCX). Section 3.4 describes the major support equipment for these assemblies.

Finally, special experimental equipment, special instrumentation for radiation monitoring and water detection, alarm systems, and fire protection are discussed in Sections 3.5 through 3.7.

3.2 PHYSICAL DESCRIPTION OF FACILITIES

3.2.1 General

Figure 3-1 shows the complete Westinghouse Waltz Mill Site. An exclusion area restricted by an 8 foot security fence is indicated by the shaded area. Figure 3-2 is a plan view of the WANEF-WREC facility including the earthen shield mound and fence. The earthen radiation shield consists of local soil material approximately 12 to 17 feet high, 30 feet in width at the base, 10 feet in width at the top, and 200 feet in length. This mound provides direct line shielding from either side of the facility to personnel outside the exclusion area at the rear of the building. Figure 3-2 also indicates building entrances and exits provided for both WREC and WANEF personnel.

As can be seen in the photograph (Figure 3-3), the basic structure of the building is standard 8 inch concrete building block faced with red hard-fired brick. The test cell walls are fabricated from poured reinforced concrete and rest on footer bases capable of supporting all necessary walls, internal structures, and building equipment. The strength of concrete specified for various areas of the modified facility was in accordance with the ASTM Specifications - designation C-39 and C-31, latest amendment.

In general, all material and construction procedures conform to the requirements and procedures of industrial building codes as established by applicable respective industries and geographic localities. These codes are as follows: structural steel - AISC; concrete - AECI; electrical - National Electric Code; plumbing - code identical to requirements of the city of Pittsburgh, Pennsylvania; general - Pennsylvania State Department of Labor and Industry building regulations.

Figure 3-4 is a layout of the WANEF rooms, passageways, laboratories, and the test cell. Two exits are provided, one located at the front (west side) of the building for general use and an emergency exit located at the side (south side) of the building. Both doors provide escape exits during an emergency. All exits (including truck doors) into the rear area are interlocked with the reactor control system (see Sections 6.3 and 6.7 for details).

Figure 3-1. Westinghouse Waltz Mill Site Map

Figure 3-2. WANEF-WREC Building Layout

Figure 3-3. WANEF-WREC Facility, Aerial View

Figure 3-4. WANEF Facility Layout

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An 8 foot fence divides the rear apron within the exclusion area adjacent to the east side of the WANEF-WREC facility. This fence acts as a security barrier and is equipped with various indicators to warn personnel of radiation hazards due to operation of either the WANEF or the WREC reactors. External warning devices are described in Section 6.7 and radiation levels around the facility for PAX and FCX operations are given in Paragraph 7.4.3.

3.2.2 Test Cell

3.2.2.1 Physical Description

Elevation drawings of the WANEF test cell are shown in Figures 3-5 and 3-6. The height of the cell above the ground level is 52 feet and the basement depth is 22 feet 6 inches. Above ground level, the north and west walls are 4 feet 6 inches thick to a height of 20 feet. Above this, the west wall is cut back to a thickness of 2 feet and the north wall to 1 foot for the remaining height of the test cell. The 4 foot-6 inch north wall provides radiation shielding for personnel who may be working in the adjacent WREC test cell during operation at WANEF. The east and south walls are 2 feet thick to a height of 20 feet above ground level and are then 1 foot thick above this level. The basement floor is a 6 inch thick concrete slab and the cell roof is 9 inch thick reinforced concrete covered with a mixture of tar and gravel. A covered ventilator opening in the roof of the test cell, approximately 30 inches square, is equipped with an exhaust fan and vent. Flashing, bonded to the roof and rising approximately 15 inches above roof level, prevents water leakage into the opening.

The 6 foot 6 inch wide access and storage area between the WREC and WANEF test cells is accessible only from the WREC facility. However, a 12 inch thick poured concrete slab above this area serves as a storage shelf for reactor fuel and miscellaneous experimental hardware for the WANEF facility. This storage space, 11 feet wide and extending the entire east-west length of the test cell, is designed

Figure 3-5. Test Cell, North-South Elevation

Figure 3-6. Test Cell, East-West Elevation

to hold a uniformly distributed load of 50 tons. Fuel clustering will occasionally be conducted in this area. Two fuel storage racks are bolted to the cell wall and shelf floor. When loaded each has a minimum safety factor of 20. As shown in the isometric projection (Figure 3-7), access to the storage shelf is provided by a stairway from the PAX test stand. Access to the overhead crane is provided by a ladder from the storage shelf.

Two doors provide access to the test cell: a 20 foot square electrically operated roll-up truck door, located in the east wall, and a personnel entrance, located in the south wall. The roll-up door is of 18 gauge galvanized metal with interlocking ribs, designed to withstand a loading of 25 pounds per square foot. The sides of the metal door jamb are recessed into the concrete wall to provide rigidity when closed. The personnel door is a "B" label hollow metal fire door which opens from the labyrinth into the cell. A seal is provided such that pressure within the cell assists in sealing the opening. The personnel door, formerly located about 1-1/2 feet from the west wall, has been moved to a point near the center of the south wall, as shown in Figure 3-4. The door was relocated because the FCX test stand did not permit adequate clearance for movement of personnel and equipment through the door. Because of the proximity of the original door to the reactor, neutron and gamma radiation levels in the entrance labyrinth would have been abnormally high during reactor operations. It is also believed that radiation levels in the counting room and control room would have increased due to scattering. It should be noted that no personnel are allowed in the labyrinth area during reactor operations. The original doorway has been filled in with reinforced concrete to conform to the original specifications of the wall, and a concrete shadow wall, 15 inches thick and 6 feet high along the outside of the storage room, limits the radiation levels along the side of the facility to approximately the same values as before. The most intense radiation is directed toward the earth mound at the rear of the facility.

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Figure 3-7. WANEF Test Cell - Isometric View

The structural design of the test cell is such that the roof would be the second portion of the structure to fail, after the rear truck door, and would require an internal pressure in excess of 110 psf for failure.

The floor structure of the test cell is constructed of I-beams which are covered with metal checker plate. In order to permit trucks to enter the building, the north half of the floor has heavier construction, consisting of type 14W43 I-beams located 1 foot - 11-1/2 inches on centers and covered with 1 inch thick checker plate. Allowable truck loading is 20 tons per axle. The south half of the floor is constructed more lightly to support 150 psf live load plus the weight of the turning fixture when it is required. When the turning fixture is not present, the southeast corner of the test cell will be used, along with the remainder of the east side, for the storage of materials and equipment associated with reactor experiments subject to the limitations on the amount of liquids and combustible materials set forth in Section 7.4. This half of the floor is covered with 3/8 inch thick metal checker plate.

3.2.2.2 Test Cell Heating System

Installation of the FCX test stand necessitated removal of a section of the heating duct in the southwest corner of the test cell. Since this change was required, the then-existing heating system, which was connected to the WREC central heating system, was removed and replaced with a natural gas space heater mounted on the south wall of the test cell approximately 4 feet from the southeast corner about 16 feet from the floor. Exhaust gases are vented to the outside through a motor-controlled damper designed to withstand 20 psi over-pressure. A limit switch on the damper indicating the fully open position must be actuated before the fan and heater will operate. A gamma radiation monitor closes the furnace vent and shuts off the fan and heater. A schematic diagram of the heating system is shown in Figure 3-8 and the physical location of the heater is indicated in Figure 3-5.

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Figure 3-8. Heating and Ventilating System Schematic of Test Cell

3.2.2.3 Exclusion of Water

Since water or other liquids in the test cell constitute a potential hazard when reactors are present, the use of liquids in the cell is governed by the controls set forth in Paragraph 7.4.2. There are no water lines in the test cell area and a sump pump, located in the northeast corner of the test cell basement, removes any water that may seep through basement walls. This water is discharged through a vertical pipe to the outside and a below-grade drain. The discharge line contains a check valve to guard against back up. A water level alarm, discussed in more detail in Paragraph 3.5.2, is activated if the pump should fail and water accumulates.

3.2.3 Labyrinth and Storage Rooms

A concrete labyrinth is constructed at the entrance to the test cell for protection against radiation streaming during reactor operation. The door, located at the entrance to the labyrinth, remains locked during and immediately following reactor operation, when radiation levels in the area are above normal tolerances. An interlock on this door is connected into the reactor scram system (refer to Section 6.3).

A storage room is located at the end of the labyrinth, as shown in Figure 3-4.

Personnel access to the storage room from the labyrinth is through a "B" label hollow metal fire door which can be locked on the storage area side. A 10 foot metal roll-up door, located in the east wall of the storage area, opens on the rear area. During reactor operations, the truck door is locked from the inside. Electrical interlocks on both the truck door and personnel door are a part of the reactor control system (see Sections 6.3 and 6.7).

3.2.4 Vault and Fuel Assembly Room

3.2.4.1 General

A vault, of solid block construction covered with a poured concrete roof and equipped with a merchandise type vault door, is used for the storage of fuel. Racks are provided in which the fuel is stored in special containers such that the

array will be safe if flooded. The load safety factor for the racks, when fully loaded, is greater than 10. Water lines do not terminate in, pass through or over the vault. Adjacent to the vault, and connected to it by a small pass-through, is the fuel assembly room which is used for assembling and disassembling fuel clusters and for other experimental operations where airborne contamination is possible. Area contamination is held to a minimum by use of a vacuum table constructed of welded sheet metal with a metal grating top.

3.2.4.2 Ventilation System

The ventilation system in the vault and fuel assembly room is an integral unit serving only these two areas. Its supply and exhaust blowers are balanced to maintain a negative pressure relative to the surrounding areas. Outside air is supplied to the storage vault and fuel assembly room by a 1000 cfm blower through a heating and cooling unit. The main purpose of air circulation in the vault is to prevent moisture condensation, particularly on the overhead metal deck. Stale air is exhausted from the fuel assembly area through the vacuum table by means of a 1000 cfm exhaust blower. The exhaust passes through a coarse filter, then an absolute filter, and is discharged to the atmosphere. Air supplied to the vault is returned through a duct to the assembly room for exhaust. Since the assembly room may be a contaminated area, care is taken to maintain a negative pressure in the room. The system is diagrammed in Figure 3-9.

3-16

Figure 3-9. Fuel Vault and Assembly Room Heating and Ventilating System Schematic

3.3 BUILDING SERVICES

3.3.1 Water and Heat

Water is supplied to the facility by Westmoreland Water Authority through a 3-inch water main. There are two shutoff valves in the supply line, one located outside the building on the west side, the other inside the building. The hot and cold water supply is shown in Figures 3-10 and 3-11. It should be noted there is no water supply in the test cell area, fuel storage vault or fuel assembly area nor are there water lines running through these areas, thus eliminating the possibility of flooding or accidental release of water in these areas.

Heat is supplied to the facility by Modine fin type hot water baseboard heaters, which are fed by a natural gas-fired hot water boiler, and a circulating hot air system, as shown in Figure 3-12.

3.3.2 Power

Electrical power for the WANEF-WREC facilities is supplied by the West Penn Power Company, through a 12 KV transmission line. A step down transformer outside the facility provides 3 phase, 4 wire, 120/208 volt service which enters the facility through two 4-inch conduits to the main power distribution panel D-1. Power is fed from the main distribution panel D-1 to the various load centers throughout the facility as shown in Figure 3-13. The physical locations of the various power and lighting panels which supply the WANEF facility are shown in Figure 3-14. As indicated, all major equipment at WANEF excepting the test cell crane and rollup door receives power from breaker panels located in the WANEF part of the WREC-WANEF facility; even though the power for these two items is supplied from the WREC breaker panel, the actual controls are located in the WANEF facility.

In addition to the 120/208 V service, a 440 volt single phase line supplies power, primarily to a welder. The panel circuit breaker for this service is located in the storage room. There are two power outlets--one in the hall near panel D-4 and one in the storage room.

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Figure 3-10. Facility Water System Schematic

Figure 3-11. Facility Water System Layout

LEGEND

CONCRETE

SUPPLY

RETURN

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Figure 3-12. Heating and Ventilating System Layout

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Figure 3-13. Electrical System Schematic

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Figure 3-14. Location of Power Distribution Panels

3.4 MAJOR SUPPORT EQUIPMENT

3.4.1 PAX Test Stand

The test stand used for critical experiments on the PAX is located in the northwest corner of the test cell. The stand, including pertinent design characteristics, is shown in Figure 3-15. It is constructed of commercial steel items with the load resting on six 3-1/2 inch diameter by 1/4 inch wall tubular legs. Each leg, bearing a maximum load of 3100 pounds, is threaded into a 10 inch diameter leveling pad which is in turn bolted to the test cell floor support structure with four 3/4 inch bolts. A 1/2 inch thick steel plate, welded to the legs 22.75 inches above the floor, is used for mounting the control drum drive system. The plate is given additional mechanical stability by use of 1/2 inch steel gussets welded to the legs and control drive support plate. The total weight of the control drum drive mechanisms is approximately 2500 pounds.

Located 60 inches above the floor is a two-inch thick reactor support plate. It has a 51-inch diameter hole in the center to accommodate the pressure vessel closure. The flange of the closure is bolted to the machined surface of the support plate with six 3/4 - 10 bolts. The support plate is welded to the test stand legs and is reinforced with support blocks and gussets. The load safety factor of the support plate with the complete reactor in place is 20 with a deflection of 0.024 inches.

A work platform located 96 inches above the floor is mounted to the test stand legs in the same fashion as the reactor support plate. It completely surrounds the reactor pressure vessel cylinder and allows easy access to the nozzle end of the core which is located approximately 50 inches above the work platform area. Steps providing access to the working area are provided with guard rails. An approximately 150 degree sector of the work platform is removable for certain experiments.

The overall failure load for the test stand is 378,000 pounds. With a maximum load provided by reactor, test stand, drum drives and 2000 pounds of personnel, a load safety factor of 19 is realized.

Figure 3-15. PAX Reactor and Test Stand

3.4.2 FCX Test Stand

The FCX test stand is located in the southwest corner of the test cell. The stand, which is shown in Figure 3-16, has three different levels. The bottom level, flush with the top of the FCX core, is used as a work platform from which core changes can be accomplished. On this level are also the various neutron detectors and the FCX lateral support mechanism. The second level, which is 7 feet 2 inches above the top of the core, houses the control rod electromagnet armatures and their associated limit switches and locks, together with the control rod stations which provide power to the various control rod drives. The top level, which is 13 feet 5 inches above the top of the core, supports the control rod drive mechanisms.

The test stand is supported by four 6-1/2 inch diameter by 3/4 inch wall tubular legs. Each leg, bearing a maximum load of approximately 4000 pounds, is threaded into a leveling pad which is in turn fastened to the test cell floor supports. The floor structure will support this combined weight of core and stand (approximately 30,000 pounds) with a safety factor of 3.43. The FCX core is not supported by the test stand, but, rather, it is supported by a 1-1/2 inch thick aluminum core plate. This plate is in turn supported by a number of cross members which rest on the test cell floor supports. Since the core is at floor level and exposed, a removable fence is attached to the test stand to limit access of personnel to the core.

3.4.3 Crane

The bridge crane located in the test cell is a ten-ton load lifter "Ne-Suh" type with a bridge span of 36 feet 5 inches and travel speeds of 15 and 5 feet/minute. Trolley travel is over the entire length of the bridge, thus allowing the crane access to any portion of the test cell. Trolley speeds are 25 and 5 feet/minute. The vertical lift distance of the crane is 70 feet with two hoist speeds, 15 and 5 feet/minute. The crane is operated by a cable pendant pushbutton station which provides FAST and SLOW speeds for each of the three motions plus RAISE-LOWER for the hoist and FORWARD-REVERSE for the bridge and trolley motions.

Figure 3-16. FCX Reactor and Test Stand

3.5 AUXILIARY INSTRUMENTATION

3.5.1 Radiation Monitors

3.5.1.1 Criticality Alarm System

Located throughout the facility, as shown in Figure 3-17, are a number of gamma radiation detectors which provide an immediate audible alarm in the event of high radiation levels resulting from accidental nuclear criticality resulting from the use, handling, or storage of special nuclear material other than reactor operation.

Each monitor, as shown in the block diagram of Figure 3-18 consists of a gamma sensing ion chamber detector, a local indication and alarm, and an indication and alarm in the control room. The system is provided with a standby emergency power supply which is switched in when line voltage decreases to approximately 70 percent of its normal value.

Each detector is equipped with a Sr^{90} calibrating source (to enable remote calibration of the detectors) and a variable alarm set-point. The ranges of the various channels and their set-points are listed below. The criticality alarm monitors comply with 10 CFR-70.24.

<u>Location</u>	<u>Range R/hr</u>	<u>Typical Alarm Set-Points R/hr</u>
Reactor Gamma	0.01 to 10,000	0.10 [*]
Test Cell Fuel Storage Rack	0.001 to 1.0	0.02 [*]
Counting Room	0.001 to 1.0	0.02
Fuel Assembly Room	0.001 to 1.0	0.02
Fuel Storage Room	0.001 to 1.0	0.02

* Because of the radiation levels that occur in the test cell during operation of the WANEF reactors, the audible alarm will be by-passed during such operations.

Figure 3-17. Location of Radiation Monitors

Figure 3-18. Radiation Monitors Block Diagram

3.5.1.2 Portable Detectors

In addition to the stationary gamma radiation monitoring system, the facility has portable instruments for measuring neutron, gamma, alpha, and beta radiation levels, and a continuous air monitoring system for sampling air in the test cell. Additional instruments are available at the Waltz Mill Site Health Physics Office.

3.5.1.3 Radiation Dosimeters

To provide for measurements of radiation dosages in the event of a nuclear excursion, neutron and gamma dosimeters have been located at suitable points as indicated in Figure 3-17. These dosimeters are capable of measuring neutron dose and a five energy group neutron spectrum distribution which gives the single collision theory RBE neutron dose with an accuracy of $\pm 15\%$. Gamma dose measurement in the range of 10 - 10,000 rads is included.

3.5.2 Test Cell Basement High Water Level Detector

To detect water seepage into the test cell basement, an automatic water level detector has been installed six inches above the basement floor level (twenty-two feet below ground level). If water should accumulate in the basement, this unit activates alarms in the control room and the guard house.

3.6 ALARM SYSTEMS

3.6.1 WANEF Building Alarm System

The WANEF building emergency alarm system warns all personnel in the WREC-WANEF facility of the existence of a condition in either facility that may be hazardous to personnel and/or equipment. It can be manually actuated from the WANEF control room, test cell or the building entrance, as shown in Figure 3-19, or from the WREC control room. Action following the sounding of the building alarm is outlined in the Emergency Procedures section of the Operations Manual. The alarm horns produce a tone similar to that of a standard automobile horn. Power to the alarm system is provided by a standard power supply capable of supplying 12 VDC to each horn.

3.6.2 General Site Alarm

The general site alarm is the ALERT signal (a high-pitched rapid warbler sound) which can be actuated at WREC, WANEF and the Health Physics office in the WAPD Facilities Building. All emergency conditions requiring general site cognizance will be announced over the public address system in conjunction with the ALERT signal. The ALL CLEAR signal is a steady tone over the public address system. The Emergency Alarm speakers for the WANEF facility are also indicated in Figure 3-19.

3.6.3 Guard House Alarm System

To provide protection and alarm monitoring after the facility is secured at the end of the working day, a remote alarm panel is located in the security guard house. This panel annunciates on (a) loss of electric power to the facility, (b) presence of water in the basement of the test cell, or (c) opening of the personnel test cell door.

Figure 3-19. Locations of Emergency Alarms

3.7 WANEF FIRE PROTECTION

3.7.1 Fire Protection

Carbon dioxide extinguishers are available at strategic locations for fire control purposes as shown in Figure 3-20. Due to the hazards arising from the use of water on or around NERVA fuel and reactors, no fire hoses are available in the facility. To ensure that proper fire fighting methods are employed in the event of a fire at WANEF, an organized fire brigade, adequately trained and equipped, is available. This fire brigade is comprised of Waltz Mill Site personnel including two from WANEF, and, through frequent fire drills and instruction in the methods and equipment to be used in combatting fire involving critical assemblies, fuel elements, and other radioactive material, maintains full preparedness.

3.7.2 Fire Loading* of Test Cell

a) Combustible Material

The fire loading of combustible material in the test cell area will be maintained at the lowest possible level. Only those combustible materials necessary for the operation of the reactors or related activity (refer to Paragraph 7.4.2) will be maintained in the test cell.

b) Liquids

The fire loading of liquid in the test cell area will be maintained at the lowest possible level. Only those fluid liquids necessary for the operation of the reactors or related activity will be permitted in the test cell. Furthermore restrictions on the presence of liquids in the test cell when a reactor is present are given in Paragraph 7.4.2.

* Defined as the potential number of BTU per square foot.

Figure 3-20. Location of Fire Extinguishers

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CHAPTER 4

PAX REACTOR DESCRIPTION

4.1 GENERAL

The Permanent Assembly Experiment (PAX) critical assembly, which went critical in February 1964, is a nuclear and mechanical mockup of the NRX reactor which is being developed as the propulsion unit for the NERVA engine. PAX originally consisted primarily of rejected or borrowed NRX components, although subsequent modifications and/or substitutions have been made to simplify core changes and to permit a nuclear mockup of new design concepts. As illustrated in Figures 4-1 and 4-2, the basic PAX assembly consists of the reactor subassembly, the radiation shield mockup, and the pressure vessel.

The PAX reactor subassembly (Figure 4-2) includes the core assembly, graphite inner reflector, beryllium outer reflector, and the lateral support. Clusters of graphite-uranium fuel elements, which constitute the core assembly, rest upon an aluminum core support plate which, in turn, is supported by the pressure vessel dome via a transition ring and the outer reflector support ring. Immediately surrounding the core assembly are three band and pulley arrangements which provide lateral support for the PAX core. A mockup of the actual NRX lateral support springs is provided. The beryllium outer reflector consists of twelve 30° sectors which are held vertically between the reflector support ring and the lower (forward) support ring by means of through-bolts; each sector contains a rotating control drum. As seen in Figure 4-2, one sector of the outer reflector (with drum) and one corresponding sector of the graphite reflector are removable. An aluminum shield (Figure 4-1) bolted to the core support plate serves as a mockup of the radiation shield that will be utilized in the actual NERVA engine.

Surrounding the reactor subassembly and shield is an aluminum pressure vessel which is comprised of three sections: the dome (Figure 4-1) which contains the shield assembly, the cylinder body which contains the reactor subassembly, and (Figure 4-2) a removable section at the nozzle end of the pressure vessel. The nozzle end mechanically supports either the protective cover, or the dummy nozzle, or the components of end-on experiments.

Figure 4-1. PAX Reactor Assembly

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Figure 4-2. PAX Reactor - Cutaway View

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Figure 4-3. Standard and Partial NERVA Fuel Elements

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The test stands and associated hardware for the PAX assembly are described in paragraph 3.4.1. The PAX auxiliary shutdown system, installed in 1966, is described in Paragraph 4.2.7 while the nuclear characteristics of PAX are given in Section 4.5.

4.2 BASIC PAX ASSEMBLY

Portions of the current NRX reactor design have been modified to provide experimental and operational flexibility in the PAX critical assembly which are not possible in the actual NRX reactor. Present (Summer, 1966) modifications (Figure 4-2) include the use of non-interlocking support blocks, modified cluster tie rods, a special lateral support mechanism, and various changes to the inner and outer reflectors. Future NRX engine reactor designs will be mocked up in PAX by making appropriate modifications to the PAX or using components or fuel more representative of the newer designs (see Section 4.3).

4.2.1 PAX Fuel Elements

The PAX fuel elements are a nuclear equivalent of the current NRX design and are mechanically identical to those used in current NRX reactors. Normally, elements rejected in manufacture are used in the PAX, although reactor grade fuel will also be used from time to time, such as in the performance of Fuel Reactivity Check (FRC) Experiments. This exchange can be accomplished directly without modification to the elements.

4.2.1.1 NERVA Fuel Elements

The present standard fuel element for the NERVA engine, shown in Figure 4-3, is a 52 inch long hexagonal rod, nominally 0.753 inches across the flats, with 19 full length coolant channels arranged in a modified triangular pitch. Each coolant channel, reamed to a diameter of 98 mils, is coated internally with 1.5 mils of niobium carbide as protection against the corrosive effect of the high temperature hydrogen coolant. The external surfaces of all elements are coated at the exhaust end for additional protection. Partial elements used at the periphery of the core, as well as the standard element, are shown in Figure 4-3.

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The fuel material is a dispersion of highly enriched (> 93 percent U-235) uranium dicarbide particles, nominally 100 microns in diameter, coated with 25 microns of pyrolytic graphite, in a graphite matrix. The zoned fuel loading varies from approximately 413 mg U-235/cm³ for the material in the central region to approximately 134 mg U-235/cm³ at the periphery of the core (refer to Paragraph 4.2.3). Fuel elements are clearly identified with a seven-digit serial number engraved on the surface near one end.

The environmental conditions in a NERVA engine reactor impose severe requirements on the physical and mechanical properties of core materials. A graphite-base fuel material was chosen for its resistance to thermal shock, high temperature capability and resistance to irradiation damage.

The NERVA fuel element is very resistant to combustion effects and oxidation is not self-sustaining in air below 2000°C. Although carbides can enter into chemical reactions with a large number of materials, the hydrolysis reaction with water is the only one of practical importance in the PAX. This reaction, which is accompanied by expansion, gas evolution and porosity of the fuel matrix, has essentially been eliminated in the NERVA fuel elements by the pyrolytic graphite coating around the UC₂ particles.

4.2.1.2 Other Fuel Elements

LASL NRX type KIWI fuel elements, which are described in Section 5.2, may be used in PAX. As shown in Paragraph 8.3.2, this type of fuel does not release fission products during the maximum credible accident.

Other fuel elements anticipated for use in PAX are FCX fuel elements, which are described in Section 5.2, and fuel which is more representative of future NERVA reactors, such as higher void fraction fuel. Fuel elements could have 115 mil diameter coolant channels or larger and also have molybdenum as an additional coating. The use of such fuel in the PAX experimental program will be reviewed by the Nuclear Safety Committee to ensure that it does not represent an unevaluated safety question.

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4.2.2 PAX Fuel Cluster

4.2.2.1 Present Design

Figure 4-4 illustrates the present PAX cluster assembly showing the non-interlocking support blocks as well as the cluster tie rod cup to be used for cluster handling. Six fueled elements (1) are attached to a central unfueled graphite element (2) by means of a cluster support plate (not shown) on the dome (lower) end and a 2.0 inch thick non-interlocking graphite support block (10) on the exhaust (nozzle) end. A tie rod (8) (also see Figure 4-5a) extends through the unfueled graphite support block, central unfueled element, and cluster support plate. The six surrounding fueled elements (1) are held together by means of bushings pressed into the cluster support plate. Each bushing carries an orifice which is inserted into the central coolant channel of the fueled element. The graphite support block grips the nozzle end of the fueled elements by friction imparted by the clamped tie rod through the molybdenum cone (7) which seats into the support block. A small metal cluster cup (9) is attached under the tie rod head on the exhaust (upper) end. A hooked tool may be inserted into the hole in the cup wall, thus allowing individual cluster removal from the critical experiment assembly. The nominal width of a fuel cluster (across the flats of the hexes) is 2.25 inches. The overall length, including support blocks and cluster plate, is 54.3 inches. The tie rod extends down an additional 3.0 inches to allow insertion into the aluminum core support plate, by which all clusters are supported. Alignment flats on the tie rod holder, as well as threads on the tie rod holder and fuel cluster nut, are machined off from the standard NERVA units to allow easier PAX cluster insertion and removal.

4.2.2.2 Future Designs

Future cluster designs being considered for use in the PAX assembly include the use of (a) a two-pass tie rod and (b) a separable tie rod.

The two-pass tie rod is in reality a tube that allows coolant to pass around the tube during the inlet phase of flow and to pass through the tube during coolant exhaust.

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Figure 4-4. PAX Fuel Cluster, Exhaust End

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(a) STANDARD CLUSTER ASSEMBLY

(b) CLUSTER ASSEMBLY WITH SEPARABLE TIE ROD

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Figure 4-5. PAX Cluster Tie Rod Assemblies

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Future designs also include reduction in thickness or elimination of the graphite support block.

The separable tie rod, shown in Figure 4-5b, was designed to provide more flexibility in PAX experiments. As compared to the standard tie rod, the major features of the separable tie rod are the replacement of the tie rod holder and acorn nut with a simplified one piece design as shown in Figure 4-5b. The tie rod spring can also be replaced with three cylinders which mock up the spring length to provide correct cluster length, but do not apply the loading characteristics that the spring did. The tie rod cup has been brazed to the end of the tie rod to enable removal of the threaded upper portion of the tie rod. The upper removable portion of the tie rod is approximately the same length as the support block so that when the top of the tie rod and the support block are removed there is no part of the tie rod protruding above the top of the fuel elements.

In using this separable tie rod design there is no possibility that cluster elements can inadvertently move when the tie rod ends and support blocks are removed, since the core is held in the lateral direction by the core lateral support. This feature reduces uncertainties in certain experimental measurements, since only those items of interest are moved and all other core components are undisturbed. The separable tie rods and removable support blocks also permit mocking up the inlet area of the core at the nozzle end, which is the only end readily accessible for experiments.

4.2.3 PAX Core Assembly

The core assembly normally consists of 241 fuel clusters which are supported by the core support plate via the tie rods as shown in Figure 4-6. The present support plate is a perforated aluminum structure with a diameter of 38.5 inches. It varies in thickness from 4.12 inches at the edge to 6.0 inches at the center. In addition to coolant holes, the nozzle-end (upper) face of the plate is bored to accept the core inlet ends of the cluster assemblies while the dome-end (lower) face is counterbored to accept tie-rod nuts and locking devices. A cylindrical periphery for the core is achieved by adding unfueled graphite filler strips to the peripheral

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Figure 4-6. PAX Core Assembly

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Figure 4-7. PAX Core Loading Pattern

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clusters. The present core has an overall diameter of 35.5 inches, an active core length of 52 inches, and a total assembled weight of approximately 2750 pounds.

A plan view of a symmetrical sector of the present core, illustrating cluster orientation, filler piece location, and fuel loading regions is presented in Figure 4-7. In order to reduce power peaking at the outer boundary of the core, fuel loading is varied as a function of radial position. The composition of a typical core is described in Table 4-1.

4.2.4 PAX Inner Reflector and Lateral Support

From a nuclear standpoint the PAX inner reflector and lateral support system is similar to that of the NRX reactor. However, several mechanical differences exist which allow the necessary flexibility for performance of critical tests.

4.2.4.1 Inner Reflector

The graphite inner reflector was designed to allow experimental flexibility not possible in the NRX assembly. The reflector is 54.0 inches long with an inside diameter of approximately 36 inches and an outside diameter of approximately 40 inches. In the PAX assembly the lateral support springs, used for core bundling in NRX reactors, are mocked up. Core bundling in PAX is accomplished as described in Paragraph 4.2.4.2. When the core is bundled, a gap of approximately 0.2 inches exists between the filler strips and the inner reflector.

Another reflector is available which has a removable 40° segment, as shown in Figure 4-2. This sector is normally held in place in the reflector by means of a tongue and groove arrangement which mates with the remaining 320° portion of the inner reflector. Two 5° segments are also available which may be substituted into the 40° area, giving a net 30° sector gap. This feature, which permits 30° sector mockups of new inner reflector designs, was successfully used for experiments in support of NRX-A6 in early 1966.

4.2.4.2 Lateral Support

To permit convenient banding and unbanding of the core, a special lateral support system was designed and fabricated for the PAX. Three circumferential stainless

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TABLE 4-1
TYPICAL PAX CORE INVENTORY

Region	Material	PAX Mass (kg)
Core	Carbon	
	Fuel Elements	854.97
	Unfueled Elements	163.599
	Pyrographite	26.152
	Negative Shims	2.147
	Positive Shims	5.346
		<u>1,052.214</u>
	Uranium	173.82
	Niobium	67.71
	Tantalum	6.072
Filler	Stainless Steel	27.349
	Hydrogen (Absorbed)	0.0158
Strip Region	Carbon	86.886
	Stainless Steel	9.736
Lateral	Carbon	293.913
Supports	Aluminum	25.86
	Steel	11.05
	SiO ₂	10.83
Beryllium	Beryllium	711.10
Reflector	Aluminum	1.71
	Titanium	27.14
Control	Beryllium	209.63
Drums	Aluminum	18.80
	Boron-Aluminum Vanes	5.10
Cluster		
Plate	Gadolinium-Stainless Steel	8.30
	Stainless Steel	20.43
	Aluminum	2.49

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steel bands are placed under tension by pulley and cable system operable from the top of the reactor. Figure 4-8 shows one unit of the mechanism. Each band is 1-inch wide by 0.032-inches thick; one is located near the top, one at the middle, and one near the bottom of the core. One end of each band is secured to a steel pulley frame which, in turn, is attached to the inner reflector. The other end of each band is attached to a 3/32-inch diameter, multistrand stainless steel cable, which passes around the pulley and up a groove in the graphite reflector to the top of the reactor. Swaged on the end of each cable is a threaded bolt that protrudes through a support bracket which is attached to the beryllium (outer) reflector. A spring, washer, and nut are attached to the threaded bolt, thus allowing the bands to be tightened and supplying the lateral support necessary to give a tightly packed core. The system provides a range of core bundling from 0 pounds to that present in the NRX reactor. A calibrated spring on the support bracket indicates the tensions on the steel bands.

Together with the interlocking nature of the cluster design, the lateral support system permits operation of the reactor with some fuel clusters removed, as long as the remaining clusters in the core are physically interlocked with one another.

4.2.5 Outer Reflector

The present outer reflector assembly, shown in Figure 4-2, consists of twelve 30° beryllium sectors which are clamped vertically between two support rings by thirty-six 5/8-inch titanium tie bolts. The assembly is 52 inches long and is approximately 4.6 inches thick. The standard NRX reflector is modified to permit removal of any 30° sector and its drum without disassembling the core or the remaining eleven sectors. This feature, together with the removal of one mating inner reflector sector, permits 30° sector mockups of new reflector designs. One control drum assembly has been modified to permit its removal from the sector without removing the sector from the reflector. This permits the installation of new control drum designs involving new cylinders and absorber material. Operation with only eleven sectors and drums is also now possible, making it feasible to inject neutrons directly into the core from an external source such as a neutron generator.

Each reflector sector contains a 4.124-inch diameter vertical hole at its center for a reactor control drum. A nominal 0.050-inch clearance exists between drum and reflector sector.

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Figure 4-8. PAX Core Lateral Support Device

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Laboratory

WANL-TNR-099
Revision A

Each control drum is supported by two ball bearings which are located in bearing housings on the support rings.

In addition to coolant holes, sectors may contain holes for insertion of aluminum or beryllium rods or other materials for reactivity shimming.

4.2.6 Control Drum

The present control element for the PAX reactor, shown in Figure 4-9, is a 52-inch long beryllium cylinder to which a 120° span by 0.220-inch thick poison plate is held by an aluminum sleeve. Each end of the sleeve is welded to an aluminum flange which has an extension that serves as a bearing shaft.

Reactivity control is achieved by rotating the poison section of the control drum relative to the core. The maximum poison effect is attained when the poison plate is closest to the core and located symmetrically between the reflector and core. This is its "full-in" position. Control drum position is regulated by individual external motor-driven mechanisms via drive shafts which penetrate the pressure vessel dome. Connection between the drive shaft and drum coupling is by means of a spline; no axial locking scheme is provided with this spline so that support of the shaft by the drive mechanism is relied upon to maintain this coupling. A mechanical stop is provided to prevent the drum from rotating beyond either its full-in or its 180° full-out position. A locking pin can be inserted through this stop into the bearing housing at the full-in position to prevent any drum rotation when such protection is desired. A torsion spring (which is separate from the scram spring), having a torque of approximately 16-24 inch-pounds keeps the drum in the full-in position when the drive shaft is not connected.

The present poison plate is a dispersion of elemental boron (fully enriched in the B^{10} isotope) in an aluminum matrix having a nominal composition of Al - 20 w/o B-10. It is fabricated by a powder metallurgy process using hot pressing and hot rolling. The 0.070-inch thick matrix is metallurgically bonded between 0.075-inch thick aluminum. Rivets attached to the plate center it in the space between the drum housing and the beryllium cylinder and provide two annuli, each approximately $1/8$ -inch thick.

[REDACTED]

Figure 4-9. PAX Control Drum

4.2.7 PAX Auxiliary Shutdown System (PASS)

4.2.7.1 Introduction

An auxiliary shutdown system utilizing boron carbide has been designed and constructed for use on the PAX reactor. The mechanism is designed to provide a "last ditch" reactor shutdown capability by inserting nuclear poison worth at least 1\$ in reactivity, thereby cancelling the positive reactivity insertion associated with the maximum credible accident in the PAX (i.e., a one-drum rollout accident). It is not intended that this system be interlocked to provide an additional automatic scram; rather, it will be manually actuated from the control room if the need exists. The time necessary to insert the auxiliary poison will be of the order of a minute, in contrast to the automatic shutdown system (control drums) which takes less than one second.

4.2.7.2 Description of the Shutdown System

The PAX Auxiliary Shutdown System shown in Figure 4-10 consists of two complete and independent units, each of which consists of a poison storage hopper, a valve-operating gearmotor, a dump valve, a flexible stainless steel discharge tube, and a discharge nozzle. Both units are supported by a welded support structure so that they are approximately 14 feet above and laterally displaced about 4 feet from the PAX.

The support structure, which is fabricated from aluminum pipe, is bolted to steel brackets which are, in turn, bolted to the test cell west wall ledge above the PAX. The structure has a safety factor of 4.8 with a 200 lb man standing on the first rung down from the top. The two poison storage hoppers, each of which contain about 20 lb of poison granules, are bolted to the ends of the support structure. The dump valve was designed to open in the direction of poison flow so that there would be no binding. It consists of a tapered plug held up against the hopper bottom opening by a rod which passes up through the center of the hopper and is attached through a linkage to the gearmotor. When the system is actuated the gearmotor drives the rod and plug down away from the hopper bottom, permitting the poison granules to flow out of the hopper. Limit switches

Figure 4-10. PAX Auxiliary Shutdown System

control the linear travel of the plug.

The upper end of the stainless steel discharge tube has a funnel permanently attached to it and this funnel is bolted to the tapered bottom of the poison storage hopper. The other end of the flexible discharge tube attaches, by means of a fitting, to a nozzle which fits into the reactor annulus between the inner and outer reflectors and bolts to the nozzle end support ring. The poison material is thus discharged into the annulus between the outer and inner reflectors.

The 12-volt DC gearmotors are supplied from a single storage battery and both are actuated by means of a dump switch on the console. The switch is conveniently located on the console for easy access by the operator or other control room personnel. A switch cover prevents inadvertent dumping of the poison material but is easily removed to permit operating the switch when required. A pilot light, also mounted on the console, lights when the battery voltage drops below a preset value such that the gearmotor would not function properly. Both the dump switch and pilot light are appropriately labeled.

The poison material used is natural boron carbide of a -14+20 mesh size. These B_4C granules measure between 1/32 and 1/16 inch in diameter and have excellent flow characteristics.

4.2.7.3 Tests

A full-height model of a 120° section of the annular gap between the inner and outer reflectors (complete with mockup springs) was constructed with viewing ports in the outer wall to permit visual observation of the poison flow and distribution. When tests were conducted using 5 lb of B_4C granules in a prototype poison storage hopper, about one-half minute was required for all of the material to run into the model after being released. The B_4C granules flowed smoothly like dry sand around the mockup springs, building up into a pyramidal distribution in the annular gap. From the observed distribution of the granules it is estimated that 12 lb of B_4C from one hopper would form a similar pyramid whose base surrounds one-half of the core and which extends

vertically to the discharge nozzle. That is, it would fill one-fourth of the annular gap volume. Subsequent tests, conducted after system installation, in which 30 lb of B_4C were released from each hopper and discharged into buckets, required 40 seconds for complete discharge.

Reactivity worth measurements on a small sample of the B_4C placed in the reflector region of PAX qualitatively agree with calculations verifying that the material is, indeed, natural boron carbide.

4.2.7.4 Analytical Determination of Shutdown Reactivity

Analyses have been made to determine the shutdown available under several different situations. These results indicate that one hopper, dumping approximately 12 lb of B_4C into the reflector gap, would provide approximately 3.70\$ worth of negative reactivity. The distribution of B_4C pellets in the reflector gap is such that at the base of the reflector, B_4C would enclose 180° of the reactor and assume a pyramidal effect with a peak at the reflector top. Two hoppers dumping approximately 24 lbs of B_4C into the reflector gap on opposite sides would provide approximately 7.40\$ worth of negative reactivity. Thus, there is no reactivity interaction between the two pyramids of poison. The negative reactivity inserted, due to a full height rectangular segment of B_4C between the inner and outer reflectors with a width equivalent to one 30° sector, is approximately 1.25\$.

4.2.7.5 Safety Analysis

Modes of possible failure of the system and the resulting consequences are identified below, together with an evaluation of the probability of such failures.

1) System Failures

The failure of house power does not affect the operability of the system inasmuch as it is supplied by a 12V storage battery. This supply is monitored at the console by a circuit which visually annunciates before battery voltage falls below the level required to operate the motors. It is considered incredible that the

system would fail to operate when required because of a power failure.

The poison material chosen is not hygroscopic and is among the most chemically inert substances so that it will not lump or cake in the storage hopper even when undisturbed for long periods of time. Periodic tests will be made to verify this. Failure of the material to flow when the dump valve is opened is therefore considered to be incredible.

If the lower limit switch on a motor fails, either electrically or mechanically, its dump valve cannot cycle back to the closed position because of the physical arrangement of the mechanical linkage. The torque capacity of the motor will cause it to stall before the pins in the mechanical linkage will shear. Over-travel of the crank arm on the gearmotor is limited by the limit switch bodies and support brackets. The tapered plug which forms the dump valve might conceivably block the funnel opening if it came off the vertical rod. However, it is fastened with a jam nut against a pre-loaded spring. It is considered incredible, therefore, that the mechanical operation of the valve can fail in a way that will prevent the material from flowing down the discharge tubes.

2) Reactor-Associated Failures

The system will fail to shut down the reactor in the event that the poison material cannot enter the reactor. Such a situation might develop if, under the condition of a reactor excursion, thermal expansion of the core in the radial direction would reduce the annulus in the reflector region to such an extent that the poison would be blocked. A calculation was made of the radial expansion for a hypothetical accident of 200 MW-sec, using the mean coefficient of thermal expansivity in the transverse direction in a current NERVA fuel specification* and the average temperature rise in PAX caused by the accident. The annulus remaining

* PDS 30093 dated 15 December 1965 (CRD)

will be 0.165 inches wide, considerably larger than the size of the granules. Therefore, even under the conditions of this hypothetical accident, the system will function.

If the B_4C material should pass on through the reactor and leak out the bottom, shutdown might not be achieved. The annular space into which the poison goes, however, has no openings where leaks can occur. The reflector support, starting with the PAX-D build, has been designed with this in mind. Full length seal strips have been provided to use when the removable 30° sector of the PAX-D inner reflector is removed.

Although spillage of the entire amount of B_4C over the top of the core would result in a positive reactivity insertion of about 25% , this would not lead to an accident when the reactor is shut down. During operations at delayed critical, such an insertion would result in a period of about 25 seconds, easily capable of being manually controlled. Such an accident is unlikely, however, because of the design of the connection of the flexible hose at the discharge nozzle and the administrative requirement that the hoses, when not connected during reactor operations, be mechanically retained away from the reactor.

3) Human Failure

The most credible failure of the system is the human failure to connect the hoses after a reactor change. A procedural requirement that the EIC check off on the Pre-Startup Checklist the fact that the hoses are either connected or retained away from the reactor before each reactor operation begins, is an effective administrative control.

4.2.7.6 Use of the PAX Auxiliary Shutdown System (PASS)

Because PAX will have to be disassembled in order to remove the poison material, the system will be used only under emergency conditions when the automatic safety system has failed to shut the reactor down. If such an emergency should exist, the system

will be actuated by a member of the control room staff at the direction of the Experimentalist-in-Charge. Periodic checkouts of the system will be performed by removing the discharge nozzles from the reactor and dumping the B_4C into buckets.

Although both of the PASS units can be connected to the reactor during operation, one unit by itself provides more than enough shutdown capability to cancel the reactivity insertion which leads to the maximum credible accident. Therefore, only one unit is required normally.

During manual reactivity changes or other operational reactor changes when the reactor is shut down, the flexible tubes may be disconnected from the reactor. The tube(s) will be re-connected to the nozzle(s) prior to the startup of PAX and an appropriate entry will be made on the Pre-Startup Checklist.

It is expected that certain experiments will prevent the use of the PASS. In such cases, the approval of AGC and SNPO-C as well as the recommendation for approval by the WANL Nuclear Safety Committee will be obtained prior to performing the experiment.

Detailed procedures concerning the periodicity and detail of system checkouts, entries to the Pre-Startup Checklist, and use in emergency situations are included in the WANEF Reactor Operations Manual.

4.2.8 Poison Wire System

A nuclear poison is required to supplement the regular control system when PAX is disconnected from the control console, and during certain assembly operations to insure that the limitations on the shutdown are not exceeded.

Since only the fuel element coolant channels are available for the insertion of the poison, a poison wire system was chosen in which full length 0.050 inch ID stainless steel tubes filled with natural boron carbide (80 percent theoretical density) are inserted into the fuel coolant channels. Each wire is coated with 0.0055 inches of polyethylene to protect the NbC coating in the coolant channels from excessive abrasion. For handling convenience, the wires are ganged by means of a cluster cup.

4.2.9 Procedures for Assembling the PAX

The WANEF procedure for assembling the PAX is as follows:

- 1) The pressure vessel closure, shield, and core support plate are bolted to the PAX test stand.
- 2) The beryllium outer reflector sectors are then installed with support rings and attached to the core support plate by means of the transition ring.
- 3) Drum drive units are connected to the control drums located in the beryllium reflector, using modified drive shafts; and the drums are checked for maneuverability and scram times.
- 4) The PAX inner graphite reflector, with mocked up NRX support springs and the previously described PAX lateral support system, is installed inside the outer reflector on the test stand. Note: The PAX can be assembled without the inner reflector.
- 5) An aluminum pressure vessel is installed over the reactor assembly.
- 6) Clusters of fuel elements previously poisoned with wires, are loaded into the reactor proceeding from the center outward to the reflector in increments, according to loading zones, using a core assembly fixture as a means of supporting the tops of the clusters during loading. Following the insertion of all clusters, filler strips are added and the core is banded by means of the PAX lateral support system.

Note: During the incremental loading of fuel, standard operating procedures contained in the WANEF Operations Manual are followed.

4.3 MODIFICATIONS TO PAX

In order to satisfy the requirements of the NERVA program, modifications to PAX are required from time to time in order to more closely simulate advanced reactor designs. These include modifications to provide a mockup for NRX-A6 and future reactors and also for engine component tests.

4.3.1 PAX for NRX-A6

The PAX reactor, modified as shown in Figure 4-11, is an exact nuclear simulation of the NRX-A6 beryllium reflector throughout, as far as mass is concerned. In the mockup quadrant, three NRX-A3 type 30° sectors, modified as shown in Figure 4-12a, to simulate NRX-A6 exactly. Throughout the remaining 270 degrees, the reflector consists of 9 KIWI sectors augmented on the inner diameter with additional removable beryllium liners, as shown in Figure 4-12b.

In the 90 degree sector, the fuel loading mocks up the NRX-A6 configuration. The PAX loading (refer to Figure 4-7) in the central fuel zone is essentially the same as that for NRX-A6. However, the heavier Nb content of the A6 elements is simulated by addition of Nb wires in the PAX fuel elements. In the 90 degree sector these elements are replaced by elements made to the NRX-A6 fuel specification. The actual seal, retainer, and load segments are simulated by a cylindrical shell of graphite about 2.4 inches thick. This mockup has simplified the design at negligible loss of nuclear equivalence. The same type of core banding device as described in Section 4.2.4 is used to support the core laterally.

Since the KIWI sectors have adjustable beryllium shims, the effect of varying beryllium mass on reactivity and drum span may be studied experimentally. If small changes to the NRX-A6 reflector design are made, this flexibility will be used to interpolate to the actual beryllium mass. Similarly, a reduction in graphite content of the seals can be handled by drilling holes in the graphite shells if the NRX-A6 seal, retainer, and load segments are reduced in size. The three NRX-A3 type sectors are replaceable by three additional KIWI type sectors.

4.3.2 Future PAX Modifications

It is anticipated that other modifications will be necessary in future PAX reactors in order to facilitate experiments not presently defined. These experiments would include:

- 1) Engine component testing in support of the XE-1 and XE-2 reactors. These would

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Figure 4-11. PAX Assembly for NRX-A6

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

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

Figure 4-12. Sector Design for NRX-A6 PAX Experiment

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include, for example, nuclear mockups of the flight shield, propellant tank, pump, and other engine components.

- 2) Mockup of the ETS/1 compartment shield. A nuclear mockup of this shield, using paraffin in place of water, will permit measurements of the reactivity worth of the shield and of the attenuation of neutron and gamma fluxes through it.
- 3) Operation of the PAX assembly with one sector and/or one drum removed. This would permit such measurements as the determination of the leakage flux spectrum without the beryllium sector in place, insertion of neutrons from the pulse neutron generator into the core rather than through the reflector, and the reactivity worth of a single sector or control drum.

4.4 PROOF TEST EXPERIMENT (PTX)

Prior to shipping an assembled, poisoned, NRX reactor from WANL to NRDS, the reactor may be brought to WANEF for low power testing and checkout. As the nuclear, mechanical, and operational features of the NRX are identical or compatible with the PAX, the NRX can be supported on the PAX test stand and be operated by the PAX control and instrumentation system. Poison wire removal, criticality, and low power operation of the NRX/PTX are identical to that for PAX. For a description of the NRX, reference should be made to the NRX Safety Analysis Report, WANL-TNR-148, and the latest NRX safety documentation.

4.5 NUCLEAR CHARACTERISTICS

The very purpose of the PAX assembly requires that it have essentially the same nuclear characteristics as the NRX reactor. During the early stages of the NERVA program at WANEF, these characteristics could only be inferred from data obtained on the KIWI-B4A reactor from which the basic NRX design had evolved. Additional information now available from the NRX-A2 through NRX-A5 test series, as well as from the PAX experimental program, has provided reliable values for such fundamental parameters as control drum worth, effective delayed

neutron fraction, temperature coefficient, material reactivity coefficients, etc. The internal and external core changes that must be made on PAX in support of NERVA development can therefore be accomplished with predictable reactivity effects.

Typical nuclear parameters of the basic PAX are summarized in Tables 4-2, 4-3, and 4-4, although variations in these values may result from reactor changes required in the course of the program. The reactivity worths of materials external to the reactor assembly are presented in Appendix A.

It should be noted that there is a significant increase in control drum span (approximately 10 percent) and temperature coefficient (approximately 40 percent) over those values used in the previous PAX hazards analyses (refer to Section 8.3). The validity of this new temperature coefficient has been confirmed by the successful predictions of the NRX-A2 and NRX-A3 hot tests, and by the agreement with results of the NRX-A2 temperature coefficient test.* The nuclear safety significance of changes in the magnitude of drum span and temperature coefficient will be discussed in Chapter 8.

The shutdown insertion capability of the PAX Auxiliary Shutdown System is listed in Table 4-2 for various insertion conditions. The required shutdown insertion for this system is given in the WANEF Operating Limits document.

* WANL-TNR-199, Analysis of NRX-A2 Reactor Test Results, April 1965

TABLE 4-2

TYPICAL NUCLEAR CHARACTERISTICS OF PAX

	Approximate Values
** Effective Delayed Neutron Fraction ($\bar{\beta}$)	0.0071
** Prompt Neutron Lifetime (ℓ^*)	24 μ sec
Control Drum Span	8.5\$
Critical Drum Angle (Normal)	90°
Excess Reactivity	4.0\$ for 90° control drum angle
Shutdown Reactivity	4.5\$ for 90° control drum angle
Poison System - Shutdown Reactivity Full Core (6990 Wires)	66\$ dry 32\$ flooded
Central Core (798 Wires)	12.3\$ drums at 0° 3.5\$ drums at 180°
** Temperature Coefficient ($\Delta k/\Delta t$)	- 0.20 $\%$ /°C
Auxiliary Shutdown System - Shutdown Reactivity	
One Dispenser (5.5 kg B ₄ C)	3.7\$
Two Dispensers (11 kg B ₄ C)	7.4\$
One Dispenser into 30° Annular Sector	1.25\$
** Shutdown Coefficient (B)	-9.1 $\times 10^{-10}$ \$/watt-sec

** These parameters are used in accident analysis calculations, paragraph 8.3.3.

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TABLE 4-3
TYPICAL CORE REACTIVITY COEFFICIENTS FOR
52 INCH LENGTH MATERIALS, CENTS/KG

<u>Material</u>	<u>Radial Region</u>	<u>I</u>	<u>II</u>	<u>III</u>	<u>IV</u>	<u>Core Average</u>
	Region Outer Radius, cm	36.366	40.279	41.938	43.804	
Carbon		6.20	4.64	4.15	3.80	5.62
Uranium		14.4	10.26	11.07	12.18	13.33
Niobium		-6.46	-3.67	-3.11	-2.75	-5.48
Stainless Steel		0.81	-0.13	-1.29	-1.66	0.33
Inconel-X		1.31	-0.26	-1.51	-2.30	0.61
Boron Carbide		-73*				

* For an approximate 0.2 inch thickness layer of loosely packed boron carbide

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TABLE 4-4
TYPICAL REFLECTOR REGION REACTIVITY COEFFICIENTS
FOR CORE LENGTH MATERIALS

	<u>Cents/kg</u>
Filler Strip Region	
Carbon	3.39
Stainless Steel	-1.23
Lateral Support Region	
Carbon	2.22
Stainless Steel	-2.3
Aluminum	0.26
Silica	1.2
Beryllium Reflector Region	
Beryllium	2.2
Aluminum	0.2
Titanium	-0.45
Control Drum Region	
Beryllium	2.2
Aluminum	0.26

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CHAPTER 5

FCX REACTOR DESCRIPTION

5.1 GENERAL

The Flexible Critical Experiment reactor is designed to permit "clean" experiments to be performed, unperturbed by the varying quantities of extraneous material present in the NRX and NR reactors, thus mocking up simple models which can be treated analytically. This ability to verify the analytical predictions experimentally will lead to an increased degree of confidence in the computer codes and design methods used in nuclear calculations. Although the fuel used in the FCX is the same as, or similar to, that used in the PAX assembly, the mechanical design is different in order to provide: 1) a reactor with a minimum of internal hardware and 2) the flexibility required for variation of core materials, C/U ratios, reflector materials and thicknesses and reactor size. In particular, the FCX can be operated as a bare (unreflected) core. Control of the reactor is effected by centrally located, two-part control rods, each consisting of a neutron absorbing section and a fueled follower. Manually adjustable control drums or mockups located in the reflector may be employed for cores too large for the PAX assembly.

The FCX was taken critical initially in April 1966 and operated for several weeks with a graphite reflected, NRX fueled core. During this time the following measurements were made: control rod worth, reactivity coefficients of core materials, neutron flux and spectra, and power distributions through the core and reflector. The loading for criticality and the control rod worths were in good agreement with predictions. The reactivity coefficients and neutron flux and spectra measurements at the center of the FCX were similar to those measured at the center of the PAX core attesting to the similarity of the two cores at the center.

5.2 CORE AND REFLECTOR COMPONENTS

5.2.1 Fuel Clusters

The FCX core consists of elements which are held together in clusters, as shown in Figure 5-1, similar to the modules used in the NRX reactors. However, these clusters have been designed so that non-fuel material is kept to a minimum and so that individual

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Figure 5-1. FCX Cluster Assembly

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elements within a cluster can be easily removed for certain types of experiments. The central element of the cluster acts as the tie rod. Each end is counterbored and threaded 3/4 inches to receive a 1 inch long by 0.5 inch diameter aluminum stud which holds the cluster plate in place. The aluminum cluster plates, which have the same external shape as the cluster, have pins positioned in them so as to fit either FCX, NRX or KIWI fuel elements. The lower cluster plate is solid while the upper cluster plate contains 1/2 inch diameter holes permitting access to the central fuel element channels. Strength tests have been performed on these elements with this type of arrangement and the graphite has withstood a longitudinal force of 600 pounds.

5.2.1.1 FCX Fuel

The basic building block of the FCX reactor is a fueled or unfueled graphite hexagon with the same outside dimensions as the NERVA elements. The fuel material consists of a matrix of graphite and highly enriched uranium in the form of UO_2 . These FCX elements are nominally 0.754 inch across the flats and 52 inches long. As shown in Figure 5-1, an axial hole traverses the entire length of the fueled elements. These central holes will be used to accommodate graphite rods when varying the C/U ratios or to accommodate the wires or miniature chambers used in power or flux mapping.

5.2.1.2 NRX Reject Fuel

Reject NERVA fuel may be used as well as the standard FCX fuel. Since both kinds of elements have the same external shape and dimensions, only minor modification to the NRX reject fuel is needed to allow the FCX cluster hardware to be used directly. This modification, shown in Figure 5-1, requires that an NRX element to be used in the center of FCX cluster must first be counterbored and threaded to a depth of approximately 1 inch on each end to accommodate the studs which hold the cluster together. The remaining six elements of the cluster need no modification; they are held in place by the pins protruding from the cluster plate into the element.

5.2.1.3 KIWI Fuel Elements

Spare KIWI fuel elements and reject elements from the B4A, B4B, B5A and B6A cores are available for use in the FCX. The KIWI elements, which are geometrically identical to the NERVA elements, contain enriched uranium in the form of UC_2 . The UC_2 was

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converted from UO_2 during graphitization of the $\text{UO}_2\text{-C}$ matrix. Since they are not protected by a pyrolytic coating as are the NERVA elements, a small amount of the UC_2 has reverted back to UO_2 and U_3O_8 by hydrolysis with atmospheric moisture. A measurement of the hydrogen content in KIWI fuel stored at WANEF for 8 months yields a value of not over 175 ppm.

5.2.2 Reflector Design

Beryllium, graphite, aluminum, and other potential NERVA reflector materials will be employed in the FCX reactor as reflector materials. The physical size, composition, and shape of these components will vary depending on the experimental requirements, cost factors, and the availability of materials. Mechanical position restraints will be provided to locate these components positively. In particular, unfueled graphite elements may be clustered using the cluster plates and studs in the same manner as for fuel elements.

5.3 CORE AND REFLECTOR SUPPORT STRUCTURES

An isometric projection (Figure 3-16) shows the first FCX reactor in the test stand. The fuel and reflector cluster assemblies in the FCX are supported by a 1-1/2 inch thick aluminum core plate. This core plate, which rests on and is bolted to the cross beams of the core tank^{*}, has accurately spaced holes to accommodate the cluster support studs. Although the presently installed core plate is 5 feet wide by 6 feet long, the cross members of the core tank are capable of supporting larger diameter cores.

The top support for the FCX cluster assemblies is provided by a series of hexagonal shaped rings similar to those employed on the PAX reactor during assembly and disassembly operations. The top support structure consists of a center hex-shaped ring to which is welded three rectangular bars. These bars are bolted to the work platform at the first level of the test stand.

Each successive ring, larger than the previous ring, is bolted onto the rectangular bars as the core size increases. Each ring has accurately spaced holes which line up with the top

* The 12 foot deep x 10 foot diameter core tank was used in previous heavy water moderated critical studies. The tank and the cross beams supported cores up to 40 tons in weight.

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studs on the installed fuel clusters. After each cluster is installed in the proper hole in the bottom core support plate, a special tube is inserted through the hole on the hexagonal ring and captures the cluster top support stud. In this manner the cluster is held firmly in position at both the top and bottom. After the core has been assembled, the inner hexagonal rings are removed from the rectangular bars and bolted to the bottom of the next test stand level for storage. Additional lateral constraint can be provided by circumferential ring quadrants which are attached to the test stand and located about mid-core height at the periphery of the core.

5.4 CONTROL ROD AND GUIDE TUBE

The FCX control rod is a "fuel follower" type utilizing a rod approximately twice the length of the core. As shown in Figure 5-2, the top part of the rod is a 2.00 inch diameter stainless steel tube, 0.053 inch wall thickness which encases a cadmium-covered polyethylene rod. The absorber section contains a 0.032 inch thick cadmium sleeve which surrounds a 1.805 inch diameter polyethylene rod. Five screws, through the lap joint of the cadmium sleeve into the polyethylene rod, prevent any relative displacement between the two components. To allow for the longitudinal expansion and contraction of the polyethylene due to normal temperature variations, the polyethylene has been sectioned 3 inches from the top end and a small spring has been inserted into a cutout section of the polyethylene. The gap between the two sections of polyethylene is 0.125 inches at ambient temperature. This gap provides enough capability for a temperature increase of greater than 25 F^o, yet the amount of polyethylene removed from the rod is not significant enough to affect its reactivity worth. The spring, which is mounted in the lower portion of the polyethylene, has been sized to maintain the required compression supplied by the control rod tie rod during normal temperature variations. From purely thermal effects¹, the specific volume of polyethylene increases from 1.09 cm³/gm to 1.25 cm³/gm as the temperature increases from 20°C to 120°C. This represents a 15 percent increase in volume and a 5 percent increase in radius. In addition to the void space within the control element itself, there is a radial clearance of 80 mils between the con-

¹ International Series of Monographs on: Radiation Effects in Materials, Chapter 13, Figure 13.14

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Figure 5-2. FCX Control Rod Assembly

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trol rod and the guide tube. A 5 percent increase in polyethylene radius is equivalent to approximately 50 mils. Since polyethylene at 120°C would be a viscous fluid, expansion into the void region of the hardware above polyethylene cylinder would relieve the stress on the steel sleeve.

This type of rod, commonly termed a "flux trap", allows high-energy neutrons to enter the polyethylene, thermalize and be captured when attempting to return to the core through the cadmium. The bottom portion of the control rod tube contains fuel and graphite elements which permit a cluster array similar to that which is in the core. A rod, 0.312 inch in diameter and 112.06 inches long traverses the axial length of the control rod and holds the internal components together in compression. An aluminum guide tube, 0.108 inch wall thickness, extends the distance from the third level of the test stand where it is anchored, through the core and to the bottom of the control rod support structure beneath the core as shown in Figure 5-3. A shock-absorbing mechanism located in the guide tube structure is designed to withstand a force in excess of 35 G's. Mechanical locks with capture keys will be used to secure the control rods.

An analysis of the possible damage to the polyethylene from the radiation environment shows that radiation damage is not expected. The neutron flux in a NERVA core (PAX) by energy group is:¹

		<u>Group</u>	<u>ϕ (nv/fission/sec)</u>
10.0 Mev	-	0.821 Mev	1.5×10^{-4} (max)
0.821 Mev	-	5.53 Kev	5.0×10^{-4} (max)
5.53 Mev	-	1.86 ev	1.8×10^{-4} (max)
		< 1.86 ev	0.4×10^{-4} (max)
			$\sim 9 \times 10^{-4}$ (max)

$$nv \text{ (total)} = \frac{9 \times 10^{-4} \text{ nv/fissions/sec}}{3.1 \times 10^{-11} \text{ watts/fissions/sec}} = 3 \times 10^7 \text{ nv/watt (max)}$$

$$\text{For 2 KW operation, } \bar{\phi} \text{ (max)} = 6 \times 10^{10} \text{ nv}$$

¹ WANL-TNR-128, Reactor Analysis of NRX-A, Volume II, Figure 5-7 to 5-10.

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FLEXIBLE CRITICAL EXPERIMENT FCX CONTROL RODS

Figure 5-3. FCX Control Rod Support Structure

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For one year of continuous (24 hour) operation,

$$\text{nvt (max)} = 6 \times 10^{10} \text{ nv} \times 3.2 \times 10^7 \text{ sec} = 2 \times 10^{18} \text{ nvt}$$

The change^{1, 2} in the specific volume of polyethylene that has been subjected to a radiation dose of 2×10^{18} nvt at room temperature (20°C) is 1 percent. (The specific volume increases from 1.09 cm³/gm to 1.10 cm³/gm). The total radial clearance between the tie rod and the steel sleeve in the control element is 0.020 inches. This is more than adequate to accept the expansion of polyethylene due to irradiation effects (0.003 inches), based on one year of continuous operation. However, in actual practice, the rods will be almost completely withdrawn from the core during reactor operation and should accumulate a much smaller radiation dose. During a postulated accidental excursion leading to an energy release as great as 3×10^9 watt-sec, the integrated neutron flux is 1×10^{17} nvt - a low value as far as radiation damage is concerned.

Since there is no information available on the dimensional stability of polyethylene in a fast neutron-gamma environment, the above analysis utilized the results of radiation damage experiments in a thermal reactor. This evaluation of the irradiation behavior of FCX control rods may therefore not be completely valid. In order to ensure reliability of the reactor control system, a periodic inspection will be performed to assess the overall mechanical integrity of the control rods including dimensional changes of the polyethylene. Routine rod worth calibrations will provide a very sensitive indication of any chemical changes in the polyethylene, particularly carbon/hydrogen ratio.

5.5 FUTURE MODIFICATIONS

The FCX will be modified to achieve a nuclear mockup of the NR-1 reactor so that standard reactor experiments can be carried out during the NR-1 design stage. The mockup design, which is based upon the NR-1 conceptual design report³, preserves the central region

¹ Atomic Radiation and Polymers, A. Charlesby, Pergamon Press - 1960

² International Series of Monographs on: Radiation Effects in Materials, Chapter 13, figure 13.15

³ WANL-TME-1315, NERVA II Reactor Conceptual Design Report

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Figure 5-4. FCX NR-1 Nuclear Mockup

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of the FCX containing the control rods and adds a beryllium reflector, as shown in Figures 5-4 and 5-5. This reflector will be made up of sectors, each of which consists of 3 pieces. The reflector contains mockup control drums with plates of poison material, e.g., a dispersion of B_4C in a copper matrix. Other poison material may also be used. These mockup drums can be manually rotated to various fixed positions and locked in place in accordance with paragraph 7.4.5.4 for manual reactivity changes to the shutdown FCX. As with the present FCX configuration, reactor operation is accomplished with the control rods as shown in Figure 5-3.

Some slight modification to the FCX core support plate and the first work platform will be required because of the larger size of the NR-1. Nevertheless, the fueled region of the reactor will be assembled from clusters held in place in the usual way as shown in Figure 5-4. The fuel clusters will be assembled from reject NRX and, if necessary, KIWI fuel elements together with unfueled elements using the standard FCX aluminum cluster plates and studs. The number and type of fueled and unfueled elements will be arranged to simulate as closely as possible the currently proposed NR-1 fuel loading, zoning and void fraction. Because the FCX cluster assembly contains no internal hardware, counterflow tie tubes will be mocked up in the central element.

The graphite lateral support region of the NR-1 will be mocked up in the FCX by utilizing graphite liner plates inside the beryllium reflector and graphite filler segments to fill the irregular-shaped spaces between the liner plates and the fuel clusters. Inserts, which can be attached to the liner plates, provide the capability of varying the graphite content in this region to reflect current and proposed lateral support designs.

The detailed design, experimental plan and procedural safeguards to be used in the conduct of NR-1 experiments will be reviewed by the Nuclear Safety Committee to ensure that this experimental program does not contain any unevaluated safety questions.

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Figure 5-5. NR-1 FCX Core and Reflector Assembly

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5.6 NUCLEAR DESIGN

5.6.1 Experimental Verification

The FCX was taken critical initially in April 1966 with a core consisting of 246 fueled clusters and a reflector of 2240 elements of solid AGOT graphite ($\rho = 1.72$). The fueled clusters each contained six reject NERVA elements of the highest loading (417 mg U/cc). A limited analysis of this core configuration was made using 40 group transport theory. As shown in Table 5-1, the predicted values of the number of clusters required for criticality and the reactivity worth of a peripheral AGOT graphite cluster agree well with the experimental values. Also shown in Table 5-1 is a comparison of predicted¹ and experimental control rod worths.

TABLE 5-1

FCX CORE EXPERIMENTAL VERIFICATION

<u>Item</u>	<u>Predicted</u>	<u>Actual</u>
Number of clusters for criticality	538	563
Reactivity Worth of Peripheral AGOT cluster	6.8¢	6.8 ¢
Control Rod Worths		
Central	3.4\$	3.2\$
Bank	20 \$	17.8\$

5.6.2 Parametric Analysis

An earlier extensive analysis¹ was made of the nuclear characteristics of graphite-uranium critical assemblies considered to fall within the scope of an FCX experimental program. By means of a parametric survey, such properties as critical radius, control rod worth, reactivity coefficients, and kinetic parameters are provided for a wide range of fuel loadings, void fractions, and reflector types and thicknesses.

Since the FCX program presently uses reject NRX fuel elements, the critical parameters most significantly affected by the presence of niobium carbide in the coolant channels (i.e., critical radius and control rod worths), were reanalyzed using the highest loading

¹ WANL-TME-749, Analysis in Support of the Flexible Critical Experiment, April 1964

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(417 mg U/cc) in a regular NRX-A2 fuel element (19 coolant channels). Also, the fuel cluster was assumed to consist of an assembly of seven fueled elements. A summary of the nuclear characteristics of certain key core configurations is provided in Table 5-2.

Since the excess reactivity of any FCX configuration is limited to 0.50\$, the predicted core dimensions of an FCX assembly will differ very little from the critical values presented herein and in the referenced study¹.

TABLE 5-2

FCX NUCLEAR DESIGN SUMMARY

Fuel Type: NRX-A2 fuel element coated with NbC

Typical Composition: 123 gms U/element (417 mg/cc)
528 gms C/element
41 gms Nb/element

Critical Core Radius (cm)

Bare	68.0
5" Graphite Reflector ($\rho = 1.75$ gm/cc)	52.0
12" Graphite Reflector ($\rho = 1.75$ gm/cc)	33.3
12" Water Reflector	53.3

Control Rod Worth (\$) *

Bare	13.6 (Central Rod = 2.41 \$)
5" Graphite Reflector ($\rho = 1.75$ gm/cc)	14.6 (Central Rod = 2.51 \$)
12" Graphite Reflector ($\rho = 1.75$ gm/cc)	16.6 (Central Rod = 2.83 \$)
12" Water Reflector	12.8 (Central Rod = 2.17 \$)

Neutron Lifetime (μ sec)

Bare	2.5
12" Graphite Reflector	73

Effective Delayed Neutron Fraction

β_{eff}	0.0071
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* Control rod configuration consists of one central rod and six peripheral rods.

¹ WANL-TME-749, Analysis in Support of the Flexible Critical Experiment, April 1964

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5.6.2.1 Critical Radius

The critical dimensions of a graphite-reflected assembly of NRX-A2 fuel elements in cylindrical geometry are summarized in Curve A of Figure 5-6. From the critical radius of a bare core, it is estimated that 650 fuel clusters (4550 elements) are required for an unreflected experiment. The minimum critical dimension is reached at the relatively large reflector thickness of 50 cm (20 inches) due to the modest slowing-down properties of graphite; approximately 80 fuel clusters (560 elements) are required for this configuration.

The reactivity effect of reflector density can be assessed from Curve B where low density graphite ($\rho = 1.60 \text{ gm/cc}$) was used for reflector material.

The reactivity worth of niobium carbide as a nuclear poison is demonstrated in Curve C where removal of this coating material has resulted in a significant decrease in the critical radius. The 530 clusters that comprise the bare, critical loading of such a core constitute 120 fewer clusters than the number contained in a core of coated fuel. Finally, the critical size of a core surrounded by a thick water reflector is shown in Curve D. In contrast with a graphite-reflected assembly, the critical radius reaches its asymptotic value ($\sim 53\text{cm}$) at a relatively small reflector thickness. For use in nuclear safety analyses, the excess reactivity of a critical bare core surrounded by a thick water reflector was calculated to be approximately 10.5\$.

5.6.2.2 Control Rod Worth

The reactivity worth of the FCX flux-trap control rods, arranged in a six peripheral rods - one central rod configuration, is summarized in Table 5-3 for various reactor assemblies. Since the rods are provided with fuel followers, it is obvious that these worths result from two distinct reactivity effects: removal of positive reactivity material (fuel cluster) from the core, and insertion of a nuclear poison (cadmium-wrapped polyethylene) in its place. The magnitude of these worths reveals that any one control rod is more than sufficient to override the 0.50\$ excess reactivity of the core. Furthermore, the overall control system will provide an expected shutdown margin in excess of 13\$.

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Figure 5-6. Critical Dimensions of FCX Assemblies

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TABLE 5-3
REACTIVITY WORTH OF FCX CONTROL RODS

<u>Reactor</u>	<u>Bare</u>	<u>Reflector 5 in. Graphite</u>	<u>Reflector 12 in. Graphite</u>	<u>Reflector 12 in. Water</u>
1. <u>Coated Fuel</u>				
Central Rod Worth (\$)	2.41	2.51	2.83	2.17
Total Rod Worth* (\$)	13.6	14.6	16.6	12.8
2. <u>Uncoated Fuel</u>				
Central Rod Worth (\$)	2.93	----	3.62	----
Total Rod Worth* (\$)	16.6	----	21.2	----
3. <u>Experimental Results</u>				
Central Rod Worth (\$)	----	----	----	----
Total Rod Worth (\$)	----	----	----	----

* The total rod worth includes the central rod and the six peripheral rods located approximately 18 cms from the axial centerline.

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Figure 5-7. Relative Control Rod Worth as a Function of the Depth of Insertion in Core

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A comparison of the data presented in Table 5-3 also indicates that rod worth in an assembly of uncoated fuel is approximately 20 percent higher than the value for a similar configuration using coated NRX-A2 components.

The rod worth of any axial position in the core is given by the expression:

$$\Delta k (Z) = \Delta k \frac{Z}{H} - \frac{1}{2\pi} \sin (2\pi Z/H)$$

where

k = reactivity worth of single rod or full bank of seven rods

H = core height

This curve is shown in Figure 5-7 where the total worth has been normalized to 1.0.

5.6.2.3 Kinetic Parameters

1) Temperature Coefficient

The temperature coefficient of an FCX core is derived from the physical effects of thermal expansion as well as the nuclear effect of Doppler broadening and the shift in thermal base.

Studies¹ on highly enriched uranium-graphite systems have shown that the magnitude and importance of the Doppler effect increases as the average neutron energy approaches the resolved resonance region (2-100 ev). Specifically, large negative Doppler contributions were calculated for reflected systems with C/U ratios in the range 200-600 and associated core-averaged median* fission energies ranging from several hundred ev to about 0.1 ev. A parametric study² of the Doppler coefficient

¹WANL-TNR-128, Volume 1, Reactor Analysis of NRX-A

²WANL-TME-749, Analysis in Support of the Flexible Critical Experiment, April 1964

* Half of the fission events occur below this neutron energy.

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for various FCX configurations, which have a minimum C/U ratio of 90, established that central median fission energies will fall in the range 30-1000 ev. (The core-averaged values will be approximately the same for bare assemblies and about half these values for reflected systems; FCX cores with a larger C/U ratio will have correspondingly lower median fission energies). In all FCX configurations, therefore, a large fraction of the uranium capture events will occur in the resolved resonance region so that these cores will exhibit a significant Doppler effect. The apparent success^{1, 2} in predicting the cold-hot reactivity swing measured during KIWI and NRX-A2 power tests indicates that the analytical model used in these studies is sound and that the conclusions reached above are valid.

As stated above, the predicted worth of the NRX-A temperature coefficient ($-0.20\%/^{\circ}\text{C}$) was essentially confirmed in the NRX-A2 power test. Because of the relatively loose bundling of the FCX core by its lateral support system, only axial elongation can be assumed for thermal expansion effects. If the contribution of radial expansion ($0.044\%/^{\circ}\text{C}$) is deducted therefrom, this value is considered to be a realistic estimate of the temperature coefficient for the reflected FCX assemblies as well as a conservative assessment of the value for a bare core.

2) Neutron Lifetime (ℓ^*)

The prompt neutron lifetime of an FCX assembly was found to vary from $2.5 \mu\text{sec}$ for a bare core to $73 \mu\text{sec}$ when the assembly is surrounded by a thick (12") graphite reflector.

3) Effective Delayed Neutron Fraction (β_{eff})

Based on a bare equivalent of the NRX-A reactor which is reported to have a β_{eff} of 0.0071 which is an accurate estimate of the effective delayed neutron fraction of an FCX assembly.

¹ WANL-TME-936, Development of a Method for Computing the Cold-to-Hot Reactivity Change in the NRX-A Assemblies, September 1964.

² NRX-A2 Six-Month Report.

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CHAPTER 6

PAX/FCX INSTRUMENTATION AND CONTROL

6.1 INTRODUCTION

Operation of the instrumentation and control system for the PAX and FCX reactors was initiated in early 1964 for the first PAX criticality. Since that time the system has been in almost continuous operation and has demonstrated a remarkable degree of reliability. In mid-1965 the control system was modified to allow operation of either the PAX reactor or the FCX reactors. Since the completion of these modifications both the PAX and FCX reactors have been successfully operated and the control and instrumentation system has performed satisfactorily.

Figure 6-1 is a diagram of the PAX/FCX instrumentation and control system. It indicates the relationship of the reactors, reactor console, and associated functional systems in a format which combines certain features of block, schematic and wiring diagrams. The information is presented in such a manner as to show detail where necessary for an understanding of system operation, and to retain some units in block form where detail would provide no information pertinent to an understanding of safety or control system operations. The format is arranged to indicate the flow of information from the neutron detectors to the control element drive system via either the automatic safety system or operator control. Dashed lines on this diagram show the instrumentation and control system interconnections between the reactor test cell and the control room or auxiliary areas. Certain individual functional units are identified by either dashed or solid lines. Control power and indicating light supplies, at 120V and 5V, 60 cycles, respectively, are obtained from distribution panels which are not shown.

The discussion of the instrumentation and control system is presented in the following sequence:

- 1) Section 6.2 discusses the means by which the PAX and FCX reactors are controlled from a single console.
- 2) Section 6.3 describes the overall reactor safety system which includes the reactor scram and interlock circuits. This section also defines the response characteristics of the safety system.
- 3) Section 6.4 describes the nuclear instrumentation which provides the reactor operator with information on the neutron flux or reactor power level and rate of change of flux level. Included is a discussion of the startup source.
- 4) Section 6.5 describes the PAX control drum drive mechanism and the FCX control rod drive mechanism.
- 5) Section 6.6 describes the control circuits which operate the drive mechanisms.
- 6) Section 6.7 discusses devices which are used to warn personnel of radiation hazards.
- 7) Section 6.8 discusses the control room and reactor console.
- 8) Section 6.9 discusses the normal and emergency power supplies.

6.2 SINGLE CONSOLE CONTROL OF PAX AND FCX

6.2.1 General

The PAX and FCX reactors are operated from a single control console; only one reactor can be operated at a time. Changing from the operation of one reactor to the operation of the other, which requires disconnecting the control and nuclear detection circuits of one reactor from the console and re-connecting those of the other reactor, is quickly and accurately done by means of two multiple-circuit patch panels.

6.2.2 Patch Panel Description (Figure 6-1, page 3)

Similar patch panels, which are safety interlocked (requires full panel insertion to be operable), are in wide use in many fields of circuitry and are considered extremely reliable. At WANEF, one panel, F, contains control element circuits, control position indicator

TEST CELL

Figure 6-1. Instrumentation and Control System for WANEF
(Sheet 1 of 4)

Figure 6-1. Instrumentation and Control System for WANEF
(Sheet 3 of 4)

Figure 6-1. Instrumentation and Control System for WANEF
(Sheet 4 of 4)

circuits, a neutron source transfer switch circuit (STC) and the patch panel interlock circuit. A second patch panel, G, contains signal leads for nuclear instrumentation, gamma channel leads and the patch panel interlock circuit. Each patch panel consists of a spring-loaded, hinged rack that locks one of two program boards into position with a motion that causes all mating pins to make positive contact simultaneously. Each program board, on which the circuitry for one of the reactors is wired in a semi-permanent manner, functions as a multiple pole, single throw switch. Figure 6-2 shows the relationship between the two reactor systems and the console. The PAX/FCX control circuit is as shown in Figure 6-1 except that two parallel circuits are available through the patch panels to support either the PAX or the FCX.

6.2.3 Neutron Source Transfer Circuit (STC) (Figure 6-1, pages 6-3, 6-4)

A single startup source is used for both FCX and PAX. The source is driven through a guide tube by means of a flexible spiral cable from the storage pig to the reactor. A double-solenoid-actuated switching mechanism enables either one of two paths to be taken by the source: one leading to the FCX reactor and one leading to the PAX reactor. Energization of the switch solenoids is wired into patch panel F such that insertion of either the FCX or the PAX program board ensures that the source path will be switched to the corresponding reactor.

6.2.4 Patch Panel/STC Interlock (Figure 6-1, pages 6-3, 6-6)

The two program boards pertaining to either one reactor or the other must be inserted in panel F and panel G, respectively, in order to complete the required logic to obtain any reactor control element drive OUT motor current. Unique circuits, which are patched in each program board, are continuous only if the two program boards for the same reactor are inserted and if the source guide tube is consistent with the patch panels. Any other combination of program boards and source guide tube position prevents completion of the interlock circuit; therefore OUT drive motor current is not obtainable.

Figure 6-3 shows the interlock circuit. It consists of two 90 V DC relays (K16, K17); their coil circuits go through both panels F and G to the Source Transfer Circuit where microswitches detect the position of the source transfer switch. When the program board-operated transfer switch is in position for the source to go to PAX, only the PAX-select relay

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Figure 6-2. Block Diagram of FCX-PAX Control System

Figure 6-3. Patch Panel/STC Interlock Circuit

K16 - PAX SELECT RELAY
 K17 - FCX SELECT RELAY
 KS - KEY SWITCH
 K14 - CRM FLUX UP
 K15 - TIME DELAY RELAY
 K24 - KEY-CAPTURE BYPASS CIRCUIT RELAY
 K50 - AC POWER RELAY
 K51 - BYPASS RELAY

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Figure 6-4. Control Element Drive Circuit

K-16 will be energized. Likewise, when the transfer switch is in position for the source to go to FCX, only FCX-select relay K-17 will be energized. The program boards for panels F and G are wired so that only the K-16 relay circuit is wired on the PAX program boards and only the K-17 relay circuit is wired on the FCX program boards. If the program panels are inter-mixed (e.g., PAX panel F and FCX panel G) neither K-16 nor K-17 can be energized. Thus, satisfaction of the interlock requires the startup source, the instrumentation, and the control elements to be consistent. From Figure 6-4 it can be seen that the control elements cannot be driven out unless the interlock circuit is satisfied and either K-16 or K-17 is energized to close contact K16B or K17B.

6.3 REACTOR SAFETY SYSTEM

6.3.1 Introduction

The reactor safety system consists of (1) interlock circuits which prevent reactor operation unless pre-established criteria are met, (2) automatic and manual scram circuits which will shut down the reactor if potentially unsafe conditions develop during operation, and (3) the control elements which provide the nuclear shutdown capability.

Nuclear shutdown for PAX is accomplished with the control drums described in Paragraph 4.2.6. To move these drums and hold them from their greatest shutdown position requires that current be supplied to the magnetic clutch coupling. On interruption of this current, the control drums are rotated to the least reactive or 0 degree position by the torque exerted through the scram spring. A detailed description of these components is given in Paragraph 6.5.1. PAX also has an auxiliary shutdown system which is described in Paragraph 4.2.7.

Nuclear shutdown for FCX is accomplished with the control rods described in Chapter 5. To move these rods and hold them from their greatest shutdown position in the core requires that current be supplied to the electromagnet on the drive mechanism. On interruption of this current, the control rods fall, by gravity, to their least reactive position with the poison section fully inserted into the core.

Figure 6-5. Simplified Scram and Interlock Circuits Functional Logic

Paragraph 6.3.2 discusses the scram and interlock circuits, and Paragraph 6.3.3 the response of the safety system to a scram. Paragraph 6.3.4 discusses the circuitry for the PAX Auxiliary Shutdown System.

6.3.2 Reactor Scram and Interlock Circuits

6.3.2.1 Interlock System (Figure 6-1, page 6-4)

The interlock system (see Figure 6-5c), which is part of the overall scram system (Figure 6-5b), provides (1) protection against personnel entry into regions of potentially high radiation fields during reactor operations, (2) protects against operation of more than one control element when these interlocks are by-passed for checkout maintenance, and (3) limits control element drive current so that no more than one element can be driven out at a time. All doors and gates to areas which may have high radiation fields during reactor operations have fool-proof interlock switches mounted on them. These switches control interlock relays (K54 through K59) whose contacts, in turn, control the Master Interlock relay K-18. In order to have access to the test cell and yard areas while maintenance or checkouts involving a single control element are in progress, the entry interlocks may be conditionally by-passed singly, or in unison by means of the Master By-pass (K66) (Figure 6-1, page 6-6), by switches located in the locked interlock panel, panel A (Figure 6-2). This panel can be observed directly by the reactor operator; amber lights on the control console and the interlock panel indicate when a by-pass is in effect. When any entry interlock is by-passed, the magnet current is automatically limited by a Simplytrol meter (S2)* to the value required for one magnet, and relay K26 can be satisfied only if 11 of the 12 keys to the control element mechanical locks are captured in the console. Thus, operation of no more than one control element is assured.

* "Simplytrol" is the trade name of a contact-making ammeter as manufactured by the Assembly Products Company of Chesterland, Ohio. The instrument contains break and make contacts which may be set to operate at a specific current value. As used at WANEF, instrument S1 has been modified to include a time delay to eliminate momentary contact operation on current surges. Instrument S2 does not have the time delay feature.

TABLE 6-1
REACTOR SCRAM CONDITIONS

<u>Item</u>	<u>Actuating Means</u>	<u>Actuating Signal</u>
Reactor Power	Electronic Trip	High power on either of the two controlling linear compensated ion chamber channels.
Reactor Power	Electronic Trip	High power level on controlling Log N compensated ion chamber channel (either Log N ₁ or Log N ₂).
Reactor Period	Electronic Trip	Short period on controlling reactor period compensated ion chamber channel. (PER-1 or PER-2)
Test Cell Veto	Manual Switch	Opened and reset manually by operating personnel. Located adjacent to the test stand in the test cell
Door Interlocks*	Limit Switches	Opening one of the following: Truck Doors Test Cell Door Rear Gate Labyrinth Doors Front Gate Emergency Exit
PAX Out-of-Reactor Position Interlock**	Limit Switches	Movement of "End-On" or "Side-by-Side" reactor experiment
Control Element Drive OUT Motor Current	Simplytrol Meter	50% overload of normal operating current for reactor in operation.
Console	Key Switch	Key turned to "OFF"
Console	Scram Magnet Switches (Individual)	Manual Activation

* May be bypassed during checkout and maintenance. Only one control element can be moved from "Full In" position because of restrictions by mechanical locks on control elements and by electrical switches and magnet current limiting circuit.

** Used when required.

Simplytrol ammeter S1 limits the drive motor armature current in the OUT direction to a value 50 percent above that required for one motor. Thus, a total scram will occur if, through a malfunction, two control drive motors are energized in the OUT direction at the same time, or if a control element sticks when being driven OUT.

6.3.2.2 Reactor Scram Circuit (Figure 6-1, pages 6-4, 6-5, 6-6)

6.3.2.2.1 General

The scram magnet circuit consists of a 90 V DC magnet power supply (which is energized through the console key switch), a manually-operate master scram pushbutton switch, the contacts of three heavy duty scram relays switches, as shown in Figure 6-5a. A scram is achieved by depressing the master scram switch, opening the solenoid circuit of one of the scram relays, or, in the case of a single control element, by depressing the appropriate individual switch.

The scram relay circuit is shown in Figure 6-5b. Table 6-1 summarizes the conditions under which the magnet current is interrupted causing a scram to take place. The complete circuitry is included in Figure 6-1, (Pages 6-4, 6-5, 6-6).

6.3.2.2.2 Safeguards Against Failure

The principle of using redundancy in the scram circuit as a safeguard against failure is carried out by (1) opening a manual master scram switch which has "break" contacts in both the 90 V DC supply and in the AC coil circuit of one of the scram relays (K21), (2) controlling each of the three separate scram relays (K21, K22 and K23) by some of the nuclear safety instrumentation, and (3) wiring two contacts of each of the scram relays in series in the scram circuit.

These three relays, K21, K22 and K23, have large contacts designed to prevent heavy DC arcing. Scramming twelve control drums simultaneously will open a 90 VDC circuit at 4.2 amps with an inductive load of 72 henries.

6.3.2.2.3 Test Cell Veto Switch (Figure 6-1, page 6-3)

The reactor test cell veto switch (Figure 6-5d) can prevent application of current to the magnetic clutch by means of K19. It is a large, clearly labeled and readily distinguished switch located just inside the test cell entrance. It must be reset from the same location. This switch provides personnel who are working in the test cell with the capability to forbid reactor control element motion. Appropriate red and green indicator lights display the status of the veto both in the test cell and at the reactor console.

6.3.2.2.4 Position Interlock for Reactor Experiments

Certain reactor experiments, as defined in paragraph 7.4.6.1, require the use of two position interlocks for each independently supported experiment. The interlock function is to scram the reactor if the experimental setup moves. These interlocks are heavy duty adjustable roller-arm switches (e.g., Unimax type KLBP-LH) which have been modified slightly to reduce the amount of overtravel. Such switches open their contacts in about 10 msec, well under the limit of 150 msec specified in paragraph 8.3.3.4. In order to achieve complete redundancy, one position interlock switch for each experiment is wired into the coil circuit of Position Interlock Relay No. 1 K-68 and the other switch is wired into the coil circuit of Position Interlock Relay No. 2 K-69. Contacts of relay K-68 are wired into one of the scram relays while contacts of relay K-69 are wired into another scram relay. A light on the console indicates when any switch is open. A reset button will be located in the test cell.

6.3.3 Response Characteristics of the Safety System

The discussion of the scram response time is divided into sections dealing with the electronics, magnet release, and control element insertion delay times for both PAX and FCX.

6.3.3.1 Electronics

Although the log N and linear amplifiers (presently Keithley Models 420A and 410) are DC amplifiers, they have an inherent bandwidth response for step changes in input current of 1KC to 10KC, nominally 3 db down at 1KC. Manufacturer's specifica-

tions indicate response times varying from 50 milliseconds to 6 seconds, according to input current level and/or range scale chosen. However, under reactor excursion conditions where the effective input voltage exceeds the feedback voltage limit (detector chamber saturation), the characteristics become independent of the RC feedback parameters, and the amplifier responds at its bandwidth frequency (1KC to 10KC). Thus, an instantaneous step change in current of several decades overload, as one would expect from a reactor excursion, causes an electronic scram response from 0.1 to 1.0 milliseconds. The relay in each electronic trip circuit (presently a Keithley Model 4103) requires about 10 milliseconds to separate its contacts. Tables 6-2 and 6-3 contain response times, including the Model 4103 electronic trip relay time, for the log N, period, and linear channels for several step and exponential ramp current inputs. The instrumentation used in these tests included detector cables approximately as long as those in use.

The scram relay (K21, K22, K23) dropout time is about 20 milliseconds.

6.3.3.2 PAX Magnetic Clutch and Drum Insertion

PAX scram magnet release time is about 35 milliseconds. Control drum rotation scram time depends on angular position of the poison plate prior to scram. The average control drum rotational speed is 550 degrees per second and rotation from 180 degrees to 15 degrees occurs in 0.3 seconds. Total scram time is about 375 milliseconds, as shown in Table 6-4.

6.3.3.3 FCX Magnet and Rod Insertion

FCX magnet release time is about 75 milliseconds. Rod insertion time from the full out position to the point where shock absorber action begins is 0.5 seconds. Total scram time is about 655 milliseconds as shown in Table 6-4.

6.3.4 PAX Auxiliary Shutdown System (Figure 6-1, pages 6-3, 6-4)

This system, described in Section 4.2.7, which involves the discharge of poison pellets into the reactor, is manually activated from the control room. Storage batteries provide the power to electrically operate the 12 V DC gear motors attached to the mechanical arms that open or close the valves in the storage hoppers.

TABLE 6-2

LOG N AND PERIOD TRIP RESPONSE

STEP FUNCTION

Period trip set at 5 sec		Log N trip set at 9×10^{-7} amps	
Initial Current Amp	Final Current Amps	Period Scram Time Millisec	Log N Trip Time Millisec
1×10^{-12}	1×10^{-6}	5.5	7.0
1×10^{-12}	1×10^{-3}	3.5	4.0
1×10^{-11}	1×10^{-6}	8	5.2
1×10^{-11}	1×10^{-3}	5	4.0
1×10^{-10}	1×10^{-6}	9	4.5
1×10^{-10}	1×10^{-3}	7	4.5
1×10^{-9}	1×10^{-6}	10	4.7
1×10^{-9}	1×10^{-3}	7	4.0
1×10^{-8}	1×10^{-6}	17	4.8
1×10^{-8}	1×10^{-3}	10	4.0
1×10^{-7}	1×10^{-6}	33	9.0
1×10^{-7}	1×10^{-3}	13	6.0

EXPONENTIAL FUNCTION

Period applied 5 sec		Period trip set at 5 sec		
Initial Current Amp	Final Current Amp	Period Scram Time Sec	Log N Scram Time Sec	Log N Trip Point Setting Amp
1×10^{-12}	1×10^{-9}	5.6	32.3	9×10^{-10}
1×10^{-11}	1×10^{-8}	4.4	29.0	9×10^{-9}
1×10^{-10}	1×10^{-7}	4.7	30.9	9×10^{-8}
1×10^{-9}	1×10^{-6}	5.1	29.5	9×10^{-7}
1×10^{-8}	1×10^{-5}	5.0	21.5	9×10^{-7}
1×10^{-7}	1×10^{-4}	4.5	9.7	9×10^{-7}

TABLE 6-3

LINEAR CHANNEL TRIP RESPONSE - STEP AND EXPONENTIAL FUNCTIONS

<u>Step Functions</u>				<u>Exponential Functions (Period Applied-5 Sec)</u>		
<u>Initial Current (Amp)</u>	<u>Final Current (Amp)</u>	<u>Step Function Trip Time (Millisec)</u>	<u>Trip Set (Amp)</u>	<u>Final Current (Amp)</u>	<u>Exponential Function Trip Time (Sec)</u>	<u>Trip Set (Amp)</u>
1×10^{-12}	1×10^{-7}	7	9×10^{-12}	1×10^{-11}	10.8	9×10^{-12}
1×10^{-12}	1×10^{-6}	5.4	9×10^{-12}			
1×10^{-12}	1×10^{-5}	4	9×10^{-12}			
1×10^{-11}	1×10^{-6}	6.5	9×10^{-11}	1×10^{-10}	11	9×10^{-11}
1×10^{-11}	1×10^{-5}	4	9×10^{-11}			
1×10^{-10}	1×10^{-6}	5	9×10^{-11}	1×10^{-9}	12.3	9×10^{-10}
1×10^{-10}	1×10^{-3}	6	9×10^{-10}			
1×10^{-9}				1×10^{-9}	12	9×10^{-9}
1×10^{-8}	1×10^{-6}	6.3	9×10^{-8}	1×10^{-7}	11.4	9×10^{-8}
1×10^{-8}	1×10^{-3}	4.5	9×10^{-8}			
1×10^{-7}	1×10^{-6}	6.6	9×10^{-7}	1×10^{-6}	11	9×10^{-7}
1×10^{-7}	1×10^{-3}	5.2	9×10^{-7}			

TABLE 6-4

TYPICAL COMPONENT SCRAM DELAY TIMES

	<u>PAX</u>	<u>FCX</u>
Log N level-period and linear amplifier*	10 millisecond	10 millisecond
Electronic trip	10 millisecond	10 millisecond
Scram Relay K21, 22 or 23	20 millisecond	20 millisecond
Magnet Release	35 millisecond	75 millisecond
Control Element Insertion	300 millisecond (180° to 15°)	540 millisecond (100% to 15%)
Total	375 millisecond	655 millisecond

* Measured with a 5 decade step current change simulating a reactor power excursion.

- ① LOG N & LINEAR N CHANNEL DETECTOR - WESTINGHOUSE 7741
- ② START-UP CHANNEL DETECTOR AND AUXILIARY DETECTOR CHANNEL DETECTOR - WESTINGHOUSE 6998
- ③ FOR 7.5 CURIE PoBe SOURCE

NOTE: DETECTORS POSITIONED 4.0 FEET FROM REACTOR CORE CENTER

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Figure 6-6. Typical Neutron Detector Ranges.

6.4 NUCLEAR INSTRUMENTATION

6.4.1 Reactor Nuclear Instrumentation - General (Figure 6-1, pages 6-3, 6-4)

The reactor instrumentation, applicable to both PAX and FCX, consists of one startup channel, two log neutron level-period channels, three linear neutron channels, and a reactor gamma level monitor. The startup channel, one log neutron level-period channel and two linear neutron channels are the minimum required for operation, with the log and linear channels being in the scram circuit. During reactor startup, as the reactivity and flux of the reactor is increased from shutdown level, a log N-period and a linear channel indicate on-scale and are used for information of flux level. The log N-period channel requires no scale changes during reactor operations, but the linear channels do require manual range changes as flux level is increased. The general block diagram of the instrumentation system, Figure 6-2, illustrates the relative layout of the facility network connecting the reactor to the control console.

The instrumentation channel detectors are positioned at several different azimuths around the periphery of the reactor assembly. Figure 6-6 illustrates the range and overlap of the detector current, neutron flux, and count rate of typical detectors.

The signal cable of the five ion chambers, the startup channel detector, the reactor gamma detector and the three auxiliary detectors of each reactor are routed to the separate sides of panel G which is the instrumentation multiple-contact signal patch panel (Figures 6-2 and 6-3). At the center of panel G are the common input signal lines to the console instrumentation. These are direct circuits to the instrumentation units (log N and linear N, etc.) in the control console in the control room.

6.4.2 Startup Channel (Figure 6-1, pages 6-5, 6-6)

A BF_3 proportional detector, for example a Westinghouse type 6998 with a sensitivity of 13 counts/nv, is normally used to monitor the neutron population from multiplication of the startup neutron source. The chamber-generated pulses are amplified by a pre-amp and a linear amplifier containing a pulse height discriminator.

The amplified and shaped pulses are fed to a log count rate (LCR) device which provides a logarithmic meter display of count rate. Within this LCR meter is a relay (K14) which will be energized when the LCR exceeds a minimum count rate level. Contact K14B (refer to Figure 6-4) then applies power to a time delay relay (K15) which will sound a warning horn for the duration of its activation period when the console is energized. After this delay interval, the "flux up" interlock in the control drive OUT circuit is satisfied. In parallel with the contact K14B are the contacts of a minimum neutron radiation level on the log neutron channel recorder. Thus, a neutron flux which exceeds the pre-set level on either the log channel or the LCR will actuate K15. The startup channel range essentially overlaps the ion chamber channels.

The output from the amplifier also drives a multi-decade scaler which provides pulses to drive a "popper" speaker. This "popper" provides audible indication, in both the test cell and the control room, of the flux level sensed by the startup channel. In addition, the "traffic lights", which give a visual indication of startup flux level in the test cell, are also controlled by the startup channel.

6.4.3 Log Neutron Level - Period Channels (Figure 6-1, page 6-4)

A compensated ion chamber, for example a Westinghouse Type 7741 having a sensitivity of 4×10^{-14} amperes per nv, is normally used to supply current to the log neutron level - period channel. The electronic instrument is presently a Keithley Model 420A log N - period amplifier that covers the range of current from 10^{-13} to 10^{-6} amperes. The output of both log N and period indications provide signal levels to separate electronic trips. Although this is a DC amplifier, upon step changes in input current as would occur in a reactor excursion, the amplifier responds with an inherent bandwidth response of 1 kc or more. This is equivalent to a theoretical response time of 1 millisecond. Actual measurements of the response of the log channel, including opening of the electronic trip relay, yield a response time of less than 10 milliseconds for a 5-decade current step from 10^{-12} amps to 10^{-7} amps for both the level and period trips. The damping control on the amplifier is set at minimum for this response. A factor of 2 increase in response time occurs with the damping set at maximum. The damping is normally set at a minimum value ($< 1/2$) consistent with no noise-induced scrams. This is equivalent to a step increase in power from about 1 milliwatt to the

100 watt level. The shutdown power level with the source in the core is approximately 1 milliwatt.

With step changes in power occurring at a higher power level, the log N channel has a faster response time.

6.4.4 Linear Channels (Figure 6-1, page 6-4)

The linear channels also normally employ compensated ion chambers, for example a Westinghouse Type 7741, to supply a micro-microampere Keithley Model 410 amplifier. The range of this instrument is from 3×10^{-13} to 1×10^{-3} amperes through twenty overlapping current ranges. The output of this amplifier provides a signal level to an electronic trip. A mechanical stop is provided to limit the maximum range interval consistent with the 2 kw maximum power level restriction. The response characteristics of the linear amplifier are essentially identical to the log amplifier with minimum damping described in the preceding section.

Both a linear channel and a log N level channel are monitored on a two pen crossover recorder used to obtain a continuous record of reactor operations.

6.4.5 Gamma Level

The reactor gamma monitor described in Paragraph 6.7.3 can be utilized for monitoring reactor power from the gamma flux measured with a gamma ion chamber.

6.4.6 Miscellaneous

6.4.6.1 Electronic Trip (Figure 6-1, page 6-4)

An electronic trip circuit, normally a Keithley Model 4103, is integrally incorporated into each of the scram channels - log N, period and linear. This unit is a voltage level sensing amplifier which operates a relay. The relay is normally energized; when de-energized it initiates a scram by opening one of the main scram relays (Figure 6-5b). The trip has a response time of 10 milliseconds.

6.4.6.2 Ionization Chamber Power Supplies (Figure 6-1, page 6-3)

The operating voltage to the ionization chambers is supplied by 300 volt batteries. The compensating voltage is normally supplied by a 22 1/2 volt battery. A separate set of batteries is located in the test cell for each ionization chamber.

6.4.6.3 Auxiliary Detector System (ADS) (Figure 6-1, page 6-3)

An auxiliary detector system, consisting of 3 BF_3 pulse channels, is provided for loading and initial criticality experiments. Each channel normally consists of a BF_3 proportional counter (W 6998), preamplifier, linear amplifier and discriminator scaler, power supply, and timing circuits. These channels will be used with the other reactor neutron channels to determine the effective multiplication ($1/M$) of the reactor during loading and the approach to critical.

6.4.6.4 Neutron Source (Figure 6-1, pages 6-3, 6-4)

A neutron source that is remotely inserted into or near the reactor is employed during reactor startup. The increase in neutron population raises the neutron flux above the background level of the startup instrumentation. This allows the reactor operator to observe changes in the neutron flux level as the control elements are moved to increase the multiplication of the reactor.

The polonium-beryllium neutron source has an initial strength of 7.5 curies, or approximately 2×10^7 neutrons/second. The source is doubly encapsulated in a capsule 1/4 inch in diameter, and approximately 1-1/2 inch long. It has a snap-on type connection for quick tie-in to the flexible drive cable. The neutron source is stored in an "S"-tube in the center of a borated paraffin-filled steel cylinder, 3-feet in diameter by 3-feet long. The source is moved through a stainless tube between the storage cylinder and either the PAX or FCX reactor by means of a flexible cable. The spiral-wrap cable is driven by a motor and geared worm arrangement. The motor is equipped with a magnetic brake. Capsule travel is controlled by cam-operated micro-switches which stop the drive motor when the capsule is at either the IN (at reactor) position or the OUT (centered in the storage cylinder) position. Position indication is provided at the reactor operator's console by IN and OUT indicator lights which are energized by the drive limit switches.

For PAX the neutron source IN position is normally at mid-core elevation just outside the pressure vessel cylinder; for FCX it is normally at the centerline under the core support plate.

6.5 CONTROL ELEMENT DRIVE MECHANISMS

6.5.1 PAX Control Drum Drive Mechanism

6.5.1.1 General

Figure 6-7 illustrates the control drum rotating mechanism. A DC shunt wound motor (1), having separate field and armature power supplies (Figure 6-1, page 6-3) and integral worm gear reducer (Figure 6-7, item 1) drives the vertical shaft of the control drum through a secondary right-angle speed reducer (2) and magnetic clutch (3). Mechanical and electrical stops (5, 8) limit angular travel of the drum between 0 and 180 mechanical degrees. A scram and shock absorption device (6) returns the drum to the 0 degree position when the magnetic clutch is disengaged. Tapered roller bearings (4), above the clutch and above the scram device carry radial and axial loads of the drum and scram mechanism. Drum position is transmitted to the control console by an electro-mechanical servo system transmitter (7).

6.5.1.2 Drive Motor and Speed Reducers

The 1725 RPM shunt wound fixed-field reversible gearmotor provides good speed regulation because the field strength of a shunt wound motor produces essentially constant speed regardless of load variation. Five steps of reduced-speed operation are available by means of armature circuit resistors, shunting relays, and speed selector switches (Figure 6-1, pages 6-3, 6-4). The resistors may be adjusted to provide steps as desired in an overall range of 9 to 1, resulting in control drum angular rotation of from approximately 10 degrees to 90 degrees per minute. Section 6.6 describes the control circuitry in more detail. The gearmotor - right-angle speed reducer maximum output speed at the control drum is 1/4 RPM for 90 degrees per minute; output torque is 30 foot-pounds.

Figure 6-7. PAX Control Drum Drive Assembly

6.5.1.3 Magnetic Clutch

The DC magnetic clutch, energized when the drive motor is energized, couples the control drum shaft to the driving unit. The clutch has a torque rating of 40 foot-pounds when energized. Pick-up and release times are each from 0.020 to 0.022 seconds at rated voltage. The control is further described in Paragraph 6.6.

6.5.1.4 Mechanical Stop

The adjustable mechanical stop (Figure 6-7, item 5) limits, within plus or minus 10 degrees, the angular movement of the control drum shaft from 0 to 180 degrees. This range permits angular adjustment of the stop with respect to the control drum, when required for purposes of either mechanical or nuclear alignment. The mechanical stop consists of two adjustable circular segments, each of which is fastened to the main bearing housing with three 5/16-inch diameter bolts. These are tightened in accordance with the stipulated design torque when the segments are adjusted to the desired positions. Maintenance procedures require that all bolts related to the mechanical stop be periodically retorqued.

In addition to the barrier provided by the mechanical stop, further angular movement of the shaft past 0 degrees is limited by backseating of the scram spring rack (Figure 6-7, item 6) against its range housing. Backseating of the rack is an absolute mechanical restraint against motion of the drum past the minimum reactivity position; however, the mechanical stop will normally halt drum motion, and is designed to allow close adjustment.

6.5.1.5 Scram and Shock Absorption Device

In order to scram a control drum, a spring loaded, hydraulically damped, device has been provided. When the control drum is rotated in a direction to increase reactivity, the spring is loaded and, following disengagement of the magnetic clutch, causes the drum to return to zero degrees. The spring provides an average scram velocity, between 180 and 15 degrees, of 550 degrees per second. Hydraulic damping is initiated

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Figure 6-8. FCX Control Rod Drive Unit

at the 10 degree position. After initiation of hydraulic damping, the control drive mechanism shaft continues to rotate at a reduced velocity until its angular movement is arrested by the mechanical stop. The spring provides a torque of 234 inch-pounds at 180 degrees and 16 inch-pounds at 0 degrees.

6.5.1.6 Position Indication (Figure 6-1, page 6-6)

Control drum angular position indication is provided by an electro-mechanical servo system. The rotor of the servo transmitter is mechanically engaged to the mechanism shafting through a 1:1 ratio gearing arrangement. The transmitted angular intelligence is reproduced at the servo receiver assembly, located at the reactor console, where angular position in degrees is read on a digital type indicator. The system is capable of indicating angular movement in either direction with an accuracy of six minutes and a reproducibility of one minute.

6.5.1.7 Limit Switches

In addition to the mechanism drive controls which are located at the reactor console, two adjustable microswitches are mounted in the drive assembly (Figure 6-7, item 8). Further described in Section 6.6, these switches normally stop the drive motor when the control drum reaches either travel limit.

6.5.2 FCX Control Rod Drive Mechanism

The drive mechanism shown in Figure 6-8 for the FCX reactor control rods uses a DC shunt wound motor connected to a pinion through a gear reducer. The upper portion of the control rod is a rack which mates with the pinion and is driven either up or down by the reversible motor. An electromagnet is fastened to the lower end of the rack by means of a universal joint; the magnet then mates with a corresponding armature attached to the control rod. Upon scram signal from the console, the magnet power is severed and the control rod enters the core by free fall. Rod withdrawal speed is adjustable in 5 steps from 15 inches per minute to 40 inches per minute by changing the motor voltage at the console.

* LOGIC ALLOWS ONLY ONE CONTROL ELEMENT TO RECEIVE POWER AT A TIME

Figure 6-9. Control Element Drive Circuit, Block Diagram

6.6 CONTROL ELEMENT DRIVE CIRCUITRY

6.6.1 General

The control element drive circuitry for either the PAX or FCX consists of the drive motors, drive motor relays mounted on the drive units, an array of lighted pushbutton switches which control the drive motor relays, and interlocks which prevent withdrawing control elements unless certain safety criteria are met; along with associated wiring and power supplies. The PAX control drum drive units and the FCX control rod drive units are essentially identical electrically. Therefore, the same control circuitry in the reactor console, the same position indicators and the same drive motor controllers are used for both the FCX rod drives and the PAX drum drives. Switching from one reactor's control element drive to the other is accomplished by changing the program board in patch panel F and manually re-setting the trip point of the Simplytrol^{*} armature current ammeter (S1).

A significant difference between the FCX and the PAX drive units is that during a scram on the FCX the control rod physically separates from the rack (an actual linear separation) whereas magnet release on the PAX drive causes control drum rotation (scram action) but no linear-wise separation (except the very small magnet-drive decoupling). One other difference is that the position indicators read directly in degrees for PAX operation and in centimeters for FCX operation.

6.6.2 Control Circuits (Figure 6-1, page 6-3)

The motion of a control element drive motor is controlled through a pair of drive motor relays mounted on the element drive units and labeled IN relay and OUT relay. Figure 6-9 shows a block diagram of the controls for the element drive circuits. The complete circuit is included in Figure 6-1. IN and OUT limit switches prevent over-driving the element against the physical stops. An array of lighted pushbutton switches on the reactor console

* "Simplytrol" = contact-making ammeter (see footnote for Paragraph 6.3.2.1)

TABLE 6-5

STATUS LIGHT COMBINATIONS FOR PAX AND FCX OPERATIONS

LIGHTS				Magnet Energized	FCX		PAX Drum
MAGNET	SCRAM	IN	OUT		Rod	Rack	
On	Off	On	Off	Yes	Full in	Not full out *	Full in
On	Off	Off	On	Yes	Full out	Full out	Full out
On	Off	Off	Off	Yes	Partially * withdrawn	Partially * withdrawn	Partially * withdrawn
On	Off	On	On	Yes	Full in	Full out	Not applicable
Off	On	On	Off	No	Full in	Not full out *	Full in
Off	On	Off	Off	No	Partially withdrawn and hung up	Partially * withdrawn	Partially * withdrawn
Off	On	On	On	No	Full in	Full out	Not applicable

* Position must be determined from digital readout device.

TABLE 6-6

CONTROL DRIVE STATUS LIGHTS

Light	Color	Status	
		PAX	FCX
IN	Green	Control drum fully inserted	Control rod fully inserted
OUT	Amber	Control drum fully withdrawn	Rack fully withdrawn
MAGNET	Blue	Clutch magnet energized	Rod coupling magnet energized
SCRAM	Red	Clutch magnet not energized	Rod coupling magnet not energized

contains IN and OUT switches for each drive motor. These switches, which are in the drive motor IN and OUT relay circuits, are wired so that, 1) if more than one OUT switch is depressed at one time, the current is interrupted to all OUT drive motor relays, and 2) if the IN and OUT switches for the same drive motor are depressed simultaneously, only the IN drive motor relay will be actuated. The array of lighted pushbutton switches also contains an individual scram switch for each magnet. These control circuits are wired from the reactor console via the main terminal panels B (in the control room) and C (in the test cell) to the central section of patch panel F. The PAX program board contains patches which connect the central section to the upper section from which the circuitry goes to the control drive stations (CDS) on the PAX test stand via panel D. The FCX program board contains patches which connect the central section to the lower section of panel F from which the circuitry goes to the CDS on the FCX test stand via panel E.

In addition to the individual IN switches, all control elements may be inserted simultaneously by means of the MULTIPLE IN pushbutton switch which activates four relays, each one of which has three sets of normally open contacts. These twelve sets of contacts activate the IN relays in the twelve control element drive units.

6.6.3 Control Drive Status Lights

The status of the control elements and magnets for PAX and FCX operations as indicated by combinations of the control drive status lights is shown in Table 6-5. The lights built into the IN and OUT push button switches are operated by the IN and OUT limit switch on the drum drive in the case of PAX and by the rod IN switch and the rack OUT limit switch in the case of FCX. The lights built into each individual scram switch are operated by a current sensitive relay in the individual magnet circuit which detects whether the magnet is energized or not. Table 6-6 summarizes these conditions.

6.6.4 Interlocks in Drive OUT Circuitry (Figure 6-1, pages 6-6, 6-5, 6-3)

There are several interlocks which prevent withdrawing control elements unless certain safety criteria are met. These interlocks do not apply to drive IN circuitry. Figure 6-4 shows the interlock relays. They require that the console key switch be turned ON and that the "flux-up" interlock be satisfied by having attained a preset flux level on either the log count rate meter (see Paragraph 6.4.2) or the log N recorder and by sounding the start-up warning horn. In order to have access to the test cell and yard areas while maintenance or checkouts are in progress involving a single control element, the entry interlocks may be by-passed. When in the BY-PASS condition the magnet current is limited by a Simplytrol ammeter, and all but one of the keys to the mechanical locks on the control elements must be captured in the reactor console. This condition is also described in Paragraph 6.3.2.1. During BY-PASS operations the flux-up interlock on control element drive OUT power is by-passed, because it is not needed when only one control element is to be moved.

To ensure indication of a "stuck rod" condition and also to prevent such a condition as two control elements being driven out simultaneously (e.g., as the result of a shorted control drive OUT pushbutton switch), the circuit from the control drive motor armature power supply for OUT motion is routed to the console through a Simplytrol ammeter. This Simplytrol (#1) will sense a 50 percent overload current for the maximum direct current drawn from only one control element drive motor and cause a scram. In addition, a self-latching relay (K 105) within the power supply opens and must be manually reset. A small delay action within the Simplytrol prevents starting loads from causing a scram.

6.6.5 Power Supplies

The closing of one of the five control element drive speed selector pushbutton switches on the console energizes a corresponding relay (K 107, K 108, K 109, or K 110) in the drive power supply located in the test cell. This relay closes its contacts to provide the necessary drive motor armature circuit resistance such that one of five voltages is selected which will move the control element at a certain fixed speed. The speed selector pushbutton, when

depressed, is lighted by a white light, thereby providing a continuous knowledge of the speed selected. There are two output voltage circuits, one for drive IN, and one for drive OUT.

A 95 volt DC power supply (Deltron) provides the armature current to the motor being controlled. An independent power supply (Deltron, DP 125-4) provides all motors simultaneously with non-adjustable field current.

6.6.6 Summary

To summarize the description of the control element drive circuit:

- 1) Each control element has an IN button and an OUT button on the console to control the direction of element motion. If both buttons are pushed simultaneously, the circuit is designed such that the IN direction has priority.
- 2) If the IN button of one element and the OUT button of another element are pushed simultaneously, both elements will simultaneously move in their respective directions.
- 3) Failure of the OUT switching logic which would permit both elements to move out if both OUT buttons were pushed simultaneously, is countered by the Simplytrol circuit. The Simplytrol circuit will deactivate the magnet and armature power supply circuits if the total current supplied to both drive motors exceeds the trip setting.

6.7 PERSONNEL SAFETY DEVICES

6.7.1 Reactor Status (Figure 6-1, page 6-6)

Energizing the reactor control console by use of the key switch on the console will cause an indication of REACTOR - STANDBY in green reactor status lights located in (1) control room, (2) front truck gate, (3) corridor, at main entrance, (4) rear truck gate, (5) corridor at south entrance and (6) WREC. The purpose of these lights is to inform facility personnel in all areas of the status of the reactor. After the control system has been checked out and all alarms cleared, if it is desired to insert the neutron source, the reactor status lights must be switched to a red REACTOR-ON position. The switch which activates the red REACTOR-ON status lights also removes the neutron source interlock (by energizing K 53), thus enabling source insertion, and disables the audible portion of the radiation alarms in the test cell. These

alarms would normally be activated by any critical operation; they are designed for monitoring during reactor shutdown. The last step, then, prior to driving the neutron source IN for reactor startup is to change the reactor status lights to REACTOR-ON.

Since indicator lights are used to describe reactor operational status, the absence of lights shall be considered to indicate a non-operational condition. Also, since some personnel engaged as reactor operators may not observe color with normal perception, all control switches and lights are unambiguously lettered.

The front and rear truck gates are interlocked closed through the scram circuit; each also contains a manual lock to prevent the gate from accidentally being opened. In the rear apron area there is one revolving amber light which is energized automatically when the radiation level in this area exceeds a preset value. A revolving red light on the front gate operates when the reactor status light is in the REACTOR-ON position.

6.7.2 Start-Up Warning Horn (Figure 6-1, pages 6-3, 6-6)

After energizing the console by the key switch, introduction of the neutron source will cause a start-up warning alarm to sound in the test cell area. A time delay relay (K 15), adjusted to 25 seconds, will automatically turn off the horn. The purpose of this alarm is to warn any personnel in the test cell of the reactor console status.

6.7.3 Reactor Gamma Monitor

A gamma dose monitor, presently a Victoreen ion chamber, is located one meter from each reactor. These monitors have their alarm points set at 100 mr/hr. The output is routed through patch panel G to an indicator located on the reactor console. When the monitor adjacent to the operable reactor alarms, a red light is displayed in the labyrinth access to the test cell and an alarm bell sounds in the test cell and at the console. Thus, a condition of high gamma radiation level is presented at the test cell entrance or as required for protection of personnel in test cell. During reactor operation, the audible portion of the alarm is deactivated by the REACTOR-ON console switch.

6.7.4 "Popper" and "Traffic Light" Monitors

During manual core changes, the neutron flux level is indicated by an audible "popper" that is activated by the startup channel. Thus, any increase in multiplication due to additional reactivity is indicated to personnel involved in the core change by the increase in "popper" frequency. To further indicate radiation levels, three "traffic lights" --one green, one yellow and one red-- are visibly positioned to the area during core loading changes. A green light indicates a low range count rate, a yellow light indicates a medium range count rate, and a red light indicates a high range.

6.8 CONTROL ROOM

6.8.1 General Description

The control room for the PAX and FCX reactors is located in the WANEF building area (Figure 3-4). This room, arranged as shown in Figure 6-10, contains the reactor control console, the main 120-volt AC entrance service power panel (BP1), the key-locked relay interlock panel (A), and main circuit wiring panel (B). Also located in this room are the Auxiliary Detector System control console with its separate 120-volt AC power entrance panel (BP2), and a panel (AA) which contains coaxial circuits leading to the test cell area and the counting room for interconnection of pulse circuitry.

6.8.2 Reactor Control Console

The reactor console contains the following major instrumentation and controls:

- 1) The linear and Log N channel micro-microammeters.
- 2) The startup channel and log count rate meter.
- 3) A two-channel strip chart recorder for continuous recording of a Log N Channel and a linear channel.
- 4) Position indicators for the control elements.
- 5) The control element IN, OUT, SCRAM, and MULTIPLE INSERTION pushbuttons.
- 6) The key switch, and console breaker switch.
- 7) The master scram button.
- 8) The DUMP button for the PAX auxiliary shutdown system.

CONTROL ROOM LAYOUT

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Figure 6-10. PAX/FCX Control Room Arrangement

- 9) The armature and magnet current Simplytrols.
- 10) Reset switches for the door and gate interlocks.
- 11) By-pass key switch circuit.
- 12) The readouts and trips for the reactor and area gamma monitors.
- 13) Drum speed switches, scram lights and resets, and other status lights and switches.
- 14) A meter showing reactor power level of the WREC reactor.
- 15) Insertion control for the neutron startup source.
- 16) Scaler and circuitry for the "popper".

6.8.3 Communication Systems

An intercom system provides continuous communication between the test cell and control room. The startup range "popper" is also transmitted to the test cell.

A closed circuit television system is normally provided for continuous control room monitoring of test cell activities. The camera has remote zoom lens control, remote focus (both electrostatic and mechanical), and pan and tilt control. Camera location in the test cell area is fixed for the greatest field of observation.

6.9 ELECTRICAL POWER

6.9.1 Power Distribution

Figure 6-11 shows the location of the various wiring panels throughout the WANEF. Power at 120 V, 60 cps, single phase, enter the building through Panel BP1 to Panel A (the interlock Panel). It also enters the reactor console directly. Single phase 110 V, 400 cps, power is contained in the console. All scram, safety and control circuits enter Panel B through terminal boards TB-1 through -24 located at the rear of the console. Interlock circuits from Panel B are routed through Panel A and then return through Panel B to the console. Panel C, the main test cell electrical distribution center, is terminal-to-terminal connected to Panel B. Non-reactor electrical circuits enter Panel C directly while all control circuits associated with the reactor enter Panel F. From there they are routed either to Panel D (PAX) or Panel E (FCX) depending on which program board is in place. All the control drive stations (CDS) feed Panels D and E. The signal cables from the detectors enter Panel G and from there are routed directly to the console instruments.

Figure 6-11. PAX/FCX Wiring Panel Locations

6.9.2 Emergency Power Supplies

There is no provision for emergency power 120 V, 60 cps for operation of the reactor instrumentation. Loss of normal AC power will cause a total scram.

The gamma radiation monitoring system is equipped with a battery emergency power supply so that the fuel storage vault, fuel assembly room, counting room and test cell can be monitored during power outage. Portable battery powered instruments are also available.

Standard wet cell emergency lighting is provided in the control room, machine shop, test cell, corridor, labyrinth, counting room, and fuel assembly area.

The PAX Auxiliary Shutdown System is also supplied with emergency power.

CHAPTER 7

CONDUCT OF OPERATIONS

7.1 INTRODUCTION

This chapter summarizes the measures established by WANL to ensure nuclear safety at WANEF, and describes the organization established to conduct the planned experiments.

Section 7.2, WANEF Organization, introduces the relationships of WANEF to affiliated organizations, defines the responsibilities and qualifications of the operating staff, and describes the technical support activities. Section 7.3 discusses the Experimental Program and Procedures in general terms, while Section 7.4, Administrative Control and Procedures, defines generalized procedures for: the control of liquids and flammable material in the test cell, control of personnel exposure, manual reactivity changes to the reactors, and the physical design of reactor experiments. Also given is a list of the written records required.

Because of the flexibility required in the operation of a critical facility, frequent modifications to the critical assembly and its associated equipment and to operational limitations and procedures are necessary. Clearly defined procedures for the review and approval of such modifications, documents and procedures which may arise during the course of operation, are set forth in the Operating Limits document.

7.2 WANEF ORGANIZATION

7.2.1 General

The administration and operation of WANEF is a responsibility of the Thermal and Nuclear Design Department of the NERVA Project of WANL. The organization of WANL is shown in

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Figure 7-1. WANEF/WANL Organization Chart

Figure 7-1. The test operation is under the direction of the Scientist-in-Charge, who is also the Manager, WANEF. His responsibilities extend to all activities, including reactor operations and personnel, on the WANEF premises. WANEF staff personnel and others extended access to the premises are responsible to the Manager for adherence to all rules and regulations established for access, experiment operations, and other activities at WANEF. When his absence is necessary, the Manager appoints a member of the WANEF staff (Experimentalist-in-Charge) to be responsible for the safe operation of the facility.

7.2.2 Relationship to the WANL Nuclear Safety Committee

The WANL Nuclear Safety Committee is responsible for the review of and consultation on all major aspects concerning nuclear safety of WANEF. The composition and operations of the Committee are detailed in the WANEF Operating Limits Document. Matters concerning WANEF operating procedures, modifications proposed for systems or facility designs, and all experimental programs, as affecting nuclear safety, are required to be submitted to the Committee at its regular or special meetings as found necessary.

7.2.3 Relationship to Other Waltz Mill Site Activities

The relationship of WANEF to other Waltz Mill Site activities is indicated in Figure 7-2. The dotted line connections represent communication by daily contact, committee meetings, mutual auxiliary services such as health physics, postal, works engineering, water, gas and electricity, and emergency procedures. To assure personnel safety at the critical facilities, the senior staff members of WANEF and WREC must keep each other informed of operating schedules, particularly when high radiation levels are anticipated. Senior staff members of WANEF must ensure that senior site personnel and local officials are cognizant of the proper procedures for handling WANEF emergencies.

Figure 7-2. Waltz Mill Site Organization Chart

7.2.4 Operating Staff

7.2.4.1 General

An important part of the safe operation of any critical facility is the clear designation of the personnel responsibilities and limitations. To ensure that all personnel are cognizant of their responsibilities and that each person is properly qualified for the position he holds, a list of position descriptions and the required minimum qualifications for each position is presented below. In addition to these specific categories, there are also technicians and non-technical personnel.

7.2.4.2 Position Descriptions

1) Scientist-in-Charge (WANEF Manager)

The Scientist-in-Charge has the overall final responsibility for the operation and safety of the WANEF. In addition to his technical responsibilities he makes decisions concerning personnel employed, operating procedures equipment purchased, visitors admitted, personnel training program, and all other matters that concern the safe and efficient operation of the WANEF. His decisions are subject to review and approval by higher management, but this does not relieve him of the final responsibility for the decisions.

2) Experimentalist-in-Charge

The Experimentalist-in-Charge assists the Scientist-in-Charge in meeting his responsibility for reactor operation. He has the authority to execute the experimental plan established by the Scientist-in-Charge, but not to change it in any way.

3) Experimentalist

The Experimentalist assists the Scientist-in-Charge or the Experimentalist-in-Charge to execute the experimental plan.

4) Reactor Operator

A Reactor Operator can be authorized to change the reactivity of a critical experiment from the control console as instructed by the designated Experimentalist-in-Charge. A Reactor Operator must be qualified to act in this capacity through training and a formal qualification.

5) Operator-in-Training

An Operator-in-Training is the lowest category of personnel who can change the reactivity of a critical experiment from the control console. He is only permitted to operate the control console when under the direct supervision and control of the Experimentalist-in-Charge.

6) Fuel Handler

The duties of the Fuel Handler include assisting with the loading and unloading of reactor fuel, poison wires and other operations dealing with changes to the reactor assembly. The Fuel Handler is only permitted to make changes in the reactor under the direct supervision of the Experimentalist-in-Charge and in accordance with a written procedure defining the changes and the sequence to be followed in making the changes.

7.2.4.3 Position Qualifications

To aid the Scientist-in-Charge in selecting personnel to fill the various categories of responsibility as described in the preceding paragraph, the following minimum qualifications are established for the positions involved in the operation of the reactors.

1) Fuel Handler

Those technicians which, through past performance, have demonstrated that they possess the following characteristics will be designated as Fuel Handlers by the Scientist-in-Charge:

- a. a mature and stable character,
- b. an ability to understand and follow instructions, and
- c. an understanding of the potential hazards involved in fuel operations.

2) Operator-in-Training

Selected and approved by the Scientist-in-Charge on the basis of previous performance record, in addition he must possess the characteristics of the Fuel Handler and a detailed knowledge of the reactor control system.

3) Reactor Operator

- a. Operator-in-Training for the time necessary to have completed 15 reactor checkouts, 15 approaches to critical and 15 hours of critical operation under the direct observation of at least an Experimentalist-in-Charge. It is intended that the approach time be included in the 15 hours of reactor operation.
- b. Demonstrate familiarity with the reactor, its systems, and its operation.
- c. Pass an oral, written and operating examination equivalent to those given by the AEC for licensed critical facilities.
- d. Approved as Reactor Operator by the Scientist-in-Charge.

4) Experimentalist

- a. An Experimentalist must satisfy the requirements for Reactor Operator as given above and in addition must hold a bachelor's degree or have had equivalent training.
- b. Approved as Experimentalist by the Scientist-in-Charge.

5) Experimentalist-in-Charge

- a. An Experimentalist-in-Charge must satisfy the conditions for Experimentalist and in addition hold a master's degree or have had equivalent training.
- b. A person qualified as Experimentalist-in-Charge shall have a detailed knowledge of the critical facility. His judgment and understanding of the facility and its programs shall be such that the Scientist-in-Charge may delegate the performance of experiments to him with confidence that they will be performed safely and efficiently.
- c. Pass an oral review by an examining board composed of the Scientist-in-Charge, Chairman of the Nuclear Safety Committee, and Manager of Nuclear Safety Analysis.

d. Approved as Experimentalist-in-Charge by the Scientist-in-Charge.

7) Scientist-in-Charge

- a. The Scientist-in-Charge must be a qualified Experimentalist-in-Charge.
- b. The Scientist-in-Charge must be a Manager of mature judgment, with a general knowledge of reactor physics and safety procedures, experimental physics background, and a specific knowledge of the design objectives of the project. He shall have the ability to train personnel assigned to him and to instill in them a cautious attitude with regard to safe experimental procedures.
- c. Appointed by the Project Manager.

7.2.5 Technical Support

7.2.5.1 Thermal and Nuclear Design Department

The Thermal and Nuclear Design Department of the WANL NERVA Project has the overall responsibility for development of the experimental program to be conducted at the WANEF facility. The experimental program will be developed by the WANEF staff personnel from Test Requests prepared by Thermal and Nuclear Design. These Test Requests will contain a description of the purpose, scope, and requirements of the experiment as well as a preliminary analysis of the reactivity effects and possible hazards involved in conducting the experiment. From the information contained in the Test Requests, the WANEF organization will make a final analysis of the experiment and based on the results of the analysis, will prepare Experimental Details Checklist (EDC) to be followed in performing the experiment. The review and approval procedure for the EDC is given in the Operating Limits document.

7.2.5.2 Engineering Support

The NERVA Project will supply the engineering support required for the successful operation of the facility. Persons with education and experience in mechanical,

electrical, reactor control, materials, health physics and industrial hygiene background are available as required.

In addition to the support provided by NERVA personnel, the Westinghouse Atomic Power Divisions have numerous facilities located at the Waltz Mill Site, which can be called on to lend support to the WANEF operation when necessary. These laboratory facilities include radiochemistry, spectrographic, analytical chemistry, hot cells, and the WREC. This organization is in Figure 7-2.

7.2.5.3 Health Physics

Health Physics coverage for the WANEF operations is provided by the Waltz Mill Site Health Physics Section under cognizance of the WANL Industrial Hygiene Section. Actual operations surveillance is provided by the site Health Physics Section, while the WANL Industrial Hygiene Section is responsible for maintaining the permanent exposure records for the WANEF personnel. The site Health Physics Section, composed of a health physics engineer and health physics technicians, is responsible for all normal health physics functions at the Waltz Mill Site including advising operating personnel on health hazards incident to the handling, use of, and exposure to radioactive materials and their associated ionizing radiation. The site Health Physics Section may recommend that an operation be stopped until corrective measures are taken if it feels that an undue personnel hazard exists. In addition, the site Health Physics Section is responsible for ensuring that adequate steps are taken to insure compliance with appropriate Federal and Commonwealth radiation regulations.

Normally WANEF has one full time health physics technician assigned to the facility to conduct standard radiation and contamination surveys; in the event that more coverage is required, additional personnel under the supervision of the health physics engineer will be provided.

As a supplement to the foregoing health physics coverage, a Health Physics Manual is used by the WANEF facility as a guide to safe operations. This manual, with

which all WANEF personnel are required to be familiar, includes standards, procedures, and regulations intended as precautionary measures to govern all work involving radiation or contamination hazards.

7.2.6 Security

Access to the WANEF premises is governed by the following security measures. The WANEF facility is divided into two areas in respect to security and/or safety. These are:

- 1) Operational Area - consists of the test cell, fuel storage vault, storage room and fuel assembly room, and will be accessible, after normal working hours, only to those responsible persons so designated.

- 2) Non-Operational Areas - refers to all rooms, buildings, or space occupied by WANL at the WANEF site with the exception of operational areas.

When the building has been secured, the security patrolman will make frequent rounds throughout the Non-Operational Areas, as previously defined, performing checks to ensure that all doors providing access to operational areas or points of entry to the WANEF building are secured in accordance with the WANL Security Manual. Guard reporting stations, utilizing Detex keys and clock, are located at various points throughout the WANEF premises to ensure periodic checks in these areas. Indoctrination is given to appropriately cleared security patrolmen with regard to facility safety. A check list, indicating all areas requiring inspection, is provided to the guards. Names and telephone numbers of the persons responsible for these areas are posted in the facility.

Access to the WANEF premises is controlled during normal working hours by an appropriately cleared person situated in close proximity to the front door, which is the normal access to the building. Persons other than the WANEF operating staff must indicate the purpose of the visit and will be placed in the custody of the person to whom the visit is directed. All Westinghouse Astronuclear employees shall be identified by picture badges issued and controlled by the security office, and the Westinghouse identification badge, which contains a photograph and

other identifying information. All other persons will be controlled in accordance with the WANL Security Manual.

Areas within the WANEF facility in which fuel is stored are;

- 1) The Fuel Storage Vault, which is equipped with a merchandise type vault door containing a four tumbler combination lock system.
- 2) The WANEF test cell area, which has two points of entry, is secured as follows:
 - a. The door entering the test cell from inside the WANEF facility is a steel door secured by a lock and alarmed to provide indication at the security guard house.
 - b. The 20 foot x 20 foot steel rollup door which provides truck access to the test cell is electrically operated, with controls located in the test cell. This bars all entry from the outside short of physical deformation of the door. An additional measure of security is taken by padlocking the door to the steel frame.

7.3 EXPERIMENTAL PROGRAM AND PROCEDURES

7.3.1 Experimental Program

The WANEF experimental program for both the PAX and FCX assemblies is specified in detail in a separate document covering all of the experiments envisioned over an interval of one or more contract years. The Nuclear Safety Committee reviews in advance the experimental program at least every six months.

The current program includes PAX or PTX experiments to measure core reactivity and shutdown, power distributions, neutron and gamma flux distributions and spectra inside and outside the reactor, control drum spans, fuel reactivity checks (FRC), reactivity worths of reactor components and materials both inside and outside the reactor, reactivity effects due to changes in fuel and reactor design, kinetic parameters using pulsed neutron and reactivity oscillator techniques. Part of the FCX program is directed toward developing the basic understanding of the NERVA type fuel moderator system. Measurements with bare and reflected cores will include those delineated above for PAX and PTX. The FCX reflector will consist of graphite,

beryllium, and other reflector materials that are of interest in the NERVA program. In addition to the basic studies, nuclear mockups of large NERVA reactor designs, such as NR-1, can be accomplished on FCX.

7.3.2 Experimental Procedures

The reactor experimental procedures used to implement the program are of two types:

(1) Standard Operating Procedures (SOP) which are contained in and are a part of the WANEF Reactor Operations Manual, and (2) the Experimental Details Checklists (EDC). The SOP's include procedures for the checkout and operation of the reactor and performance of experiments thereon. The experimental standard operating procedures presently include PAX and FCX Fuel Loading and Initial Criticality, Core Reactivity and Control Worths, Material, Structural and Environmental Worths, Power and Flux Measurements, Neutron Generator Measurements, Reactivity Oscillator Measurements, and Manual Changes in Reactivity.

In contrast to the SOP, which are generalized experimental procedures, an EDC is prepared for a particular experiment. The EDC describes the materials required, the hazards involved and the safeguards used to minimize them, the experimental procedure to be employed, and the data required to successfully complete the experiment. The review and approval procedures for SOPs and EDCs are explicitly set forth in the WANEF Operating Limits Document.

7.4 ADMINISTRATIVE CONTROLS AND PROCEDURES

7.4.1 General

Administrative controls and procedures for personnel, operations safety devices, and instrumentation, are contained in the Operating Limits document and the Reactor Operations Manual. The section herein specifies controls and procedures for the use of liquids and flammable materials in the test cell, control of personnel exposure, manual reactivity changes to the reactor, physical design requirements for in-reactor and out-of-reactor experiments. Also given is a list of the written records required.

7.4.2 Control of Hazardous Materials in the Test Cell

1) Liquids

The introduction of liquids into the PAX or FCX cores could present a very serious nuclear hazard. Thus, when the original safety report was written, water and other liquids were inadvertently considered to be unnecessary in the test cell and they were positively excluded. However, some liquids are in fact, required in the test cell when a reactor is present (e.g., in shock absorbers, crane crankcase, oil can, certain experiments containing liquids, etc.). To provide physical and administrative control of liquids required in the test cell when a reactor is present, so as to prevent an accidental criticality or reactor excursion, one of the following conditions must be met:

- a. Required liquids must be used in containers and in accordance with procedures such that it will be incredible for the liquid to get into the core. The Nuclear Safety Committee will review and recommend for approval the container design and the procedures to be used, such that the incredibility of liquid getting into the core is assured.
- b. When the reactors are in a shutdown condition, liquids may be used in limited amounts such that if the entire contents of the container are introduced into either

core the reactivity insertion is $<1.0\%$. Because the minimum isolated shutdown in PAX is 3.25% and the FCX is 10.0% , the accidental introduction of these reactivity insertion limits could not bring either core critical.

Detailed procedures based on these conditions are contained in the Reactor Operations Manual.

2) Combustible Materials

The use of combustible materials in the test cell was previously considered unnecessary and was strictly prohibited. However, some limited amounts of combustible materials are, in fact, required in the test cell when a reactor is present (e.g., polyethylene moderating material around the neutron detectors, paraffin mockup of a shield, plastic sheeting to limit contamination during fuel assembly and disassembly, and metal waste container for contaminated paper, shoe covers, etc.).

- a. Only those combustible materials, including liquids, necessary for the operation of the reactors or related activity may be maintained in the test cell. The presence of liquids must also satisfy the requirements of Paragraph 7.4.2, item (1).
- b. Approved fire protection equipment (refer to Section 3.7) shall be maintained in the test cell.
- c. Security guard patrols the WANEF area during non-working hours.

7.4.3 Control of Personnel Exposure

7.4.3.1 General

Personnel exposure to radiation is controlled at WANEF within the limits set forth in chapter 0524 of the AEC Manual, Standards for Radiation Protection, by several means. These safeguards are:

- 1) restriction of access to areas of potentially high radiation levels during reactor operation (see sections 3.2, 6.3) by locking doors and gates and

interlocking them into the reactor scram system, and by clearing monitoring areas outside of the exclusion area when proposed operations would lead to an integrated dose there in excess of 100 mr in one hour,

- 2) the monitoring of radiation levels in the test cell after operations before personnel may enter for work,
- 3) the use of devices such as reactor status lights, the startup warning horn, gamma radiation monitors and the "popper" and "traffic" lights in the test cell (see Sections 3.6 and 6.7),
- 4) the requirement that all personnel wear dosimeters and film badges.

The radiation levels associated with normal operation of the reactors, and those pertaining after reactor operation, are given in the following Paragraphs, 7.4.3.2 and 7.4.3.3, respectively.

7.4.3.2 Radiation Levels During Reactor Operation

Gamma and fast neutron radiation levels in and around the critical facility during operation of the PAX and FCX reactors have been determined. These are based on calculations which have been adjusted to available measured data obtained during operations from survey instruments outside of the exclusion area, and film badges and sulphur foil irradiations within the exclusion area. The dose rate contours for gamma radiation from PAX and FCX are shown in Figures 7-3 and 7-4 respectively while the fast neutron dose rate contours are shown in Figures 7-5 and 7-6 for the two reactors.

7.4.3.3 Radiation Levels Following Reactor Operation

The decay of neutron and gamma radiation following reactor shutdown determines the operations which can be performed in the test cell following reactor operation.

The gamma radiation level due to fission product decay, and the fast neutron flux ($E > 1$ Mev) level following a reactor run at constant power are shown in Figures 7-7 and 7-8. The gamma dose contribution from activation products and from delayed fissions is comparatively small compared with this fission product decay.

Figure 7-3. Gamma Radiation Levels During PAX Operation
(mrem/hr-watt)

Figure 7-4. Gamma Radiation Levels During FCX Operation (mrem/hr-watt)

Figure 7-5. Fast Neutron (>1 Mev) Radiation Levels During PAX Operation (mrem/hr-watt)

Figure 7-6. Fast Neutron (>1 MeV) Radiation Levels During FCX Operation
(mrem/hr-watt)

Figure 7-7. Fast Neutron Flux as a Function of Time After Shutdown
(10 Feet from the Reactor After a 20-Minute Run)

Figure 7-8. Gamma Dose Rates 1 Meter from Reactor
(1 Hour Run at Constant Power)

7.4.4 Required Written Records and Reports

7.4.4.1 Records

Accurate written records of the operation of the facility are kept to provide the primary evidence of compliance with the restrictions imposed by the WANEF Operating Limits document as well as data on experiments. These records will include, but not be limited to the various daily and periodic maintenance checkouts, manual reactivity changes and reactor loadings, reactor run data, unusual occurrences during reactor operation or experimental changes, explanations of all unscheduled shutdowns, control system malfunctions, descriptions and safety analyses of tests and experiments and fuel accountability records.

7.4.4.2 Reports

Westinghouse shall submit to AGC/SNPO an annual report of operating experience and activities for each calendar year. The report shall include:

- a. A description and hazards evaluation of each series of tests or experiments.
- b. Matters referred to, material submitted to, and actions taken by WANL Nuclear Safety Committee.
- c. Operational history of the facility indicating the number of experimental runs, integrated power, number of unscheduled shutdowns and their causes.

Westinghouse shall report immediately to AGC/SNPO in writing, any change in the physical condition of the reactor or its operating characteristics that might affect the safe operation of the reactor.

7.4.5 Manual Reactivity Changes to a Shutdown Reactor

7.4.5.1 Introduction

Manual changes to the reactivity of a shutdown reactor will be performed in accordance with approved detailed procedures contained in the Operations Manual. These procedures, including the installation or insertion of materials and structures in or near the reactor, assure that the reactor will remain subcritical at all times and that the operating limits are not violated.

Examples of such experiments include measurements involving the KIWI B5 shield, ETS-1 shield and engine component mockups, NTO detector environment mockup, reactivity oscillator, pulse neutron generator, insertion of important reactor materials for axial and radial worth measurements (e.g., polyethylene mockup of hydrogen enriched and depleted U, C, Nb, Ta, Al, Fe, oxygen mockup, B, Dy, Au, Ag, and Cd), insertion of mockup hydrogen for flux and power distribution measurements, insertion of neutron and gamma detectors for spectral measurements, installation or removal of core components such as a PAX beryllium sector or control drum, installation of a PAX control drum with different absorber materials or modified drum cylinder, removal of an FCX control rod and insertion of absorber materials in FCX reflector for worth and spectral studies.

7.4.5.2 General Operating Regulations

General operating regulations designed to protect personnel against the potential hazards of manual reactivity additions include the following requirements:

- 1) The experimental procedure must be reviewed and approved by the Scientist-in-Charge and the Manager of Thermal and Nuclear Design.
- 2) The procedure must be executed under the direction of an Experimentalist-in-Charge.

- 3) Two persons, one of whom is an Experimentalist-in-Charge, must be present in the test cell when the reactivity of the assembly is being changed; a third person qualified as a Reactor Operator must be on duty at the console; and the test cell-to-console intercom must be operational.
- 4) An audible indication of source range multiplication must be provided.
- 5) All control elements must be locked in the position of maximum shutdown.
- 6) The manual scram switch located in the test cell (Cell Veto) which can override any attempt to start the reactor must be in the VETO position.
- 7) No change in the reactivity will be made under the duress of a time limit.
- 8) The maximum number of personnel allowed within three feet of the reactor at any time shall be approved by the Scientist-in-Charge and shall be the more restrictive of (a) three persons, or (b) a number consistent with the minimum shutdown of that particular reactor.

A complete evaluation of the desired change based on experimental and/or analytical data will be made to determine the magnitude of the change. In making this evaluation, the effect of the proposed changes on the present reactor status will be considered in detail.

7.4.5.3 Manual Reactivity Changes in PAX

1) General

If the change is to be made to the PAX core, a conservative evaluation of the change magnitude shall be employed to ensure that the minimum shutdown of an isolated reactor is not less than 3.25\$ nor the excess reactivity more than 6.2\$ at any time during the change.

2) Reactor Assembly and Initial Operations

(1) Fuel Loading

A source will be installed at the beginning of loading with a source-core-detector geometry that will ensure an adequate detector response and a well behaved plot of inverse count rate. A minimum of three source range detectors will be used to obtain count rate data after each loading increment with the drums full in. The inverse count rate data will be plotted vs. the number of clusters loaded into the reactor. Fuel will be loaded into the PAX reactor with sufficient poison shipping wires to provide a shutdown of 20\$ if the fuel is previously untested or if the core assembly is a configuration not previously used. If, however, the core has been operated previously, a shutdown of 10\$ will be achieved using poison wires. The initial core loading will be no more than 10% of the total fuel complement and succeeding increments will be no more than 20 percent of the total fuel complement or one-half of the extrapolated additional fuel loading required for criticality, whichever is the smaller.

(2) Initial Criticality

Initial criticality is achieved by removing the poison wires from the core in increments, again making count rate measurements at each step both with the drums full in and full out. The inverse count rates are plotted against poison wire content. The initial poison wire removal increment will be no more than 20 percent of the wires and succeeding increments will be the smaller of 20 percent of the original content or 50 percent of the extrapolated additional poison removal required for criticality with drums full out. The final increment is selected by the Experimentalist-in-Charge so that the excess reactivity will be less than +0.50\$. After initial criticality is attained, the reactor shutdown reactivity will be measured.

(3) Clean Criticality

For new assemblies, the remaining poison wires are removed in reactivity steps no greater than 0.50\$ and the delayed critical bank position is

determined after each increment until all of the poison wires have been removed. The core shutdown reactivity will be monitored in the process, either by a plot of delayed critical bank position vs. poison wire content or by differential bank worth measurements.

For previously operated reactor assemblies, the remaining poison wires may be removed in a single step if the initial critical bank position agrees with prior measurements. The drum bank will be rotated in reactivity steps not larger than 0.50\$ and inverse multiplication plotted vs. bank position. If any extrapolation of delayed critical bank position deviates more than 0.50\$ from the previous delayed critical bank position, the reactor must be shut down until the discrepancy is satisfactorily explained by the Scientist-in-Charge or his designated alternate and judged not to affect safety.

3) Reactor Changes

(1) Only one type of reactor change can be made at one time.

(2) Up to 75 fuel elements may be interchanged before a criticality measurement must be made providing that the net reactivity increase from the interchange is less than 0.50\$. The core assembly frame will be used when more than three adjacent clusters or more than a total of four clusters are out of the core at one time.

(3) Before a control drum is removed from or inserted into its beryllium sector, the shutdown of the reactor will be increased by at least 1.0\$ by the insertion of poison wires into the core. After the control drum change is completed, a reactivity measurement will be made with the poison wires in place. If the control drum change has not compromised the minimum shutdown margin, the poison wires will be removed in reactivity steps of 0.50\$ or less. Only one control drum may be removed at one time.

(4) Reflector sector changes may be made, one sector at a time, providing the net increase in reactivity is first compensated for by the introduction of a corresponding negative reactivity by means of poison wires. Any drum change associated with a sector change is assumed to contribute a net reactivity increase of 1.0\$. After the interchange has taken place, a reactivity measurement will be made. If the change has not compromised the minimum shutdown margin, the poison wires will be removed in reactivity steps of 0.50\$ or less. Only one sector may be removed at one time.

4) Reactor Experiments

(1) General

Net reactivity changes will be calculated using experimentally determined reactivity coefficients based on NERVA type reactors where known. Conservative values based on polyethylene worth will be used for materials of unknown worth. Attention is called to Paragraph 7.4.6, Physical Design Requirements of Reactor Experiments.

(2) Reactivity Changes $\leq 0.50\%$

Manual changes will be made one at a time such that the net reactivity changes does not exceed +0.50\$. The reactor will often be operated following such a change but need not be for safety reasons. However, before a sequence of changes results in an accumulated net increase of greater than 0.50\$, a critical measurement of core reactivity will be made.

(3) Reactivity Changes $> +0.50\%$

If a manual change cannot be made directly in a reactivity step no greater than 0.50\$, as for example when a reactor experiment involves units which cannot be subdivided, the core will be poisoned first with material of known worth (poison wires) to compensate for the expected positive insertion. The experimental unit will be moved into position and a reactivity measurement made. The poison may then be removed in reactivity steps no larger than 0.50\$. No single experiment will be permitted which can override the minimum shutdown margin by accidental displacement during its installation.

7.4.5.4 Manual Reactivity Changes in FCX

1) General

In all changes to be made to the FCX core, where a large shutdown margin is always present ($>10\%$), consideration must be given to the 0.50% limit on excess reactivity. The procedures for the manual addition of reactivity depend upon the size of the positive reactivity insertion involved; "small" insertions where $\rho \leq +0.20\%$, "moderate" insertions where $0.20\% < \rho \leq 0.50\%$ and "large" insertions where $0.50\% < \rho \leq +5.00\%$. In all cases the net reactivity step is no more than 0.50% .

"Minimal" reactivity insertions are those which do not exceed $+0.02\%$. Such additions can be made directly (within the limitations on excess reactivity) and need not be followed by a reactivity measurement.

"Small" reactivity insertions are those which do not exceed $+0.20\%$. Such additions can be made directly (within the limitations on excess reactivity) and must be followed by a reactivity measurement.

"Moderate" positive reactivity insertions are those greater than $+0.20\%$ and not greater than $+0.50\%$. "Moderate" reactivity insertions must be preceded by the insertion of sufficient compensating negative reactivity (by supplementary poison of known worth or the removal of peripheral fuel or reflector clusters) such that after the addition of the positive reactivity, the value of k_{excess} will not exceed 0.20% . The reactivity of the assembly will be measured after the positive insertion is made. This measurement will be in the form of a period determination or a delayed critical rod position. If k_{excess} is negative, a measurement of the count rate will be made on the startup channel as a guide to the next step.

"Large" positive reactivity insertions are those greater than 0.50% and not greater than $+5.00\%$. "Large" reactivity insertions must be preceded by the insertion of sufficient compensating negative reactivity such that after the positive insertion is made, k_{excess} will not exceed -1.00% . Count rate data will be taken on the startup channel with the control rods fully withdrawn before and

after the "large" positive insertion is made. The inverse count rate data will be plotted as a function of poison removed (or clusters or positive shims added as the case may be).

Additional positive reactivity insertions will be made in "moderate" or "small" steps as defined above, each followed by count rate measurements with the control rods fully withdrawn. The extrapolation of the inverse count rate plot will provide a continual assessment of the excess reactivity of the core in the approach to critical.

After initial criticality has been achieved on an FCX core, reactivity coefficient measurements will be made in the core and reflector for materials of interest in order to make accurate estimates of the reactivity effect of subsequent reactor changes. Until the nuclear worths of these materials are experimentally verified, the magnitude of a change for known materials will be based on conservative reactivity coefficients obtained from graphite-moderated assemblies of similar composition while conservative values based on polyethylene worths are employed for materials of unknown worth.

2) Reactor Assembly and Initial Operations

A source will be installed at the beginning of loading with a source-core-detector geometry that will ensure an adequate detector response and a well behaved plot of inverse count rate. A minimum of three source range detectors will be used to obtain count rate data after each loading increment at 0, 50 and 100 percent control rod withdrawal. The inverse count rate data will be plotted vs. the total number of clusters loaded into the reactor. The initial increment of fuel clusters will not exceed 20 percent of the total anticipated complement of fuel and reflector clusters in the fully assembled reactor. Each succeeding increment will be limited to 20 percent of the total clusters or one-half of the remaining clusters required for criticality, whichever is the smaller. The final increments are selected by the Experimentalist-in-Charge so that the

excess reactivity will not exceed 0.50\$. After initial criticality has been achieved, reactor shutdown will be determined by rod drop or equivalent techniques.

3) Reactor Changes

(1) Only one type of reactor change can be made at one time.

(2) Fuel and/or reflector elements may be interchanged providing that the net increase in reactivity does not exceed 0.20\$. The core assembly frame and lateral support bars will be employed as needed to provide support during such changes. The interchange will be followed by a reactivity measurement.

(3) Control rods may be removed providing the minimum shutdown margin of 10\$ is maintained by supplementary nuclear poison or by removing fuel or reflector clusters. The amount of negative compensation will depend upon the worth of the configuration of control rods remaining. In returning a control rod to the core, additional negative compensation will be required to maintain the minimum shutdown margin of 10\$, assuming that the inadvertent omission of the cadmium part of the control rod has occurred. Such an omission would result in a positive reactivity insertion of about 2.0\$ per rod at the core center. No more than two control rods can be returned to the core at one time.

4) Reactor Experiments

Installation of reactor experiments in and around FCX will follow the criteria (given in Paragraph 1), above. Proper consideration will be given to the radial and axial variations in the nuclear worth of experimental materials or equipment. Attention is called to Paragraph 7.4.6, Physical Design Requirements of Reactor Experiments.

7.4.6 Physical Design Requirements of Reactor Experiments

The procedures for installation of reactor experiments are given in Section 7.4.5, "Manual Reactivity Changes During Reactor Shutdown." The following sections deal with the engineered safeguards, procedural requirements and limitations placed upon the physical design of reactor experiments to ensure that an unevaluated safety question is not introduced due to the failure of an experiment during reactor operation. For convenience, the discussion is divided into Out-of-Reactor and In-Reactor experiments, the former covering both "Side-by-Side" and "End-On" experiments. Out-of-Reactor experiments are those which are located outside the envelope of the reactor. In the PAX, the envelope consists of the pressure vessel, cluster plates and support blocks, while in the FCX, it contains the cluster plates and radial boundary of the core including the reflector if one is present.

7.4.6.1 Out-of-Reactor Experiments

Examples of this type of experiment include a Privy Roof Mounted Shield (PRMS) or mockup, placement of the neutron generator next to a reactor, shielding measurements with the KIWI B5 shield, dosimetry measurements about the reactor (e.g., mockup of NRDS detector tank), worths of concrete, stainless steel structures, and other materials as a function of distance from the reactors, and worth and dosimetry measurements with the ETS-1 shield and other engine component mockups.

1) Out-of-Reactor Experiments on PAX

The installation and use of Out-of-Reactor experiments is permitted on PAX if the following conditions are met:

(1) The design must ensure that the effects of neutron reflection do not reduce the shutdown reactivity below the minimum margin for planned placement nor override the control capability of the drums during failure.

(2) The mechanical design of the experiment employs engineered safeguards against a failure that would permit the material to be displaced to

a more reactive position. The structural design of these safeguards will have a minimum load safety factor of five based on ultimate strength, and gears, pulleys, cables and parts subject to wear will have a safety factor of eight.

(3) Any experiment which, if it did fail, would insert more than 1.0\$ in reactivity into PAX must include position interlocks which will initiate a reactor scram if the experiment experiences any displacement during reactor operation. These position interlocks, which are described in 6.3.2.2.4, must be checked out daily when in use and shall be supported independently of the experiment.

(4) When an experiment involves a combination of a radial and end-on reflector, each will be supported independently of the other to prevent the simultaneous collapse of both reflectors. Each part will be provided with position interlocks.

(5) Any experiment, involving hydrogenous liquids must be designed so that such fluid hydrogenous materials cannot get into either PAX or FCX. The conduct of any PAX liquid experiment will require prior review by the Nuclear Safety Committee to ensure that the entrance of liquid into either the PAX or FCX core is incredible. In PAX, for example, compliance with this requirement will take the form of a metal plate bolted to the nozzle end of the pressure vessel when a radiation shield (e.g., PRMS) or a mockup of the NTO detector tank containing water is mounted near the reactor. During an experiment on PAX (such as the PRMS) which, in failing, could splash liquid onto the FCX, the FCX reactor will then also be isolated by means of barriers between the PAX experiment and the FCX core.

2) Out-of-Reactor Experiments in FCX

The installation and use of Out-of-Reactor experiments is permitted on FCX if the following conditions are met:

(1) The reactivity effect of the experiment when placed next to the FCX must not compromise the 0.50\$ limit on excess reactivity.

(2) The reactivity effect of the experiment if it should move closer to the reactor must not compromise the 0.50\$ limit on excess reactivity.

(3) The reactivity effect of removing the experiment must not compromise the 0.50\$ limit on excess reactivity.

(4) Conditions (2) and (3) apply to each independently-supported experiment.

(5) Any FCX experiment involving liquids will require prior review by the Nuclear Safety Committee of the container design and of the procedural safeguards to ensure that entrance of such materials into either the FCX or the PAX core is incredible.

7.4.6.2 In-Reactor Experiments

Examples of In-Reactor experiments which are covered by this section include insertion of important reactor materials such as polyethylene mockup hydrogen, depleted and enriched U, C, Nb, Ta, Al, Fe, oxygen mockup, B, Dy, Au, Ag, Cd, and Lu for axial and radial worth measurements, installation of a reactivity oscillator or the neutron generator target for kinetic measurements, installation of neutron and gamma dosimeters, detectors and chambers such as Bragg-Grey chambers for flux and dose measurements, and PAX control drum replacements involving new vane materials and drum barrel modifications.

1) General Requirements

The physical design of an in-core experiment must consider not only the reactivity associated with the configuration of interest but also the accidental displacement of the experiment into a more reactive position. In the event that in-core experiments are based on other design criteria, e.g., engineered safeguards, material-behavior experiments, etc., the Nuclear Safety Committee will review the experiment design and procedures.

Inserts which are supported independently of one another are considered to be separate in-core experiments. The structural design of these supports will have a minimum load safety factor of five based on ultimate strength, and gears, pulleys, cables and parts subject to wear will have a safety factor of eight.

The design of experiments involving liquids will be based on the above general criteria and the detailed design and procedural safeguards will be reviewed by the Nuclear Safety Committee to insure that no credible accident can result in a reactivity increase in excess of 1.0\$ in PAX or Δ in FCX where $\Delta = 0.50\$ - \rho_{\text{excess}}$.

2) Additional Requirements for PAX In-Reactor Experiments

In addition to the general requirements in paragraph 1), above, the axial displacement of the experiment must not compromise the minimum shutdown margin, nor may it produce a net reactivity change in excess of 1.0\$.

3) Additional Requirements for FCX In-Reactor Experiments

In addition to the general requirements in paragraph 1), above, the axial displacement of the experiment must not compromise the minimum shutdown margin, nor may it produce a net reactivity change in excess of Δ where $\Delta = 0.50\$ - \rho_{\text{ex}}$ where ρ_{ex} is the excess of the core with the experiment located at its normal position.

CHAPTER 8

PAX SAFETY ANALYSIS

8.1 GENERAL

Since the time that criticality of the PAX reactor assembly was first achieved in 1964 much has been learned concerning its design characteristics, the effectiveness of WANEF administrative controls and procedural safeguards, and its vulnerability to various types of reactivity insertions. The safety evaluation presented herein constitutes an updating of the analysis performed in the original Safety Analysis Report as well as a review of changes in operating procedures (See Chapter 7) and extensions in the experimental program, primarily the design of side-by-side and end-on experiments.

8.2 SAFETY EVALUATION

8.2.1 General

Mechanisms by which inadvertent reactivity insertions can occur in the PAX assembly are derived from basic features of the reactor design as well as from the type of experiments that comprise the PAX experimental program. Before a detailed safety analysis of the PAX and its test program is presented, those aspects which have particular nuclear safety significance will be briefly discussed.

The reactivity span that can be achieved with the PAX control drum system is $\sim 9\%$ which includes the minimum shutdown margin of 3.25%. Since the beryllium reflector does not have sufficient thickness to provide full neutron reflection and since the graphite support blocks at the end of the core constitute only a minimal neutron reflector, the presence of massive reflector materials immediately adjacent to the reactor assembly can cause an increase of several dollars in reactivity. As can be expected from an intermediate spectrum core, the reactivity coefficient of hydrogen in PAX is large ($\sim 20\%/kg$ at the center of

the core) and hence the permissible quantity of hydrogenous material in the core is small. It is estimated that approximately one-half gallon of water in the coolant channels would override reactor shutdown.

To evaluate the detailed nuclear design of various NERVA reactor configurations, frequent fuel changes are made in the PAX core and occasional modifications to the outer reflector and control drums are required. Furthermore, equipment and materials are placed in or around the reactor assembly to mock-up test configurations. Such experiments affect the reactivity of the PAX assembly and will be conducted in a manner that will not lead to excessive or inadvertent reactivity insertions.

These are the considerations on which the safety analysis of the PAX experimental program have been based. The design of the PAX assembly and its control system, as well as the basic operating procedures which serve as a guide for the detailed experimental plans, have been reviewed to identify credible accidents. In this safety analysis, an accident which requires the simultaneous occurrence of two unlikely and independent events is considered to be incredible.

8.2.2 Manual Addition of Reactivity

During the course of the PAX experimental program, manual changes in reactivity are made routinely by operations personnel. This occurs during the very early phases of a new core configuration when core loading and initial criticality are accomplished and continues during the program when changes to the reactor assembly and installation of intra-core and extra-core experiments are frequent. During such changes personnel do not have the protection afforded by facility shielding and therefore numerous safeguards have been incorporated into this phase of the experimental program to insure effective protection against inadvertent criticality.

General operating regulations designed to protect personnel against the potential hazards of manual reactivity additions include the requirements that:

- 1) The experimental procedure is reviewed and approved by the Scientist-in-Charge and the Manager of Thermal and Nuclear Design.
- 2) The procedure is executed under the direction of an Experimentalist-in-Charge.
- 3) Two persons, one of whom is an Experimentalist-in-Charge, are present in the test cell when the reactivity of the assembly is being changed; a third person qualified as a reactor operator is on duty at the console and the test cell-to-console intercom is operational.
- 4) An audible indication of source range multiplication is provided for operations involving personnel in the test cell.
- 5) All control drums are locked in the 0° position (maximum shutdown) except as provided by checkout and reactor change procedures.
- 6) A CELL VETO is activated whenever personnel are making changes to the reactor. This interlock, which will override any attempt to start the reactor, has no bypass.

In addition, there are other operating restrictions and design features specific to each experimental procedure that will further enhance the safety of the experiment.

8.2.2.1 Reactor Assembly

The availability of experimentally determined reactivity coefficients for all major reactor materials will permit a preliminary assessment of the shutdown margin in any configuration of interest. Errors in the identification or installation of such materials during the assembly of the PAX reactor could have a direct effect on its reactivity balance, i.e., the amount of shutdown available in the control drums. For example, the fuel loading error arising from complete substitution of 500 mg U/cc fuel for the normal PAX inventory would

result in a reactivity increase of approximately 7.5\$. Increasing the beryllium reflector thickness by about 2 inches will produce a maximum reactivity addition of about 3\$.

Safeguards against hazards resulting from such errors include the use of supplementary nuclear poison. in the form of poison wires, which is placed in the fuel clusters prior to core loading. When a new core of untested fuel is involved enough poison to provide a shutdown of 20\$ will provide a suitable safety margin against loading errors during assembly of the core and against changes to the outer reflector for which adequate reactivity compensation by poison wires has not been made.

In the case of previously utilized cores. the possibility of fuel loading errors is reduced because each fuel element is inscribed with a serial number which identifies the uranium loading and permits clear identification of those elements whose uranium content has been previously verified by a reactivity measurement. A shutdown of 10\$, as provided by the control drums and supplementary poison, will result in a suitable safety margin even for a reflector change error combined with a fuel substitution error.

As assembly of the core proceeds outward to the outer reflector, segments of a core assembly frame are added to provide the necessary support for the fuel clusters. The actual core loading is performed in incremental steps with an inverse count rate measured at each step. A plot of these data during the loading sequence provides a continuous assay of core reactivity. In this manner, loading errors can be detected well before a hazardous condition can develop.

8.2.2.2 Initial Criticality

Initial criticality following assembly of a PAX core, as described above, is achieved by removing the poison wires from the core in increments and obtaining count rate data after each incremental change for several control drum positions, including the 180° position. In this manner, the reliability of the inverse multiplication plot is assured. The

initial increment of poison removed will be no more than 20 percent of the initial content; each succeeding increment removed will be no more than 20 percent of the initial poison content or one-half of the extrapolated additional poison removal required for criticality with the drums at 180 degrees, whichever is less. The final poison increment that must be removed to achieve criticality is selected such that excess reactivity is limited to 0.50\$ and an experimental measurement of reactor shutdown is made. Such a procedure provides the earliest evaluation of control span and ensures that the reactivity balance for a clean core will be acceptable. The minimum shutdown of an isolated PAX assembly has been set at 3.25\$ (see Paragraph 8.2.6) to provide an adequate margin of safety during the operational and shutdown phases of the experimental program.

For configurations of new fuel, the remaining poison wires are removed in reactivity steps no greater than 0.50\$ and the critical bank position is determined after each increment until a clean core is obtained. If the initial criticality of a previously utilized core agrees with prior measurements, the remaining poison wires are removed in a single step and a gradual approach to clean critical is performed by rotating the bank in increments not larger than 0.50\$. A plot of inverse multiplication versus bank position extrapolated to delayed critical will confirm the reactivity of the clean core. Any measurement that deviates more than 0.50\$ from the previous value must be followed by a reactor shutdown until the discrepancy is satisfactorily explained and determined by the Scientist-in-Charge or his designated alternate not to affect safety.

These procedures for achieving initial criticality provide the assurance that the minimum shutdown margin will not be compromised and that operating personnel are protected against accidental criticality.

8.2.2.3 Reactor Changes

Nuclear mockups of new core and reflector configurations will require frequent changes to the reactor assembly. For this purpose, a flexibility has been incorporated

in the PAX assembly that permits the removal and/or re-insertion of any number of fuel clusters in the core. Furthermore, the outer reflector has been designed to allow removal of any one sector and its drum without disassembling the core or the other eleven sectors.

The reactivity effects that may occur in the process of making such changes are controlled by operating procedures and are limited by the nuclear worth of the components involved. Although each type of reactor change is accomplished within specific procedural controls as described below, all are governed by the basic ground rule that only one such change can be made at one time. In this manner, the potential hazards associated with making multiple and concurrent reactivity changes are avoided.

At the center of the core where the uranium reactivity coefficient is highest, a fuel cluster has a nominal absolute worth of 0.33\$. Only minor deviations from this value are anticipated for advanced design mockups since a 50 percent increase in fuel loading would be required to augment cluster worth by 0.05\$. Operating procedures allow 75 fuel elements to be interchanged before a criticality measurement is required, providing the net reactivity increase from this interchange is analytically determined to be less than 0.50\$. For example, an inadvertent interchange of seventy-five 500 mg U/cc fuel elements for 136 mg U/cc elements at the periphery of the core yields a loading error of 1.0\$ reactivity insertion - a value which can be safely accommodated by the minimum shutdown worth of the control drums. To prevent fuel from accidentally shifting to a location of higher reactivity worth, a core assembly frame is employed when more than three adjacent clusters or more than a total of four clusters are out of the core at one time.

The effect of removing a control drum from the reflector while remaining fixed in the 0 degree position is estimated to be a net reactivity increase of approximately 0.20\$. However, allowance is made for two eventualities when a control drum change is made: inadvertent drum rotation to its most reactive position during removal from or insertion into the beryllium sector, and failure to replace a poison vane of adequate nuclear worth on the drum assembly following modifications thereto. Either situation could lead to a reactivity

insertion of about 1.0\$. Although protection against these potential human errors is provided by the shutdown inherent in the remaining control drums, the operating procedures provide that 1.0\$ in poison wires will be inserted in the core prior to removal or insertion of a control drum. After the change has been completed, a reactivity measurement is made with the supplementary poison (known worth) in place. The poison wires are subsequently removed in 0.50\$ increments only after the reactivity check has established that the control drum change has not compromised the minimum shutdown margin.

Removal of a reflector sector from the PAX assembly increases neutron leakage and therefore lowers nuclear reactivity. The effect of replacing a sector, however, depends on the extent of any modifications made to the sector itself as well as on changes affecting the worth of the control drum. In such an operation, the core is first poisoned with wires by an amount equal to the net increase in reactivity (the drum is assumed to produce a net reactivity increase of 1.0\$) with subsequent removal of the wires in 0.50\$ increments in accordance with PAX reactivity change procedures.

The design features and procedural safeguards described above will ensure that the reactor shutdown margin is not adversely affected during the course of any reactor change operation and, therefore, that inadvertent criticality will not occur.

8.2.2.4 Reactor Experiments

The effect of experiments on the reactivity of the PAX core is recognized and procedures have been instituted to limit the magnitude of the reactivity changes that can be made during installation of such experiments. These procedures provide that all changes will be accomplished in increments of 0.50\$ or less, either directly when the change can be made in small steps (e.g. calibration wires) or indirectly by the use of supplementary poison when a large, single change is involved. Furthermore, no single experiment will be permitted which can override the minimum shutdown margin by accidental displacement during its installation. Experimentally determined reactivity coefficients based on NERVA-type

reactors are used to evaluate the reactivity of known materials, while conservative values based on polyethylene worth are employed for materials of unknown worth. In this manner, the installation of reactor experiments can be achieved without risk to operating personnel.

8.2.3 Malfunction of Instrumentation and Control System

Proper performance of the nuclear instrumentation and control system is essential for the safe operation of PAX, as is the case for any critical assembly, and numerous safety features have been incorporated therein to achieve high reliability.

During shutdown periods, personnel in the vicinity of the test stand as well as in the test cell itself are protected against the potential hazards of drum runout by the use of mechanical locks and electrical interlocks in the reactor control system. Interlocks are provided at strategic access points within the WANEF facility (e.g., truck door to test cell, front gate, labyrinth door) thereby preventing reactor startup if the potentially high radiation areas have not been properly cleared of personnel. If any one of these interlocks is bypassed to permit maintenance or checkout of the control system, drum motion is restricted to a single drum by a circuit design which requires that all but one of the active control drums be key-locked and which limits the magnet current to the value needed to operate one magnet. Since a minimum of four drums is normally required to achieve criticality, operating personnel are fully protected against a nuclear excursion by drum runout during shutdown periods. The foregoing devices are further discussed in Sections 6.3 through 6.6. The safety implications of operating with less than twelve active control drums will be discussed in paragraph 8.2.8.

Since the same control console is used to operate the FCX as well as the PAX assemblies, a fool-proof patch-panel arrangement has been employed to ensure that the proper selection of neutronics, drum drive circuitry, and source path is made (refer to Section 6.2). A thorough checkout of this system has verified the logic of the design.

The operational status of the reactor control and safety systems is methodically and routinely evaluated in the daily checkout procedure. Furthermore the minimum instrumentation requirements for the scram circuit, (one log neutron level-period channel and two linear neutron channels) and the startup channel in the source region ensure that neutron flux level can be monitored over the full operating range and that a detector failure will not compromise the effectiveness of the safety system. By virtue of these design features and safeguards, the operator can assure himself that all systems necessary for safe operation are functioning properly before the reactor is brought critical.

The control system has been designed to limit the rate of positive reactivity insertion by allowing only a single drum to rotate outward at one time. This is achieved by providing individual pushbutton drive switches for each drum and employing a circuit whose logic prevents the manipulation of several pushbuttons from defeating this safety feature. Protection against a failure in this circuitry, such as an electrical short, is achieved by limiting the current that can be drawn in the drive motor circuit; a 50 percent overload of the current required to drive a single drum will trip a Simplytrol meter and scram the reactor. An additional restriction on the rate of reactivity insertion is provided by the gear reducer in the drive mechanism which mechanically limits the outward speed of the drum. On the basis of an integral drum worth of 1.0\$, a malfunction or failure in the control system could result in a power ramp whose total reactivity insertion is no larger than 1.0\$. In such an unlikely event, however, the safety system would terminate the excursion before the full 1.0\$ insertion would be realized. The fail-safe features in the scram circuit design (See Paragraph 6.3.2.2.2), the checkout procedure before daily operations, and the number of channels in the scram circuit (2 linear, 1 log, 1 period) ensure that the reactor safety system will always be effective for reactor protection.

It is concluded that the PAX control system is adequately designed against malfunctions that would produce a nuclear excursion. The most pessimistic conditions that can be postulated

for a malfunction of this type are a fast runout of a single drum followed by a power level trip at the 2 KW scram setting. The consequences of such an accident are analyzed in Paragraph 8.3.3.2.

8.2.4 Inadvertent Addition of Hydrogenous Material

As a result of the effect of moderating materials in an intermediate assembly as well as the spatial variation in neutron importance in the PAX core, the PAX reactor has a large average reactivity coefficient for hydrogen ($\sim 7\$/\text{kg}$ of H) with a peak value of approximately $20\$/\text{kg}$ at the center. Based on these values, it is estimated that a minimum of 163 grams of hydrogen concentrated at the center of the core would be required to override the minimum PAX shutdown margin of 3.25% ; the minimum volume associated with this mass for all hydrogenous liquids of interest is 1.46 liters (0.4 gallons). Extensive precautions have therefore been taken to prevent even relatively small quantities of liquids from being accidentally inserted into the core (refer to Paragraph 7.4.2).

As far as the facility itself is concerned, all water supply lines have been eliminated from the test cell; a drain line from the basement to grade level is provided with a check valve to prevent backup. The roof is constructed of 9 inches of reinforced concrete and the ventilator opening is adequately designed against water leakage. Numerous safeguards have been established to permit the use of equipment containing hydrogenous liquids in the test cell without compromising nuclear safety.

Approximately 5 gallons of oil are contained in the test cell crane for lubrication purposes. Despite the rigid construction of the crane assembly, a splash pan is provided beneath the gear box to prevent leakage of oil onto either of the cores. Procedures also ban parking the crane over either reactor when not in use and require that servicing be accomplished at the far northeast corner of the test cell.

A high voltage transformer, required for operation of the Pulse Neutron Generator, contains 60 gallons of oil in a sealed container, while a vacuum pump used during initial

pump-down of the unit holds approximately 5 quarts of oil in a completely enclosed housing. When these two components are in the test cell, they will be located so that the level of the oil is below the top of the reactor cores and all servicing, particularly oil changes, will be accomplished outside the test cell.

Other semi-permanent components and equipment containing small quantities of hydrogenous liquids (shock absorbers, turning stand, emergency lights) are located in the test cell and do not constitute a safety hazard per se. However, routine servicing of these items as well as other maintenance tasks (touch-up painting, drilling with cutting oil, cleaning with water or organic solvents) requires the introduction of limited quantities of hydrogenous materials in the test cell. The safety of these operations is assured by the requirement that no single container for test cell use may hold a quantity of liquid whose worth in the most reactive position in the core exceeds 1.0\$. For the PAX reactor, this restriction corresponds to a volume limit of 500 cc. Additional procedural safeguards to minimize the potential reactivity by effects of hydrogenous liquids in the test cell include a damp mop for cleaning and decontamination, a cover to prevent entry of extraneous materials in the core during shutdown periods, and CO₂ and dry chemicals which are used exclusively for fire fighting at WANEF. The site fire brigade, which has been indoctrinated in the special precautions that must be taken for WANEF emergencies, will handle all fires in the facility and outside aid will not be used (refer to Section 3.7).

The flood history of the Waltz Mill area and the topography of the site were discussed in Paragraph 2.3.3 in which it was concluded that WANEF is not vulnerable to natural flooding. Even if the threat of a major flood were to occur, there would be ample time to dismantle the core or poison it sufficiently to be subcritical under fully flooded conditions; the necessary quantity of poison wires for this purpose is stored in the test cell and hence immediately available if the need would arise. Seepage of subsurface water into the test cell basement is controlled by a sump pump which automatically removes any water that might accumulate in

the sump. Protection against pump failure is provided by two water detectors which activate an annunciator light on the reactor console and another at the site guard station.

When the use of liquids is required in the course of an experimental program, the Nuclear Safety Committee will review the container design, engineered safeguards, and procedures to ensure that the entrance of liquid in the PAX core is incredible. During these experiments, a metal cover secured to the pressure vessel flange or a metal plate resting on the surface of the core, will be employed to prevent the entry of extraneous materials into the core.

The design features and procedural safeguards described above, are considered adequate protection against inadvertent criticality resulting from the addition of hydrogenous material in the core.

8.2.5 Fuel Compaction

The fuel and tie rod coolant channels in the PAX core comprise a large fraction of the core volume and contribute significantly to neutron leakage. If it were possible to compact the fuel to fill these voids while maintaining the worth of the radial reflector, the reactivity would increase by approximately 20\$. However, the degree of compaction required to override the PAX shutdown margin is not possible to achieve in the WANEF facility for two reasons: (1) KIWI-model impact tests¹ demonstrated that forces equivalent to a 300 foot drop are required to produce significant compaction and (2) the structural design and support of the PAX core does not permit the reactor assembly to be moved from the test stand as a unit.

Handling of a fully assembled core at WANEF would be required in the course of a PTX experiment during installation of the reactor-in-vessel assembly on the test stand.

¹LAMS-2998, Radiation Transmission Measurements on Two KIWI-Model Impact Tests, January 1964.

Not only does the handling equipment have a minimum load safety factor of 2.8 but the maximum height to which the reactor assembly can be raised by the crane (40 ft.) is not great enough to cause significant core damage if it were dropped.

8.2.6 Neutron Reflection

Neither the beryllium reflector around the periphery of the core nor the graphite support blocks at the exit face of the reactor constitute an infinite neutron reflector; hence the presence of massive materials and/or equipment immediately adjacent to the PAX assembly can increase its nuclear reactivity by several dollars. The magnitude of this effect will, of course, depend on the neutron scattering properties of the extraneous reflector and the solid angle that this reflector subtends on the reactor assembly. An analysis of LASL and WANL experiments on NERVA-type reactors (Appendix A) has established that the maximum worth of the most effective reflector (3 inches of polyethylene) completely covering and resting on the end of the PAX core is 2.2\$ and that the most effective 360° radial reflector (3 inches of stainless steel) immediately adjacent to the pressure vessel has a value of 2.8\$. Precautions have therefore been taken to minimize the effects of neutron reflection and to ensure that a "side-by-side" or "end-on" experiment will not result in an uncontrolled reactivity insertion.

In a LASL experiment with a "paraffin man" mockup, it was determined that neutron reflection by a human body adjacent to the pressure vessel produces a maximum reactivity addition of 0.15\$. However, ready access to the exit face of the PAX assembly from the test stand as well as the design capability for removing the nozzle-end section of the pressure vessel creates a potential for a larger "fat man" effect. The surface area of a human body relative to the end of the core indicates that the reactivity worth of a "fat man" in contact with the end surface may have a magnitude in the order of 1.0\$. However, the control system, has a shutdown worth (3.25\$ minimum) which ensures that the reactor will be subcritical even for full radial or end-on reflection. Furthermore, access to the vicinity of the PAX assembly will be subject to the following restrictions when the shutdown margin is less than 10%; full body contact with the end of the core is prohibited, and no more than three persons are allowed

on the test stand in the immediate vicinity of the reactor. Allowing 15¢ per person and 3.25\$ shutdown for the isolated reactor, personnel reflection will not reduce the shutdown below the permissible limit of 2.8\$.

The upper limit on complete external reflection approaches a value of 5.0\$ and therefore the physical design of a side-by-side and/or an end-on experiment must ensure that neutron reflection does not override the control capability of the drums nor reduce shutdown reactivity below the minimum margin. This is accomplished by the standard operating procedure for manual reactivity changes which limits the effective increase in reactivity to 0.50\$ increments. Thus, even if the total reactivity addition is large, the installation of an experiment is performed in sufficiently small reactivity steps to keep the reduction in shutdown margin within acceptable limits at all times (see Paragraph 8.2.2.4). Unless the experiment is installed in direct contact with the reactor assembly, consideration must also be given to collapse of the experiment into a position of higher reactivity worth during reactor operation. No criticality hazard exists during shutdown conditions since the minimum shutdown margin (3.25\$) is greater than the worth of a full radial reflector or an end-on reflector. In the situation when an experiment involves combination of a radial and end-on reflector, each will be supported independently from the other to prevent simultaneous collapse of both reflectors by a single failure. As a safeguard against a reactivity insertion caused by failure of an extra-core experiment, a minimum load safety factor of 5 based on ultimate strength will be incorporated in the structural design of the experiment, while a safety factor of 8 will be employed for gears, pulleys cables, and parts subject to wear. Furthermore, when the failure of an experiment would insert more than 1.0\$ in reactivity, a position interlock will be included in the safety circuitry that will initiate a reactor scram if the experiment experiences any displacement during reactor operation. As discussed in Paragraph 8.2.4, additional safeguards to prevent inadvertent insertion of hydrogen in the core will be provided when hydrogenous liquids are involved.

It is concluded that the PAX assembly is adequately protected against the potential nuclear hazards of neutron reflection. The most probable excursion of significance that can be postulated for this class of accident results from the complete failure of the support structure for an end-on experiment while at delayed critical. The reactivity insertion produced by neutron reflection as the experiment falls on the core would initiate a power excursion that is terminated by the position interlock. An analysis of this accident is presented in Paragraph 8.3.3.4.

8.2.7 In-Core Experiments

There is a wide variation in the reactivity worth of extraneous materials within the PAX core which stems from the different nuclear properties of these materials and from the variety of locations where in-core experiments can be positioned. For example, hydrogen has an average (axial) reactivity coefficient of approximately $+7\$/\text{kg}$ while at the geometric center of the core, it has a worth of about $+20\$/\text{kg}$. On the other hand, the average (axial) worth of rhodium at the core centerline is $-0.35\$/\text{kg}$ and increases to $-0.60\$/\text{kg}$ at the periphery of the core. The physical design of an in-core experiment must therefore consider not only the reactivity associated with the configuration of interest but also the accidental displacement of the experiment into a more reactive position.

The operating procedures for manual reactivity changes will ensure that the installation of an in-core experiment will be accomplished in small reactivity steps and within the limit on minimum shutdown margin (See Paragraph 8.2.2.4). Furthermore, a requirement has been established that no in-core experiment will be installed which can, by axial displacement, compromise the minimum shutdown margin or produce a net reactivity change in excess of 1.0% ; inserts which are supported independently of one another are considered to be separate in-core experiments. For an in-core experiment having a positive reactivity coefficient, the reactivity change will be based on its displacement to the position of greatest neutron importance during its fall through the core. Except for rare instances when the axial

power distribution is radically distorted, a positive effect can result only when the experiment is installed above the axial centerline. Similarly, it will be assumed that an experiment consisting of a nuclear poison will relocate during its fall to the position of least neutron importance.

The design of experiments involving liquids will be based on the same general criteria and the detailed design and procedural safeguards will be reviewed by the Nuclear Safety Committee to ensure that no credible accident can result in a reactivity increase in excess of 1.0\$.

In the case of experiments containing low-melting point materials, their independent mode of support is protected by engineered safeguards which limit the core temperature rise produced by credible nuclear excursions below their melting point. Since the PAX experimental program includes in-core experiments of polyethylene whose melting point is approximately 120°C, the safety circuitry is designed to maintain internal heat generation from credible accidents below detectable levels (See Paragraph 8.3.3).

In the event that in-core experiments are based on other design criteria, e.g., engineered safeguards, material-behavior experiments, etc., the Nuclear Safety Committee will review the experiment design and procedures to ensure that failure can not produce a net reactivity increase in excess of 1.0\$.

In order to protect against failure of an in-core experiment, the design of a support structure for an in-core experiment will include a minimum load safety factor of 5 based on ultimate strength, and a factor of 8 for gears, pulleys, cables, and parts subject to wear. It should be noted here that the PAX experimental program involves reactor operation at room temperature conditions and hence a reactivity insertion produced by melting of low-melting-point materials can result only from a nuclear excursion initiated by some other mechanism.

Thus, no criticality hazard exists from collapse of an in-core experiment during a shutdown period and the effect of a failure during operation at delayed critical is limited by design to a 1.0\$ reactivity insertion and the resulting power transient would be terminated by the reactor safety system.

8.2.8 Control Drum Studies

The safety of the PAX experimental program depends heavily on the control of reactivity by the reactor control system and any modifications to the drums that could affect their worth will be accomplished with utmost precaution.

Analysis indicates that complete removal of the poison vane from a control drum would increase reactivity by 1.0\$ whereas removal from the reflector sector of a drum in the zero degree position would produce a net reactivity addition of 0.20\$. Although the magnitude of these effects does not present a safety hazard per se, the effectiveness of the reactor control system is assured by WANEF operating procedures which specify that only one control drum can be removed or altered at one time and that such a change will be accomplished in small reactivity steps (See Paragraph 8.2.3).

Rod drop measurements with a group of drums in different bank configurations have demonstrated that shutdown is not dependent upon configuration, but is determined solely by the total number of scrammable drums. Hence, no problems are created by operating with less than twelve control drums, providing the requirements on shutdown margin are satisfied.

8.2.9 Accidents of Non-Nuclear Origin

Fire

The WANEF test cell has fireproof construction throughout and procedures minimize the use or storage of combustible materials in the area. Furthermore, the fire door used for personnel access from the office section of the building isolates the two areas. Hence, the possibility of a fire within the test cell which either originates there or spreads from an adjacent area is considered very remote. Even if a fire should occur, it is not considered credible that damage would be sufficient to compromise nuclear safety.

Windstorm

Due to the structural design of the test cell roof and walls, no hazardous condition is expected to result as a consequence of windstorm.

Earthquake

Based on the seismological data presented in Section 2.5, probability of a major earthquake is negligible. Considerable force would be required to damage the reactor since it is contained in a pressure vessel which is bolted to the test stand. The test stand is bolted and welded directly to the building structural steel.

Riot and Strike

The probability of damage to the facility as a result of riot or strike is very low due to the remote location of the site, the small number of site personnel, the distance of the facility from public access, the security fence, and the 24-hour guard.

Sabotage

Explicit protection against deliberate and planned sabotage is not provided. Control over the presence and action of unauthorized personnel is maintained by procedures which require that all visitors to WANEF be accompanied by a member of the operating staff. After the facility is secured at the end of the day surveillance of the building includes frequent tours of the building by the security guard. During the working day operating procedures require the presence of no less than two persons in the test cell if the PAX cover is removed.

8.2.10 PAX-FCX Interaction

The presence of the PAX reactor in the same test cell as the FCX assembly will result in some degree of neutron interaction. LASL coupling experiments¹ with the PARKA and KIWI-TNT reactors reveal that interaction is significant only at close distances with both reactors critical and at the same power level; a reactivity effect of 0.033\$ was measured at a center-center separation of 16 feet while a 5 foot separation yielded a value of 0.25\$. No observable effect was noted with one of the assemblies shut down. Thus, neutron interaction between the PAX and FCX, which are separated by 19 feet center-center, does not present any safety problem.

¹ Private communication of unpublished data from C. Chezem (LASL).

8.3 HAZARDS ANALYSIS

8.3.1 General

A safety evaluation of the PAX assembly was presented above in which several credible accidents were identified. The radiological consequences of these accidents are analyzed below. Essentially the same approach used in the original Safety Analysis Report was employed with a few notable differences. Temperature feedback in the present analysis is based on a temperature coefficient obtained from the NRX test program which is 30 percent larger than the value previously assumed. The PAX control drum span is greater than was originally estimated in the early phases of the program and so the integral worth of a control drum has been conservatively increased from 0.65\$ to 1.0\$. Finally, safety studies in support of NRX testing have revealed that the reactor shutdown margin can be compromised by control vane melting and hence a new consideration has been established for PAX safety evaluation.

8.3.2 Radiological Hazards from a PAX Excursion

A nuclear excursion in a critical assembly creates a radiological hazard by virtue of the prompt neutron-gamma radiation emitted during the excursion and the fission product activity subsequently released to the atmosphere. The magnitude of the hazard depends on the total number of fissions produced, overall response of the reactor to the energy deposited in the core, and the particular environment in which the reactor is located.

The position of the PAX test stand and the shielding of the test cell walls provide a gamma attenuation of 10^4 in the operational area of the facility while fast neutron flux levels are reduced by a factor of 10^5 . At the side and rear of the facility where there is radiation streaming through door openings and where the shielding walls are not as thick, supplementary shielding is provided and personnel access is controlled by a security fence and door-gate interlocks. In this manner, the maximum radiation dose due to prompt radiation that can be received outside the exclusion area will not exceed 25 rem for the highest excursion energy obtained in the accident analysis (refer to Paragraph 8.3.3.2).

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Figure 8-1 Diffusion Characteristics of Various Fission Products
in NRX-A Fuel for 5 Minute Heating

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There are two characteristics of the PAX reactor that will influence the consequences of a nuclear excursion - the retention of fission products in NERVA-type fuel and the response of the poison vane in the control drum to (n, α) heating.

8.3.2.1 Fission Product Diffusion

The normal PAX fuel element consists of a dispersion of pyrographite coated UC_2 particles in a graphite matrix. The rate-limiting mechanism for the release of fission product activity from this fuel is the penetration of this coating by the various nuclides. Such a penetration can be accomplished by fission product diffusion which occurs at an elevated fuel temperature and by uranium migration whereby the fuel particle literally migrates through the protective coating and provides a direct path to the relatively porous graphite matrix. Uranium migration is dependent on the temperature history of the fuel and is not a factor for PAX operating conditions, nor is it a factor for accident conditions, as shown below.

The influence of temperature on the diffusion of fission products in NERVA fuel has been experimentally determined and is shown in Figure 8-1 for various nuclides of interest. It should be noted that there is a threshold temperature ($1500^\circ C$) below which there is essentially complete retention of activity while at approximately $2300^\circ C$, the effectiveness of the diffusion barrier is virtually lost. KIWI fuel is on hand at the WANEF facility and may be used in certain PAX experiments. These elements contain uncoated fuel particles and hence do not exhibit the same degree of fission product retention as the standard NERVA fuel. Diffusion experiments indicate that the release threshold for this type of material is approximately $1200^\circ C$. Based on a peak/average power ratio of 1.45, it is estimated that an energy release of approximately 1600 MW-sec is required to raise the central core temperature to this level.

8.3.2.2 Control Vane Melting

Reactor control is achieved by changing the orientation of the poison vane in the control drums. As the surface area of the vane between the core and reflector

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is increased, more leakage neutrons from the core are absorbed in the vane and reactivity is lowered as a result of the decreased effectiveness of the beryllium reflector. The poison vane itself is a boron-aluminum alloy whose physical properties are similar to that of aluminum. Neutron absorption is achieved from the (n,α) reaction in boron which is accompanied by heating due to the kinetic energy of the alpha particles. Since the nuclear worth of the poison vanes determines the shutdown of the PAX assembly, heat generation in the vanes may impose restrictions on the energy release that can be tolerated from a nuclear excursion analysis.

Using neutron flux profiles in the control drum region for a nominal 90° critical bank angle, the rate of heat generation in the vane was obtained and a contour of the region exposed to melting was determined. The loss of shutdown reactivity in the drums was then evaluated by importance-weighting the axial regions of the melt zone, based on the conservative assumption that the nuclear worth of any region which reaches the melting point is completely lost.

The result of this analysis indicated that a 1000 MW-Sec excursion will melt approximately 14 percent of the actual vane area (nuclear equivalent of about 30 percent) and decrease the reactor shutdown margin by 3.5\$ while a corresponding loss of about 1.5\$ was obtained for an 800 MW-Sec accident. The threshold for control vane melting occurs in the vicinity of 300-400 MW-Sec.

It is evident, therefore, that excessive control vane melting could cause secondary nuclear excursions and radiological hazards.

8.3.3 Accident Analysis

8.3.3.1 Excursion Model

The response of the PAX assembly to a reactivity insertion was obtained

with the RTS Code⁽¹⁾. This computer program solves the one-group, space-independent, reactor kinetics equation using a six-delayed-neutron-group model. The provision for representing the impressed reactivity by means of time-reactivity data points was utilized in describing the reactivity profile of the accidents analyzed. Representation of control drum scram profile was based on traces of actual PAX scram tests and the measured integral worth of the PAX control drums.

The kinetic parameters assumed for the PAX assembly in the excursion analyses are shown in Table 4-2. The temperature coefficient is the value derived from NRX testing at NRDS and hence is considered to be most reliable; this value represents a 30 percent increase over the coefficient previously assumed in the previous safety analyses.

The following expression is used in the RTS code to represent the influence of temperature feedback on reactivity:

$$\delta k(t) = \Delta k(t) - B \int_0^t n(t') dt'$$

where $\Delta k(t)$ is the impressed reactivity profile, B is the shutdown coefficient* in units of $\Delta k/\text{watt-sec}$, and the integral is the energy (watt-sec) generated in the core up to the time of interest.

(1) G. R. Keepin and C. W. Cox, Nuclear Science and Engineering, 8, 670 (1960).

* $B (\Delta k/\text{watt-sec}) = \frac{\text{Temperature Coefficient } (\Delta k/^{\circ}\text{C})}{\text{Core Heat Capacity} = \sum M_i (\text{grams}) \times C_{p_i} (\text{joules/gm-}^{\circ}\text{C})}$

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Figure 8-2. Influence of Scram Delay Time on PAX Energy Release
from Single Drum Runout (1\$ Step Insertion)

8.3.3.2 Drum Runout

A safety evaluation of the instrumentation and control system (Paragraph 8.2.3) has established that a single drum runout is the most serious credible accident that can occur as a result of a malfunction or failure in the hardware or circuitry. As a result of the redundancy that has been incorporated in the neutronics safety system, it was also concluded that the power excursion would be terminated by a reactor scram.

A conservative set of assumptions was selected to evaluate the consequences of such an accident. It was first assumed that the drum was located at the 0° position prior to the malfunction with the remaining drums at an adjusted bank position of approximately 100° . The full worth of the drum (1.0\$) was therefore included in the reactivity insertion. Despite the ability of the drum drive mechanism to limit the outward drum velocity, no credit was taken for this design feature and the reactivity insertion was represented as a step. Finally, the last trip to initiate a scram in the safety circuit (log power setting at 2 KW) was selected to terminate the excursion with only the eleven drums in the bank providing the shutdown reactivity.

The energy release produced by such a malfunction from an initial power of 1 watt is shown in Figure 8-2 as a function of scram delay time. For the purpose of this analysis, scram delay time is defined as that interval between the instant that the scram signal is initiated to the moment that the control drums begin their inward motion. A parametric study of scram velocity for the range $500\text{--}700^\circ/\text{sec}$ revealed that the variation of this parameter in the specified range does not have any influence on energy release; the PAX drums have a nominal scram velocity of approximately $600^\circ/\text{sec}$. It is evident that a scram delay time as long as 500 msec can be tolerated without exposing the PAX assembly to control vane melting. An upper limit of 200 msec has been established for the delay time of the neutronics safety circuitry in order to prevent damage to in-core experiments with a low melting point. From Figure 8-2, it can be seen that the energy release from a PAX drum runout is thereby limited to 5.0×10^4 watt-sec, leading to a maximum core temperature rise of 0.05°C based on a core heat capacity of 2.25×10^6 joules/ $^\circ\text{C}$. Hence, this malfunction will not produce any release of radioactivity to the atmosphere nor initiate any secondary nuclear excursions that might result from heating of the core, e.g., melting and redistribution of polyethylene wires to a more reactive configuration.

8.3.3.3 Failure of an In-Core Experiment

The design criteria and engineered safeguards for an in-core experiment were discussed in Paragraph 8.2.7. It was pointed out therein that axial displacement of the apparatus or material comprising the experiment would, by design, increase reactivity by no more than 1.0\$. The characteristics of this type of failure lead to an analytical model which is identical to the one developed above for the drum runout accident. Hence, the failure of an in-core experiment will also satisfy the restriction on energy release for protection against control vane meltdown and fission product release.

Any in-core experiment whose design is based on other criteria will be reviewed by the Nuclear Safety Committee to assure that an accidental energy release can not exceed the limits imposed in Paragraph 8.3.2.

8.3.3.4 Failure of End-On Experiment

The reactivity effect of locating a neutron reflector near the PAX core was discussed in Paragraph 8.2.1 and the safety implications of performing a side-by-side or end-on experiment was evaluated in Paragraph 8.2.6. It was noted that only when the experiment is installed at some distance from the surface of the PAX assembly, can this type of accident occur. Engineered safeguards that prevent the failure of a side-by-side or end-on experiment from creating a radiological hazard include rigid requirements on the structural support of the experiment and, in the case of experiments where a reactivity insertion in excess of 1.0\$ can occur, a position interlock will be provided that will produce a reactor scram at the instant that the experiment suffers any displacement from its installed position. The ability of the reactor safety system to terminate safely a 1.0\$ reactivity insertion was shown in Paragraph 8.3.3.2.

Based on the high worth of an axial reflector and the greater potential for an object to fall vertically than to move laterally, the failure of an end-on experiment is considered to be more probable and of greater consequence than a side-by-side experiment. The reactivity insertion that could result from failure of an end-on experiment which covers the entire end of the core varies with axial position as shown in Figure 8-3; it is assumed in this figure that the reactivity worth of the experiment at the point of installation has been compensated by the control drums and/or in-core shimming. Although the reactivity addition produced by such a failure increases with the height at which the experiment is installed, the rapid rise in the worth of a reflector as it approaches the surface of the core means that the reactivity inserted during the early stages of a failure is relatively small. The net effect is that the response requirements of the position interlock circuit will be determined by the failure mode of an experiment installed at some intermediate height. This is illustrated in Figure 8-4 where the energy release produced by failure of an end-on experiment from various heights is evaluated for a variety of scram delay times. Operating power level prior to the accident was chosen to be the maximum value of 1 KW. Operation at high power levels for a position interlock scram results in the maximum energy release. Upon failure of the support structure, it was assumed that the experiment fell with the acceleration of gravity directly in contact with the core. The reactivity insertion profile was determined from the position-time relationship of a free-falling body and the position-worth curve of a thick axial reflector. As used herein, scram delay time is defined as that interval between initial displacement of the experiment and the beginning of inward motion of the control drums. It is evident that a delay time of 225 msec is adequate to limit the excursion energy below the threshold of control vane melting for end-on experiments positioned at any height. For experiments installed at high elevations, the small reactivity change produced during the early stages of the failure effectively increases the allowable delay time, as previously discussed. At low elevations where the reactivity insertion is small, the power ramp proceeds at a sufficiently slow rate that a relatively large delay time is still effective in maintaining energy release within acceptable limits.

Figure 8-3. Maximum Reactivity Addition Resulting
from Failure of End-On Experiment from Various Axial Positions

Figure 8-4. Energy Release from Failure of End-On Experiment
Located at Various Heights Above Core

Figure 8-5. Influence of Delay Time in Position Interlock Circuit on Energy Release from Failure of End-On Experiment

As the delay time is decreased, the excursion is terminated more rapidly and the energy release is correspondingly lowered. It should be noted that, as the delay time is reduced, the height resulting in the maximum energy release shifts toward the face of the core where the rate of reactivity insertion from a falling experiment is greatest.

From the above analysis, the response characteristics of a position interlock circuit can be determined by plotting the peak energy release resulting from the collapse of an end-on experiment as a function of scram delay time as shown in Figure 8-5. The fact that energy release is very sensitive to delay time is most evident.

In order to satisfy the requirement that no significant core temperature rise be allowed to damage low melting in-core experiments, the delay time of the position interlock circuit will not exceed 150 msec. This value limits the energy release to a value of approximately 2×10^3 watt-sec (leading to an undetectable temperature rise) and provides a proper margin of safety from the region where energy release is most sensitive to delay time. The design of this circuit is described in Paragraph 6.3.2.2.4 and provides for a redundancy in the interlocks to ensure a reliable system.

8.4 CONCLUSIONS

The design features of the PAX assembly and the administrative controls and engineered safeguards that have been incorporated in the WANEF operating procedures are considered to be most adequate in preventing a criticality accident in the course of the NERVA experimental program, for example see below.

The consequences of those accidents which have been judged to be credible fall within the limitations imposed by control vane melting and the threshold for fission product diffusion. Operation of the PAX reactor within the constraints specified in this safety analysis can therefore be accomplished without undue risk to either operating personnel or the general public.

¹ WANL-TME-1296, Westinghouse Astronuclear Experimental Facility - Experimental Program - NERVA Contract Years 1966-1967, dated October 1965.

² WANL-TME-1490, Westinghouse Experimental Facility - Experimental Program - NERVA Contract Year 1967.

CHAPTER 9

FCX SAFETY ANALYSIS

9.1 GENERAL

The flexible critical reactor (FCX) has been designed to resolve immediate and long range NERVA problems which will be accomplished by a series of experiments on a variety of reactor configurations, see for example, WANL-TME-1296.^{1, 2}

Initial effort in the FCX program will be devoted to physics studies using NRX and KIWI fuel in bare or graphite reflected cores. Subsequent experiments will employ a beryllium reflector to approximate the NR-1 reactor as described in Section 5-5. The physics studies will be extended to include assemblies of FCX fuel elements whose fuel loadings will range from 200 mg U/cc to 500 mg U/cc, and whose void fractions may vary from zero to 0.325. The nuclear characteristics of these reactors including the NR-1 mockup have been previously described^{2, 3, 4} and do not include any features that can be considered as undue nuclear safety hazards. The flux-trap control rods (with their fuel followers) provide a minimum expected shut-down capability of 10\$, while the worth of any individual control element is sufficient to override the excess reactivity of the core. Initial FCX experiments on a graphite-reflected core of NRX-type fuel yielded a total bank worth of ~18\$, a little lower than the 20\$ predicted, as a result of interaction between rods. Nevertheless, adequacy of the control span in the FCX rods has now been experimentally verified.

As in the PAX assembly, a large increase in reactivity could result from the accidental insertion of hydrogenous material in the core. The administrative controls that have been employed during the PAX program to restrict such materials from the test cell have been demonstrated to be effective. Close attention will be given to the reactivity worth of extraneous

¹ WANL-TME-1296, WANEF Experimental Program - NERVA Contract Years 1966-1967

² WANL-TME-1490, WANEF Experimental Program - NERVA Contract Year 1967

³ WANL-TME-749, Analysis in Support of the Flexible Critical Experiment, April 1964

⁴ Section 5.6 of this report

neutron reflectors in view of the bare and minimally reflected configurations that are included in the FCX program. The worth of a thick water reflector completely surrounding the periphery of a bare core has been calculated to be approximately 11\$. The controls that will be imposed on the design of side-by-side experiments (paragraph 9.2.3) will ensure that such experiments can be installed without compromising reactor shutdown.

9.2 ANALYSIS OF CREDIBLE ACCIDENTS

By virtue of the operating limitation that excess reactivity of the FCX assembly will not exceed 0.50\$ and the attention that has been given to restricting operation within this limit, the FCX is not vulnerable to credible reactivity excursions of any consequence. The purpose of the safety analysis presented below is to demonstrate that inherent design features as well as procedural safeguards will ensure compliance with this restriction.

9.2.1 Manual Addition of Reactivity

Manual additions of reactivity are divided into several categories each with procedural safeguards commensurate with the size of the reactivity insertion involved. In all cases the changes must be made in such a way that no net reactivity step can be larger than +0.50\$, thereby insuring that the limit on excess reactivity is not compromised.

"Minimal" insertions are those which do not exceed +0.02\$. Such additions can be made directly (within the limitations on excess reactivity) and need not be followed by a reactivity measurement.

"Small" reactivity insertions are those which do not exceed + 0.20\$. Such additions can be made directly (within the limitations on excess reactivity) and must be followed by a reactivity measurement.

"Moderate" positive reactivity insertions are those greater than + 0.20\$ and not greater than +0.50\$. "Moderate" reactivity insertions must be preceded by the insertion of sufficient compensating negative reactivity (by supplementary poison of known worth or the removal of peripheral fuel or reflector clusters) such that after the addition of the positive reactivity, the value of k_{excess} will not exceed 0.20\$. The reactivity of the assembly will be measured after the positive insertion is made. This measurement will be in the form of a period determination or a delayed critical rod position. If k_{excess} is negative, a measurement of the count rate will be made on the startup channel as a guide to the next step.

"Large" positive reactivity insertions are those greater than 0.50\$ and not greater than + 5.00\$. "Large" reactivity insertions must be preceded by the insertion of sufficient compensating negative reactivity such that after the positive insertion is made, k_{excess} will not exceed -1.00\$. Count rate data will be taken on the startup channel with the control rods fully withdrawn before and after the "large" positive insertion is made. The inverse count rate data will be plotted as a function of poison removed (or clusters or positive shims added as the case may be).

Additional positive reactivity insertions will be made in "moderate" or "small" steps as defined above, each followed by count rate measurements with the control rods fully withdrawn. The extrapolation of the inverse count rate plot will provide a continual assessment of the excess reactivity of the core in the approach to critical.

The specific 1/M procedure that will be used during core loading and approach to criticality is described below. Restrictions on manual additions of reactivity subsequent to initial criticality are imposed more to protect the limit on excess reactivity than to prevent a criticality accident during the process of fuel loading since the FCX has a relatively large shutdown margin (10\$ minimum). After initial criticality has been achieved, reactivity coefficient measurements will be made in the core and reflector for materials of interest in order to make accurate estimates of the reactivity effect of subsequent reactor changes. Conservative reactivity coefficients obtained from graphite-moderated assemblies of similar composition will be used for known materials while conservative values based on polyethylene worth are employed for materials of unknown worth.

The same general operating regulations specified for the PAX assembly will be utilized during FCX manual reactivity changes to protect personnel against potential hazards.

These criteria are:

- 1) The experimental procedure is reviewed and approved by the Scientist-in-Charge and the Manager of Thermal and Nuclear Design.
- 2) The procedure is executed under the direction of an Experimentalist-in-Charge.

- 3) Two persons, one of whom is an Experimentalist-in-Charge, are present in the test cell when the reactivity of the assembly is being changed, a third person qualified as a reactor operator is on duty at the console and the test cell-to-console intercom is operational.
- 4) An audible indication of source range multiplication is provided for operations involving personnel in the test cell.
- 5) All control rods are locked in the 0 cm position (maximum shutdown) except as provided by checkout and reactor change procedures.
- 6) A CELL VETO is activated whenever personnel are making changes to the reactor. This interlock, which will override any attempt to start the reactor, has no bypass.

In addition, there are other operating restrictions and design features specific to each experimental procedure that will further enhance the safety of the experiment.

9.2.1.1 Core Assembly and Initial Criticality

By virtue of the flexible nature of the FCX assembly, the critical mass for a particular configuration will depend on fuel type, uranium loading, and reflector characteristics. Basic experimental knowledge of the nuclear properties of graphite moderated and graphite/beryllium reflected cores has been obtained from LASL-ORNL-UCRL experimental programs and extensive use of these results have been incorporated into the WANL analytical predictions.¹ This has led to a reasonable degree of confidence in analytical predictions. However, assembly of an FCX configuration in a safe and efficient manner will be accomplished by loading the reactor in increments and by taking frequent multiplication measurements that will permit an accurate prediction of criticality.

A source will be installed at the beginning of loading with a source-core-detector geometry that will ensure an adequate detector response and a well-behaved

¹ WANL-TME-749, Analysis in support of the Flexible Critical Experiment, April 1964.

plot of inverse count rate. Count rate data will be obtained after each loading increment from no less than 3 source-range detectors for 0, 50, and 100 percent control element withdrawal and a plot made of inverse count rate as a function of the total number of clusters loaded into the reactor. The initial increment of fuel clusters will not exceed 20 percent of the total complement of fuel and reflector clusters in the fully assembled reactor. Each succeeding increment will be either 20 percent of the total cluster loading in the fully assembled reactor or one-half of the remaining fuel/reflector clusters required for criticality as predicted by the inverse plot obtained with all control rods fully withdrawn, whichever is lesser. The final increment to be inserted will be specified by the Experimentalist-in-Charge such that an excess reactivity of 0.50\$ is not exceeded. A rod drop measurement will be performed after initial criticality has been achieved to verify the shutdown in the control rods.

The approach to criticality will therefore be accomplished in a controlled manner such that any anomaly in the reactivity of the assembly will be detected before a personnel hazard or a violation of the operating limit on excess reactivity can occur. The effectiveness of this procedure has been demonstrated during the first initial approach to criticality on FCX, April 1966.

In contrast to most FCX assemblies, the NR-1 mockup (Section 5.5) will be assembled in a manner similar to that used on PAX (see Paragraph 8.2.2.1). The beryllium reflector will be assembled first followed by the incremental loading of poisoned fuel clusters. Detailed procedures for core loading and initial criticality will be presented to the Nuclear Safety Committee for review as indicated in Section 5.5.

9.2.1.2 Reactor Changes

The FCX will be assembled in reactivity increments in accordance with the general change criteria described previously. Proper consideration of the radial variation in nuclear worth of reactor components will be possible through the use of region-dependent reactivity coefficients. The core assembly frame and lateral support bars will be employed as needed to provide support during such changes.

Changes affecting the control rods will also be accomplished in a manner that will ensure compliance with operating limits. The small excess in the FCX is divided almost equally between the seven control rods and hence the most significant effect of removing a rod from the assembly will be a reduction in shutdown margin. Since this operation also involves removal of a fuel follower from the core, there will be an accompanying decrease in excess reactivity equal to or less than the reactivity worth of a central fuel cluster. In the case of a $400 \text{ mg U/cm}^3 - 21.2\%$ void core, the central fuel cluster has an estimated worth of 0.45β and hence removal of the central control rod would virtually eliminate all the excess reactivity in the core. Additional reactivity will be inserted in accordance with Paragraph 9.2.1 to permit operation with less than seven rods; a minimum of four rods will be required consistent with a 10β minimum shutdown. Supplementary nuclear poison will be added as required to maintain the minimum shutdown margin of 10β while the rods are out of the core. To ensure that any modifications that might have been made to a rod will not compromise limits on excess and shutdown when it is returned thereto, the quantity of nuclear poison will be based on the assumption that the cadmium sleeve has been inadvertently omitted from each rod assembly. Such an error, which is conservatively estimated to increase reactivity by $\sim 2\beta$ per rod, will be assumed in determining the supplementary nuclear poison required to satisfy the criteria established in Paragraph 9.2.1.

9.2.1.3 Reactor Experiments *

Installation of experiments in or around the FCX assembly will be accomplished in accordance with the manual change criteria which will be satisfied under accidental reactivity insertions as well as normal conditions (see Paragraph 9.2.3 and 9.2.4). Proper consideration will be given to the radial and axial variations in the nuclear worth of experimental materials or equipment.

9.2.2 Inadvertent Addition of Hydrogenous Materials

Although the nuclear characteristics of the FCX will vary somewhat over the range of assemblies to be studied, the reactivity coefficient of hydrogenous materials in the core

* The integrity requirements on the design of these experiments are given in paragraph 7.4.6.

is large; the peak hydrogen reactivity coefficient will be approximately 20\$/kg. However, inadvertent insertion of the contents of an approved container (500 cc) for hydrogenous liquids can increase reactivity by no more than 1.0\$. The design features of the facility and test equipment, as well as the administrative controls restricting the use of such materials, have been previously described (Paragraph 8.2.4). Accidental criticality resulting from override of the FCX shutdown margin (10\$ minimum) by a liquid spill into the core during a reactor shutdown period is therefore considered to be incredible.

In addition to NERVA and UO_2 - type fuel which are not subject to hydrolysis, the FCX may also utilize KIWI fuel elements which contain uncoated UC_2 particles. Reaction of the UC_2 with atmospheric moisture has caused entrapment of released hydrogen in the graphite matrix and a concomitant increase in fuel reactivity. Extensive LASL experience with this fuel in critical assemblies has demonstrated that moisture pickup does not present an operational or a safety problem. Furthermore, it is anticipated that these KIWI elements have now reached equilibrium conditions and no further hydrolysis is expected. Since the increase in hydrogen content in the fuel is a gradual process, periodic reactivity checks will be adequate to determine any significant changes in reactivity due to this mechanism. Adjustments in core reactivity will be made if required to ensure that the 0.50\$ limit on excess reactivity will not be exceeded.

Location of the FCX at essentially grade level does not present a nuclear safety hazard. Evaluation of the vulnerability of the Waltz Mill site to flooding was discussed in Paragraph 2.3.3. The facility is considered safe from flooding by any regular flowing stream. Protection against floor flooding, as a result of run-off from the higher ground to the north during a severe storm is provided by a 3-foot interceptor trench that will divert the drainage. The area in the immediate vicinity of WANEF has also been contoured to ensure good drainage. The base of the FCX core is approximately 3-4 inches above floor level and 6-8 inches above grade level. Significant accumulation of water on the main floor is considered impossible due to the stairway opening and the numerous floor holes leading to the 22-foot deep basement. In the highly unlikely event of a major flood, there would be ample time to dis-mantle the core and store the fuel in a safe configuration.

Conduct of any FCX experiment involving liquids will require prior review by the Nuclear Safety Committee of the container design and of procedural safeguards to ensure that entrance of such materials in the core is incredible. Since the FCX is located in relatively close proximity to the PAX assembly, a similar review will also consider the implications of any PAX liquid experiment and its potential effect on the safety of the FCX.

9.2.3 Neutron Reflection

The FCX critical assembly is subject to an increase in reactivity by neutron reflection. The maximum worth of an end-on experiment will be comparable to the value of a thick axial reflector on the PAX reactor and is estimated to be in the order of 2.50\$. The type, thickness and azimuthal coverage of the radial reflector installed around the FCX core will determine the magnitude of the reactivity change that can be achieved from a side-by-side experiment. Its worth has been calculated to be as high as 11\$ for a bare core and decreases to approximately 9.5\$ for a configuration having a nominal graphite reflector (13.6 cm). Inherent design features of the FCX as well as procedural safeguards discussed below will ensure that neither the shutdown margin nor the limit on excess reactivity will be exceeded by neutron reflection.

The general criteria for manual reactivity changes described in paragraph 9.2.1 ensures that installation of any object, apparatus or structure next to an FCX assembly will be accomplished in safe reactivity increments. Furthermore, personnel access to the vicinity of the FCX is restricted such that the reactivity effect of body reflection will not exceed 2.50\$. By virtue of the 10\$ minimum shutdown margin which must be satisfied by an FCX configuration and the physical constraints imposed on control rod withdrawal when an entry interlock is bypassed, operating personnel are thereby protected against inadvertent criticality from excessive neutron reflection during shutdown periods.

A bound on the magnitude of an inadvertent reactivity insertion that can result from axial displacement of an end-on reflector or lateral motion of a side-by-side experiment will be achieved by installing only those experiments whose presence in direct contact with the reactor (actual or hypothetical) will not cause a violation of the 0.50\$ limit on excess

reactivity. Furthermore no single, independently supported side-by-side or end-on experiment whose failure would increase the excess reactivity beyond the 0.50\$ limit will be permitted.

9.2.4 In-Core Experiments

Although manual change procedures will ensure that installation of an in-core experiment will not violate the limits imposed on shutdown margin and excess reactivity, proper consideration will be given to the effects resulting from axial displacement of an experiment to a region of greater reactivity worth during reactor operation. The peak/average ratio of neutron importance along the axis of the core has a value of approximately 2.0 which indicates the magnitude of the reactivity change that could result therefrom. Such an event could conceivably occur by failure of the structural support for the experiment or, in the case of low melting point materials, by redistribution of a molten experiment following a nuclear excursion.

Protection against the latter has been achieved by designing the core with a small excess reactivity and providing a highly reliable reactor safety system such that there will not be any significant rise in core temperature during any credible malfunctions or failures (See Paragraph 9.3.1).

Protection of the limit on excess reactivity is achieved by the requirement that no in-core experiment will be installed which can, by axial displacement, exceed the positive reactivity increment $\Delta = (0.50\$ - \rho_{ex})$ where ρ_{ex} is the excess of the core with the experiment located at its normal position. Inserts which are supported independently of one another are considered to be separate in-core experiments. For an in-core experiment which has a positive reactivity coefficient, the reactivity change will be based on displacement of the material to the position of greatest neutron importance during its fall through the core. Similarly, it will be assumed that an experiment consisting of a nuclear poison will relocate during its fall to the position of least neutron importance.

As in the case of PAX, the design of experiments involving liquids will be based on this same general criteria and the detailed design and procedural safeguards will be reviewed by the Nuclear Safety Committee to ensure that no credible accident can result in a reactivity increase in excess of Δ .

In the event that in-core experiments are based on other design criteria, e.g., engineered safeguards, material behavior experiments, etc., the Nuclear Safety Committee will review the experimental design and procedures to ensure that the 0.50\$ limit on excess reactivity is not compromised.

9.2.5 Malfunction of Instrumentation and Control System

As discussed in Chapter 6, the design philosophy and safety features of the FCX instrumentation and control system are virtually identical to those provided for the PAX assembly. The same control console with its mechanical locks and electrical interlocks is utilized while neutronic behavior is monitored from the same type and number of instrumentation channels. Although described there in more detail, the most important safety features of the system are summarized below:

- 1) Personnel access interlocks are provided at strategic points within the WANEF facility thereby preventing reactor startup if potentially high radiation areas have not been properly cleared of personnel. If any one of these interlocks is bypassed through the master bypass to permit maintenance or checkout of the control system, control rod motion is automatically restricted by circuit design to a single rod.
- 2) The reactor control and safety systems are methodically and routinely checked in the daily checkout procedure.
- 3) The minimum instrumentation requirements for the scram circuit (one period-log power channel, and two linear neutron channels) and the start-up channel with the flux-up interlock ensure that neutron flux can be monitored over the full operating range and that a detector failure will not compromise the effectiveness of the safety system.

- 4) The control system is designed to limit the rate of positive reactivity insertion by allowing only a single control rod to be withdrawn at one time. Backup protection against a failure in this circuitry is achieved by limiting the current that can be drawn in the drive motor circuit. Additional control on the rate of reactivity insertion is provided by the gear arrangement in the drive motor.

Despite the integral worth of an FCX control rod which may be as large as 5.50\$, the 0.50\$ limit on excess reactivity will restrict the magnitude of the positive reactivity insertion above delayed critical that can be achieved by a failure in the rod drive system. When a rod configuration is employed whereby all but one control rod are fully withdrawn from the core, it is possible for the entire 0.50\$ to be inserted by a failure in the drive circuit of the controlling rod. The resulting power ramp would be terminated by the reactor safety system which includes period, linear power, and log power trip points. It should be noted that the insertion of any individual control element is sufficient to override the excess reactivity of the core.

9.2.6 PAX-FCX Interaction

The fact that neutron interaction between the FCX and PAX critical assemblies does not present a nuclear safety problem was discussed in Paragraph 8.2.10.

9.2.7 Accidents of Non-Nuclear Origin

The same procedural safeguards, facility design, and site characteristics described in Paragraph 8.2.9. which provide protection for the PAX assembly against fire, windstorm, earthquake, riot and strike, and sabotage are equally effective for the FCX.

9.3 ACCIDENT ANALYSIS

9.3.1 Maximum Credible Accident

By virtue of the operating limitation on excess reactivity and the design features and procedural safeguards that will ensure compliance with this limitation, the maximum credible accident of the FCX experimental program is a power ramp initiated by a 0.50\$ reactivity insertion and terminated by the reactor safety system. The mechanism by which this reactivity change is produced could be either a control rod withdrawal, failure of an in-core experiment or a combination thereof. During the subsequent power excursion, trip settings on reactor period, linear power, and log power would be activated in that order to quench the "accident" before any significant energy could be generated. The fail-safe features as well as the redundancies that have been incorporated in the scram circuit design ensure that the reactor safety system will always be effective.

In the early phases of an FCX power ramp following a 0.50\$ step insertion, reactor power will increase on a 5.5 second period. An upper limit on the delay time for the reactor safety system to terminate this excursion is estimated to be 35 seconds, which is the interval required for the power level to reach the 2 KW log power trip from low power (one watt). A delay time of this magnitude therefore implies simultaneous failures in the period and linear power trip circuits. The energy generated at this point would not have produced any detectable change in core temperature. Even a manual scram initiated 30 seconds after start of the ramp from an initial power level of 1 KW (maximum operating power) would limit peak power to 1 MW and energy release to 10 MW-sec; the peak temperature rise in the FCX core having the smallest heat capacity would be less than 50°C.

Hence, it is concluded that conduct of the FCX experimental program with an operating limitation of 0.50\$ on excess reactivity cannot present any undue hazard to the operating staff or the general public.

APPENDIX A

REACTIVITY EFFECTS OF MATERIALS ADJACENT TO PAX AND FCX

A.1 INTRODUCTION

A number of measurements have been made at WANEF and at LASL to determine the reactivity effects of materials placed side-by-side and at the end of the PAX and/or a KIWI B4 type reactor. These measurements included the effects of material accumulation as well as separation distance of the material from the reactor surface. A review of the available experimental data indicates that the worth of materials as a function of distance from the reactor surface can be expressed as a function of the square of the solid angle subtended by the material when viewed from the reactor surface. This solid angle is $2\pi(1-\cos\theta)$. The empirical function is

$$W_{\bar{X}} = W_o [1 - \cos \theta]^2 \quad (1)$$

where:

$$\begin{array}{ll}
 W_{\bar{X}} & \text{is the worth at distance } \bar{X} \text{ inches from the reactor surface} \\
 W_o & \text{is the experimentally determined worth at some point } 0 \leq \bar{X} \leq 4 \text{ inches} \\
 \theta = \tan^{-1} \frac{r}{\bar{X}}, & r \text{ being the effective radius of the material in inches.}
 \end{array}$$

Since equation (1) assumes the reactor surface to be an isotropic point source, it should be more valid as the distance of the material from the reactor surface increases.* The experimental data for polyethylene (Figure A-1) was used to normalize equation (1) in the region $\bar{X} = 0$ to 4 inches from the reactor surface. Equation (1) was then used for $\bar{X} > 4$ inches. As can be seen from Figures A-2, A-3, A-4, expression (1) fits the experimental data extremely well for graphite, aluminum and stainless steel.

* Up to 4 inches from the reactor surface equation (1) overestimated the worth by about 30 percent.

The experimental and empirically determined data as well as the calculations presented in the following sections are to be used as a guide in planning future side-by-side and/or end experiments using both the PAX and FCX reactors.

Since the PAX and the presently planned FCX reactors have the same core composition, the total axial leakage will be approximately the same. Thus equivalent thickness of materials above or below PAX and/or FCX will have approximately the same reactivity effects.

The worth of materials around (radial) FCX will be greater than around PAX mainly because of the presence of poison in the PAX reflector.

The material worths around a bare and reflected FCX will be approximately the same, provided the bare and reflected reactor have the same radial dimensions, since the graphite making up the FCX reflector has the same properties as the core graphite.

A.2 REACTIVITY EFFECTS ON PAX

A.2.1 General Analysis of Reactivity Effects of Polyethylene

1) Axial Effects

The reactivity contribution of a polyethylene disc 35 inches in diameter, added to the top end of the core, was measured¹ as a function of its thickness. The results are shown in Figure A5. The first 2 1/2 inches is seen to contribute 94 percent of infinitely thick polyethylene. The total weight of polyethylene in the experiment accumulated at the top of the core was 87 kg (74.5 kg C and 12.5 kg H). From Figure A5 we can deduce the differential worth of solid polyethylene ($\rho = 0.92 \text{ gm/cc}$) as a function of accumulation. This value ranges from approximately 6.5¢/kg at the top of the core to zero worth six inches away.

The effect of a solid block of polyethylene, 35 inches in diameter by 6 inches thick, approaching the top end of the core was also measured¹ at various distances from the core. Figure A1 shows that the variation is nearly linear from 2.2\$ in contact with the core to 1\$ four inches away. This variation can be approximated by the

¹ LASL-N-2-7162, October-November 1961

empirical function $W_{\bar{x}} = W_o [1 - \cos \theta]^2$ discussed in the introduction. This function was then used to approximate the reactivity effect at larger distances. As can be seen from Figure A-1 at a distance of fifteen inches, the worth goes down by a factor of ten.

Additional experiments with polyethylene as a function of distance from the top end of the core were performed at WANEF with the PAX core as well as at LASL with a KIWI B4 mockup. The measurements performed at WANEF² utilized polyethylene in the form of crumpled sheets, whereas the LASL measurements³ utilized thin "solid" polyethylene discs. The effective polyethylene density in the WANEF experiments was approximately 0.074 gm/cc versus a density of 0.92 gm/cc in the LASL experiments.

All these data are plotted in Figure A-6, together with the predicted variation based on $W_{\bar{x}} = W_o [1 - \cos \theta]^2$. It is concluded that Figure A-6 is valid for polyethylene masses varying from 0.5 to about 10 kg. It should be noted that the worth at the top of the core is 6.5¢/kg in agreement with the differential value derived from Figure A-5.

2) Radial Effects

The reactivity contribution of a solid polyethylene shell 52 inches high wrapped around the pressure vessel of a KIWI B4 type reactor was measured⁴ as a function of its thickness. The results are shown in Figure A-7. The first inch is seen to contribute 73 percent of infinitely radially thick polyethylene, while 3 inches seems to be equivalent to an infinite thickness giving a worth of 2.75\$. From Figure A-7 the differential worth of polyethylene at the pressure vessel (based on a thickness of 0.2 inches or a weight of 24.5 kg) is 2.45¢/kg. The worth of a solid shell of polyethylene (24.5 kg) as a function of distance from the pressure vessel is shown in Figure A-8. The shape of this curve is also valid for thicker shells (the actual worth of any size shell as a func-

² WANL-TME-1255, PAX Experiments in Support of NERVA Quarterly Report, 4th Quarter, CY65

³ LASL-N-2-7558, August-September 1962

⁴ LASL-RFS-135, Rift Safety Review

tion of distance can be calculated by differentiating Figure A-7). It should be noted that figure has a different slope in comparison to Figure A-8. This is due to a larger solid angle subtended by the curved surface. From Figure A-8 we see that for the worth to go down by a factor of ten, the material has to be approximately 25 inches away.

A.2.2 Reactivity Effects of Non-Hydrogenous Material

1) Graphite

a) Axial

Measurements of the worth of graphite discs added to the top end of a KIWI B4A mockup were made at LASL⁵. The graphite discs varied in weight between 10 and 65 kg. Interposed between the graphite discs were varying amounts of inconel (up to 50 kg) and polyethylene (up to 4 kg). All the graphite data are plotted in figure A2, together with the predicted variation. Figure A2 is approximately valid for graphite weights up to 70 kg. Above 70 kg the mass effect becomes important and the worth of graphite per kg decreases.

b) Radial

The worth of graphite around PAX as a function of its thickness is shown in Figure A-9. Figure A-9 is based on a calculation which has been normalized to a calculation/experiment involving polyethylene. The worth of infinite graphite as a function of radial distance from the pressure vessel is plotted in Figure A-10.

2) Aluminum

a) Axial

The worth of aluminum added to the top end of PAX was measured at WANEF.⁶ Plates of aluminum varying between 10 and 120 kg were used. Interposed between the aluminum plates were crumpled sheets of polyethylene ($\rho = 0.074$ gm/cc). All the aluminum data are plotted in Figure A together with the predicted variation.

⁵ LASL-N-2-7895

⁶ WANL-TME-1255, PAX Experiments in Support of NERVA Quarterly Report, 4th Quarter, CY65

b) Radial

The worth of aluminum around PAX as a function of its thickness is shown in Figure A-11. Figure A-11 is based on a calculation which has been normalized to a calculation/experiment involving polyethylene. The validity of this curve is further substantiated from a measurement of the worth of the PAX pressure vessel⁷. The aluminum pressure vessel, with a thickness of 0.70 inches, has a measured worth of about 32¢ without the dome end. The worth of infinite aluminum as a function of radial distance from the pressure vessel can be derived using Figure A-10 as a guide.

3) Stainless Steel

a) Axial

The worths of inconel and stainless steel added to the top end of PAX and a KIWI B4A mockup are plotted in Figure A-4 which is approximately valid for masses up to 100 kg.

b) Radial

The worth of stainless steel around PAX as a function of its thickness is shown in Figure A-12. Figure A-12 is based on a calculation which has been normalized to a calculation/experiment involving polyethylene. The worth of infinite stainless steel as a function of radial distance from the pressure vessel is plotted in Figure A-13.

The worth of the Privy Roof Mounted Shield (PRMS) was measured at WANEF at approximately 27 inches from the pressure vessel. The measured worth was 13.6¢. This worth is due to approximately 3500 kg of steel in an 135° arc, 52 inches high and 27 inches away from the pressure vessel. Figure A-14 shows the effect on core reactivity in the case of the PRMS moving toward the pressure vessel surface. The worth of the PRMS adjacent to the pressure vessel is 74¢. This value can be compared to the worth derived from Figure A-14. Since the worth of infinite stainless-steel around PAX is approximately

⁷ WANL-TME-1053, PAX Critical Experiments in Support of NRX-A3

2.7\$, the worth around a 135° arc is $\frac{135^\circ}{360^\circ} \times 2.7\$ = 1.0\$$, in good agreement with the 74¢, considering the voids inside the PRMS and the approximation involved in deriving the value of 1.0\$.

4) Miscellaneous

Table A-1 presents some reactivity worth measurements performed at LASL. Figures A-15 and A-16 show the worths of integral quantities of material versus axial separation. The effects of accumulation should be considered when dealing with heavier masses. Figure A-15 is only valid up to 15 kg.

A.3 REACTIVITY EFFECTS OF FCX

The presently designed FCX contains a mixture of LASL, Y-12, and NERVA (Nb coated and uncoated) type of fuel. The critical dimensions for a bare and reflected core as well as the worth of materials around them were calculated using NSDM-II,⁸ The total material inventory for both the bare and reflected cores is given in Table A-2. The results of the worth calculations are presented in Figures A-17 and A-18, where the reactivity worth is given as a function of material added around the core. It is seen that: the maximum worth of any material placed around (radially) a bare FCX core with a radius of 83.2 cm is approximately 11.0\$, the maximum worth of any material placed around (radially) an FCX core of 73.8 cm reflected with 13.6 cm of graphite is approximately 9.50\$ and the maximum worth of any material placed at the axial end of an FCX core will be approximately similar to that of PAX. The worth of materials as a function of separation from the FCX reactor surface is approximately the same (percent wise) as that of PAX.

⁸ WANL-TME-1091, Improved Nuclear Standard Design Method for NERVA Calculation - NSDM-II, March 1965

TABLE A 1
LASL REACTIVITY WORTH MEASUREMENTS

Type of Experiment	Reference LASL	Configuration	Worth
Interaction between 2-reactor system	N-2-7932		$r = 6 \text{ feet}$ $P = 27\%$
Aluminum support plate above core	N-2-7895 May 1964 June		$Al = +0.22\%/kg$
Worth of beryllium as a function of core radius	N-2-7612 October 1962 November		<ol style="list-style-type: none"> 1. $r = 18 \text{ inches}$ (adjacent to core) $5.2\%/kg$ 2. $r = 20.2 \text{ inches}$ (between core and Be reflector) $3.5\%/kg$ 3. $r = 25 \text{ inches}$ (outside Be reflector) $0.9\%/kg$
Worth of Shield	N-2-7590 September 1962 October		$+0.6\%$ with shield with or without the H_2 mockup

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TABLE A2
 FCX MATERIAL INVENTORY, kg

<u>Core Material</u>	<u>Bare FCX</u> [*]	<u>Reflected FCX</u> ^{**}
Uranium	604.9	364.8
Carbon	3022.7	1817.8
Niobium	221.5	133.3
 <u>Reflector Material</u>		
Carbon	---	982.5

^{*}Critical Core Radius = 83.25 cm
^{**}Critical Core Radius (fuel + graphite reflector) = 73.77 cm

Figure A-1. Polyethylene Reactivity Worth as a Function of Axial Distance from PAX Core

Figure A-2. Graphite Reactivity Worth as a Function of Axial Distance from PAX Core

Figure A-3. Aluminum Reactivity Worth as a Function of Axial Distance from PAX Core

Figure A-4. Stainless Steel Reactivity Worth as a Function of Axial Distance from PAX Core

Figure A-5. Polyethylene Reactivity Worth versus Thickness when in Contact with PAX Core Top

Figure A-6. Polyethylene Reactivity Worth as a Function of Distance from PAX Core Top

Figure A-7. Polyethylene Reactivity Worth when Wrapped Around Pressure Vessel, KIWI-B4 Type

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Figure A-8. Polyethylene Cylinder Reactivity Worth as a Function of Distance, PAX

Figure A-9. Graphite Reactivity Worth as a Function of Radial Thickness, PAX

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Figure A-10. Graphite Cylinder Reactivity Worth as a Function of Distance from PAX Pressure Vessel

Figure A-11. Aluminum Reactivity Worth as a Function of Radial Thickness, PAX

Figure A-12. Stainless Steel Reactivity Worth as a Function of Radial Thickness, PAX

Figure A-13. Stainless Steel Cylinder Reactivity Worth as a Function of Distance from PAX Pressure Vessel

Figure A-14. PRMS Effect on PAX Reactivity as a Function of Radial Distance

Figure A-15. Polyethylene Reactivity Worths of Specific Quantities as a Function of Axial Distance from PAX Core Top

Figure A-16. Reactivity Worths of Various Materials as Functions of Axial Distance from PAX Core Top

Figure A-17. Reactivity Worths of Materials Added Around a Reflected FCX Reactor

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Figure A-18. Reactivity Worths of Materials Added Around a Bare FCX Reactor