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MODIFICATION OF THE FORD NUCLEAR REACTOR FOR 10 MEGAWATT OPERATION

Volume I

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MODIFICATION

OF THE

FORD NUCLEAR REACTOR

FOR

10 MEGAWATT OPERATION

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ABSTRACT

The modifications to the Ford Nuclear Reactor (FNR) Facility necessary to operate that reactor at a steady state power level of 10 Megawatts are described. The changes needed are outlined in detail, where possible, and the areas where further engineering or developmental work is required are identified and discussed. Based on the design changes proposed, a Design Basis Accident (DBA) is described and the consequences of the DBA analyzed. The report concludes that the location of the FNR on the North Campus of The University of Michigan will, with the recommended design changes, satisfy the site criteria outlined in 10CFR100. It is recommended that a prompt determination be made of the extent to which this facility will have to conform to the standards for tornado and earthquake resistance and to perform any necessary structural analyses indicated. The results of these determinations will then dictate whether or not additional final design efforts are warranted.

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1. INTRODUCTION

1.1 General Background

Since the Ford Nuclear Reactor attained criticality in September, 1957, its operational program and experimental facilities have expanded to meet the needs of faculty research programs; in particular, experimenters operating neutron spectrometers. The following tabulation reviews the operating history of the FNR:

1957:	Initial criticality: operation at 100 Kw on a one-shift basis, 5 days per week
1958:	Power increased from 100 Kw to 1 Mw
1961:	Two-shift operation begun
1963:	Three-shift operation 5 days per week adopted, resulting in continuous operation of the reactor in excess of 100 hrs. per week
Aug. 1963:	Operating power level increased to 2 Mw
Feb. 1965:	A 45-gallon heavy water reflector tank installed next to the reactor core, converting two of the reactor's eight beam ports to tangential facilities and, in general, increasing the available thermal neutron flux for all reactor beam ports
Jan. 1966:	Continuous operation begun. Reactor operates on a 28-day cycle at varying power levels, followed by a 3-day maintenance shutdown

This continuous operating cycle has been maintained for eight years, and has resulted in the reactor having an over-all On-Line Factor of approximately 88%. The facility is operating at its practical limit as regards operating time and, because of the capacity of the present heat removal system, at its maximum power level as well.

In order to increase the utility of the reactor to the experimental groups using it, consideration must be given to a power level increase.

A survey of the current users of the FNR show that two major research areas would benefit from a power level increase: Neutron Spectroscopy and Activation Analysis. However, benefits would also be afforded to the research efforts of several other departments.

This report summarizes the facility design modifications considered necessary to permit operation of the FNR at 10 Megawatts using uranium-aluminum alloy, MTR-type fuel.

1.2 Other Designs Considered

Before deciding upon this fuel type (MTR) and the associated facility design changes outlined in this Report, the possibility of using other fuel element designs was investigated.

Three types of fuel are commercially available for use in a pool reactor facility like the FNR. These are:

MTR - type of fuel presently in use at the FNR.

TRIGA - a fuel which permits pulse operation and is in use at a number of university reactor facilities. The fuel has an inherent negative temperature coefficient achieved by mixing moderator (zirconium hydride) and fuel in the element.

PULSTAR - a fuel which permits pulse operation and has negative temperature coefficient based on doppler coefficient of low enrichment uranium dioxide fuel. This fuel is similar to the majority of power reactor fuels.

Approximately 30 fuel element subassemblies are required for a reactor core which uses any of these three fuel forms. The following paragraphs review the basis of the decision to use MTR fuel.

1.2.1 MTR

The MTR type fuel assembly consists of 18 fuel bearing plates fastened to side plates. The assembly dimensions are 3" x 3" x 24" long. Each fuel plate consists of a highly enriched uranium-aluminum alloy core clad with high purity aluminum approximately 3" x 24" x .060" thick.

Element assemblies with a fuel content of up to 200 grams of U-235 are commercially available from fuel element fabricators. This type of element is used in the MTR, ORR, HFBR, and the majority of the megawatt level pool-type reactors. More than 250,000 megawatt days of reactor operating experience have been accumulated with this type fuel. This fuel is not used in research reactors which operate in the pulse mode. The SPERT and BORAX tests showed that energy pulses of 10-20 Mw-sec. in cores of this fuel produced permanent fuel element deformation. Cores using TRIGA and PULSTAR elements are routinely pulsed to 40-50 Mw-sec. energy releases. Therefore, use of MTR type fuel is limited to steady-state operation.

The advantages of this type of element include:

1. Of the three fuel element types to be discussed, the MTR fuel element has the largest heat transfer surface area.

2. The element design achieves the best thermal conductivity between the fuel meat and cladding of the three element types.

3. Because of its long history of use in several of the high power research reactors, considerable engineering and core physics information is available for this fuel type.

4. Of the three fuel types considered, the unit element fabrication cost is the lowest for the MTR fuel (approximately $1,400 per element).

The disadvantages include:

1. The element cladding and core has a relatively low melting point (approximately 1000°F) compared to the other fuel forms.

2. Self-shutdown of the reactor during a transient relies on the moderator (coolant) negative temperature coefficient. The SPERT and BORAX experiments have demonstrated the inherent shutdown capability of this type of fuel.

3. This fuel element in a FNR-type operation has a useful life of approximately 70 MWD per element.

1.2.2 TRIGA

Gulf General Atomic (GGA) provides a TRIGA type fuel assembly for use in reactors originally designed for MTR-type fuel. The assembly consists of four TRIGA fuel rods attached to end fittings which are compatible with the existing reactor grid plate. Each TRIGA fuel rod consists of a mixture of zirconium dihydride and uranium (20% enriched) in the following weight proportions.

Uranium = 8%
Zirconium = 81%
Hydrogen = 1%

The effective zirconium hydride compound formed is $ZrH_{1.7}$. This compound is used in preference to ZrH_2 which tends to disassociate more readily to form free H_2 at elevated temperatures, and pressurize the interior of the fuel rods.

The fuel region of the TRIGA fuel is a 1.4 inch diameter by 15 inches long rod with a 4-inch long graphite reflector rod at each end. Each fuel rod is clad with .020 inch thick stainless steel tubing with welded end caps. Four of these rods are in each fuel assembly.

The advantages of the TRIGA include:

1. Demonstrated pulsing capability.

2. High melting point of cladding and fuel.

3. GGA maintains an engineering staff which works on current problems and, using its own operating facilities, carries on evaluations programs.

The disadvantages of the TRIGA fuel include:

1. Present power level experience limited to 1.5 Mw.

2. High fuel element cost - approximately $5,000/ element.

3. Fuel element life is approximately 55 MWD/ element. (An element design with much longer life is now being evaluated but little information is currently available.)

1.2.3 PULSTAR

AMF Atomics (AMF) also provides a conversion fuel assembly for MTR-type reactors. The assembly consists of 25 zircaloy-2 tubes approximately 0.5 inches in diameter and 24 inches long loaded with UO_2 pellets (enriched to 4-6% in U-235 isotope). The 25 tubes are assembled in a 5 x 5 array within a zircaloy-2 box similar in outside dimensions to an MTR-type fuel element. The ends of the box are provided with fittings compatible with the existing reactor grid plate.

The advantages of the PULSTAR fuel include:

1. Elements are of power reactor design and may offer certain educational advantages.

2. High melting point fuel (5000°F) and clad (3300°F).

3. Long core life (190 MWD/element).

The disadvantages of PULSTAR fuel include:

1. Present power experience limited to 2 Mw in research reactor applications.

2. High fuel element cost - approximately $3,500/element.

3. AMF has limited engineering staff for support and does not perform its own operating evaluations.

1.3 COMPARISON

The general characteristics of the three commercially available fuel types which can be used in the FNR are summarized on Table I on the next page. The TRIGA fuel is the most expensive of the fuel types with regard to both first cost as well as the annual element replacement cost. The cost of the PULSTAR and MTR fuels appear to be competitive except as regards the initial core cost. This, however, is based on fabrication costs only. A more detailed analysis of the total fuel cycle cost (including shipping, spent fuel reprocessing, and inventory charges) shows the annual fuel cycle cost for PULSTAR to be approximately $28,000 less than the MTR annual fuel cycle costs at the 10 Mw power level. Some additional comments are included below:

1. The MTR fuel is the only one with which research reactor operating experience has been gained at powers in excess of 2 Mw.

2. While all three fuel types were designed to be installed in a research reactor like the FNR, the effect of operation with TRIGA and PULSTAR fuels on the usefulness of the Heavy Water Reflector Tank used in the FNR is not known and must be evaluated.

3. As regards the thermal neutron fluxes available in the in-core experimental facilities, the three fuel types have all shown a maximum flux of 2 - 3 x 10^{13} nv/Mw for 30 element cores.

While the availability and cost of these fuel types are important to any final

TABLE I-I

BRIEF SUMMARY OF FUEL ELEMENT CHARACTERISTICS

(NOTE: 10 Mw Core Would Require Approx. 30 Elements Of Each Type)

ITEM	MTR TYPE	TRIGA	PULSTAR
General Information			
Fuel Mass	200 gms	155 gms	756 gms
Enrichment	93%	20%	6%
Active Fuel Length	24 in.	14 in.	24 in.
Minimum Flow Area	3.14 $in.^2$	1.76 $in.^2$	2.11 $in.^2$
Heat Transfer Area	2270 $in.^2$	282 $in.^2$	885 $in.^2$
Element Life	70 MWD (200 GM)	55 MWD	190 MWD
Element Fabrication Cost	\$ 1,400	\$ 5,000	\$ 3,500
Initial Core Cost	\$42,000	\$150,000	\$105,000
Annual Fuel Element Cost (Fabrication Only)	\$52,000	\$332,000	\$ 54,000
Maximum Power Level Experience To Date	40 Mw	1.5 Mw	2 Mw
Over-all Fuel Cycle Costs			
Inventory Charges	\$ 8,700	\$ 7,400	\$ 16,000
Burn-up Charges	\$ 41,000	\$ 39,600	\$ 43,400
Shipping Charges	\$ 14,000	\$ 18,000	\$ 3,500
Reprocessing Charges	\$ 37,000	Unknown	\$ 7,900 (Est. based on batching)
Fabrication Cost	\$ 52,000	\$332,000	\$ 54,000
	\$152,700	\$397,000 + Reprocessing	\$124,800

selection, the maximum power level which each type can achieve in the FNR is also important.

Figure 1 shows the primary coolant pressure drop as a function of primary coolant rate through cores containing 30 elements of each of the three fuel types. The figure also shows the maximum power level believed possible at that coolant rate. Since the maximum pool water head available to force coolant through a core in the FNR is twenty feet, the maximum flow rates and power levels for the three core types were estimated to be:

TYPE	MAXIMUM FLOW	MAXIMUM POWER
TRIGA	3000 GPM	3.3 Mw
PULSTAR	4250 GPM	6.4 Mw
MTR	5000 GPM*	10.0 Mw

* pressure drop = 13 ft.

In the case of the MTR-type fuel, an arbitrary power level limit of 10 Mw has been imposed since higher power levels would remove the FNR from the "Research Reactor" classification as defined in Title 10 of the Code of Federal Regulations. Moreover, as this Report will show, 10 Mw appears to be a practical upper limit.

The information on which Figure 1 is based was obtained either from operating reactors or from the reactor manufacturer.

Considering all of the points which were summarized above, it was felt that it would be most practical to perform a feasibility and design study which concentrated on the continued use of MTR-type fuel. This was further supported by a general lack of interest, at the FNR, in a reactor with pulsing capability.

This report summarizes the design work done to date on this project and demonstrates that operation of the FNR at 10 Mw can be achieved without excessive capital cost while maintaining a suitably high degree of safety for personnel as well as the general public.

2. REACTOR SITE

2.1 Location

The FNR and the Phoenix Memorial Laboratory (PML) are located on the North Campus of The University of Michigan. The North Campus is a tract of 900 acres located 1-1/4 miles from the Central Campus and business districts of the city of Ann Arbor. Some general features of the location are shown on Figure 2.

2.2 Population Density - General Area

As can be seen in Figure 2, the University controls all the land within 1500 feet of the reactor. Significant population densities not under University control are not encountered until 4000 feet from the site. The only exceptions to this are the Veteran's Administration Hospital (~ 1500 feet - 700 people), the Village Green Apartments (~ 3000 feet - 1000 people), and the Huron Towers Apartments (~ 2500 feet - 1400 people).

As regards housing accommodations on the North Campus which are under University control, the present population figures are as follows:

Baits Housing	800 adults
Northwood Apartments	2200 adults + 1100 children
Bursley Hall	1200 adults

Land development in the general area consists primarily of the expansion of University facilities on the North Campus and private home construction to the east of the site. The present zoning of the residential regions is for less than 10 dwelling units per net acre. Assuming four persons per family, the population density to the area east of the site will be approximately 40 persons per acre.

As can also be seen from Figure 2, the prevailing wind directions seldom carry any facility stack exhausts toward the highly populated areas of the city of Ann Arbor which lies to the southwest of the reactor site.

The city of Ann Arbor has a total population of approximately 100,000 of which approximately 30,000 are students at the University.

2.3 Population Density - Reactor Site Area

This Report concludes in Section 15 "Safety Analysis" that the immediate vicinity (1000 foot radius) of the reactor site satisfies the criterion established in 10CFR100. That being the case, the population density of the immediate area of the site will be reviewed in some detail.

Figure 3 shows the location of the existing North Campus buildings and their average occupancy for distances up to 1000 feet from the Reactor Building. The closest buildings are the Automotive Laboratory, the Research Administration Building, and the Cooley Laboratory. At the present time, these buildings house approximately 450 people during normal working hours.

All of the buildings within the 1000 foot radius are office, laboratory, conference, and/or classroom facilities with the exception of the North Campus Commons. The Commons Building is a food service area providing cafeteria food service (1 meal/day) for students and staff on a five day per week basis with weekend service available.

There are no facilities with sleeping accomodations within the 1000 foot radius. The closest living quarters are the Northwood Apartments located approximately 1500 feet to the north of the facility.

As regards the estimated future population of the North Campus site, Figure 4 shows the proposed expansion of University facilities. The buildings labelled I, II, III, IV, and V represent the proposed construction of facilities to house The College of Engineering (reference 1). The funds required to initiate the planning for these facilities were requested for inclusion in the 1974-75 operating budget of the University. It is estimated that the first building could not be completed before 1978 with all the buildings, under optimum scheduling, completed by 1983. Thus, an expansion of the North Campus facilities to achieve that shown in Figure 4 will probably involve 8-12 years.

The other facilities schematically indicated on Figure 4 have not as yet been put into the planning stage. Therefore, it is believed that those facilities will not come into being in less than 10-15 years.

2.4 Site Topography

The topography in the vicinity of the reactor site is that of level to gently rolling land. Table 2-1 is a list of elevations of the land on which buildings in the area of the site have been constructed (see Figure 2 for locations).

TABLE 2-I

SITE ELEVATIONS

Radius	Description	Elevation	Elevation Relative to FNR
0–2000'	FNR and PML	~ 845	0
	Music Bldg.	~ 850	+ 5
	Open land	~ 858	+ 13
	Bursley Hall	~ 900	+ 55
	Northwood Apts.*	~ 910	+ 65
	Space Research	~ 900	+ 55
	VA Hospital	~ 814	- 31
2000–4000'	Huron Towers	~ 790	- 55
	Public land SW of site	~ 770 to 780	- 75 to - 65
	Village Green Apts.	~ 900	+ 55
	Baits Housing*	~ 940	+ 95
4000–6000'	Huron High School	~ 780	- 65
	Medical Center	~ 850 to 870	+ 5 to + 25
	Open land north of site	~ 900	+ 55
	Highest ground elev.*	~ 940	+ 95
0–6000'	Lowest elevation (Huron River)	~ 740	- 105

* Highest Elevation Within Region

In terms of a general description, the land around the reactor site slopes generally downward towards the Huron River. The majority of areas located on elevations higher than the immediate reactor site being located to the north and west of the facility.

2.5 Site Climatology - General Area

The Ann Arbor, Michigan station of the Michigan Weather Service is operated by staff members of the Department of Meteorology and Oceanography. The station was established in 1880 at the Astronomy Observatory and remained there until July, 1944. It was then moved to the roof of the Geology Department Building. In October, 1956, the station was moved to its present location on the roof of the East Engineering Building. These locations are all on the Main Campus of The University.

Weather data for the Ann Arbor station (Reference 2 - covers period of 1930-1959) show that the highest temperature ever recorded here is 105° on July 24, 1934. The lowest of record is 21° below zero on February 10, 1912. Temperatures reach the 90° mark on the average of 15 days a summer and reach 100° or higher in about one summer out of seven. At the other extreme, temperatures fall to zero or lower on an average of twice a year. In about one winter out of three the temperature does not get as low as zero.

January, 1918, with a mean temperature of 11.4° is the coldest month of record, and July, 1955, with a mean temperature of 77.6° is the warmest month of record. The average dates of the last freezing temperature in the spring and the first freezing temperature in the fall are May 2nd and October 17th respectively. Relative humidity figures for Willow Run Airport twelve miles to the east (nearest first order Weather Bureau Office) show an annual average of 79% at 1:00 AM; 80% at 7:00 AM; 56% at 1:00 PM and 65% at 7:00 PM. Seasonal variation causes 1:00 PM humidity averages to be as low as 48% in July and as high as 71% in January. Other times also show a similar seasonal trend but with less variation.

Precipitation is heaviest during the summer months averaging 58% of the annual total for the April-September six month period. Heaviest rain is in May with an average of 3.34 inches. The greatest total monthly precipitation of record is 10.70 inches recorded in July, 1902. The driest month of record is November, 1904 when only .10 inch was recorded. The heaviest intensity of rainfall occurs in connection with thundershower activity and the heaviest recorded 24 hour amount is 3.70 inches on April 4, 1947. Hourly intensity of as much as 1.20 inches occurs with a frequency of once in two years and thunderstorms have a frequency of 34 per year in the vicinity as shown by the Weather Bureau records at Willow Run Airport.

Snowfall averages 30.2 inches per year with considerable variation over the years. Annual totals in the last thirty years have ranged from 13 inches in 1948 to 54 inches in 1951. Median annual snowfall, that is, the most usual amount, is between 25 and 35 inches. January has the greatest average 7.5 inches. The heaviest recorded amount for a single day is 6.2 inches on January 30, 1939.

Cloudiness is greatest in the late fall and early winter, while sunshine percentage is highest in the spring and summer. Sunshine percent of possible ranges from 31% in December to 71% in July at Detroit City Airport (nearest Weather Bureau Office having a long time record). Willow Run Airport Weather Bureau Office records show that Decembers average 4 clear days, 7 partly cloudy days and 20 cloudy days while Augusts average 10 clear days, 12 partly cloudy days and 9 cloudy days. Prevailing wind direction in the Ann Arbor area is southwest with all months showing the direction except March which has a prevailing direction of west northwest. Average velocities are highest in March, 12.9 miles per hour. Highest wind ever recorded in the area is 60 miles per hour from the west in May, 1959.

Since 1959, only two noteworthy weather occurrences took place outside of the general limits given.

In January, 1968, a single snowfall of 11.8 inches occurred. Reactor Operation was suspended for fear of night crews having difficulty in covering their assigned shifts. However, the crews were able to reach work, and most laboratory operations continued without interruption.

In June of 1968, a rainfall of slightly less than 5 inches occurred over a 24 hour period. Flooding of portions of the city occurred, primarily in residential areas near the river and several of the streams which empty into it. Reactor operation was not affected because of the heavy rainfall, and no flooding occurred at the reactor site. However, the interruptions to telephone and electrical supply services prompted a few unscheduled shutdowns of the reactor at this time. These shutdowns, however, did not involve safety related problems.

Based on the above information and the 12 years operating history of the FNR, it is reasonable to assume that the normal weather conditions to be expected in Ann Arbor will have no bearing on the operation of the FNR.

As regards abnormal weather conditions in Michigan which could be damaging to the reactor facility, the following information summarizes tornado activity in Michigan for 1916-1965 (reference 3).

Michigan lies at the northeastern edge of the nation's maximum frequency belt for tornadoes. Normally the number of tornadoes begins to increase during February in the central and eastern Gulf States reaching a peak in April. This increase spreads northward through the area east of the Rockies reaching a peak in May or June. In Michigan for the 50 year period, 1916-1965, June has produced the greatest frequency of tornadoes, tornado days, and number of deaths by tornadoes. Table 2-2 on the following page summarizes tornado information for this period. During this period, the months of November, December, and January have not produced a verified tornado. February has recorded only one; this occurred on February 11, 1932 in Shiawassee County.

To analyze tornado statistics by years at their face value, one would get the impression that there has been a rapid increase in tornado occurrences during the last two decades. This distortion of the figures, however, primarily results from improved communications, increased population, and a more efficient Weather Bureau reporting service rather than from some changed physical phenomena. However, one interesting difference is noticeable between the 1916-1965 period summary and that for the last decade. April has produced more tornadoes, tornado days, and tornado deaths than any other month in Michigan.

For the last decade, Michigan has averaged 9 tornadoes per year. This average appears to be in line with averages for surrounding states for the period 1953-1963; Illinois, 27; Indiana, 24; Ohio, 10; Wisconsin, 13; and Minnesota, 12. The year of 1956, with 19 recorded tornadoes, is tops in this category. In 1953, a total of 127 deaths were due to tornadoes, 125 of them in June. Analysis of the Michigan tornado occurrences indicate about 90 percent in the southern one-half of the lower peninsula. The only section failing to record an occurrence was the eastern portion of the upper peninsula (a relatively sparsely populated area).

TABLE 2-2

SUMMARY FOR 50 YEARS (1916-1965)
AND 10 YEARS (1956-1965)
OF TORNADOES IN MICHIGAN

	Jan	Feb	Mar	Apr	May	June	July	Aug	Sept	Oct	Nov	Dec	Total
					TORNADOES								
50 yrs	0	1	8	32	43	46	17	17	17	5	0	0	196
10 yrs	0	0	2	26	19	12	9	11	6	4	0	0	89
					TORNADO DAYS								
50 yrs	0	1	3	16	22	31	15	16	15	5	0	0	114
10 yrs	0	0	2	10	7	9	6	8	5	4	0	0	51
					TORNADO DEATHS								
50 yrs	0	0	14	74	17	130	3	2	1	0	0	0	241
10 yrs	0	0	0	74	15	0	0	0	0	0	0	0	89

A summary of the unusual and outstanding tornadoes in Michigan would include the following:

1. March 24, 1901. Tornado formed or was first visible as a waterspout while crossing Indian Lake, inundated a large area after leaving the lake and destroyed one home, some orchards and forested areas in Barton and Flint Townships.

2. June 6, 1917. Tornado moved northeastward at 60 mph, from just southwest of Battle Creek, produced complete destruction to everything in its path for about 70 miles. Lifted one substantial building 100 feet in the air according to eye witness. Four killed, total damage over one million.

3. March 28, 1920. Tornadoes moved through west and south portion of lower peninsula. Fourteen persons killed, property damaged exceeded $2,000,000.

4. June 17, 1946. Detroit-Windsor area, $1,500,000 in damages with one death on the American side, 15 deaths on Canadian side.

5. June 5 and 8, 1953. Series of tornadoes over southeastern Michigan caused $19,000,000 in damages, killed 125 people.

6. April 3, 1956. Series of tornadoes caused $10,500,000 in damages and killed 20 people in southwestern Michigan.

7. May 8, 1964. Series of tornadoes in eastern Michigan caused about $4,000,000 in damages. Left 11 dead.

8. April 11, 1965. Series of tornadoes in southern Michigan caused $51,000,000 in damages and left 53 dead.

Figure 5 is a summary of tornado information from Reference 11 which shows that the number of occurrences of tornadoes in the region of the reactor site (2° square is approximately 13,000 sq. miles at this point) is similar to that experienced in western and northern Pennsylvania, western New Jersey, Delaware, and West Virginia.

The following information is more specific to the tornado history of Washtenaw County in which the reactor is located. This information was taken from Vol. 6, No. 3 of Laboratory Safety, a publication of the Department of Environmental Health and Safety of the University of Michigan Health Service:

> "Washtenaw County experiences a 'normal' number of tornadoes, but relatively little damage. Surrounding counties have had more damage. The first tornado recorded in Michigan occurred in Ann Arbor on July 17, 1874, near Sunset Street. There was $20,000 damage done - 20 homes were demolished and 28 were damaged beyond repair. There were no deaths. There have been 7 other tornadoes in Washtenaw County as follows:
>
> "Salem, June 6, 1917 (2 killed); Ypsilanti, July 7, 1917 ($5,000 damage); Bridgewater, April 27, 1948; Willis, July 21, 1951 ($4,000 damage); Scio, May 31, 1954 ($3,000 damage); Willis, April 30, 1962 ($27,000 damage); Chelsea, April 14, 1967 ($5,000 damage). Thus a total of 8 tornadoes have been recorded in Washtenaw County, which is about average for counties in this area. About 80% of Michigan's tornadoes occur in the southern third of the state.
>
> "In this area, many tornadoes occur between 6 and 8 p.m. One-third of all tornadoes occur at this time, and 90% occur between 1 and 11 p.m. Prime tornado time extends from the middle of May to the middle of June, however, tornadoes can occur at any time of the year. The months of April, May, and June comprise the tornado season here.

The earliest tornado reported in this area is March 26; the latest, September 29. A tornado this late is rare and occurred in Monroe County. Tornadoes occur most frequently in the Eastern Plain states of Kansas, Missouri, Iowa, and Arkansas.

"Property damage varies greatly in southeast Michigan. The lowest reported was at Bridgewater - $50.00 and the highest at Milan on April 11, 1965 - $19,000.00. (Not in Washtenaw County).

"An analysis of 26 tornadoes in southeastern Michigan showed that most tornadoes don't kill anyone. Of 34 tornadoes in this area, 26 killed no one; 6 killed 1 to 5 people; 2 killed 6 to 15. These figures, of course, do not include the Flint area where 117 were killed during one tornado a few years ago. In 1925, tornadoes throughout the country killed 689 in one day.

"Two tornado bulletins are issued by the ESSA severe weather forecasters. Tornado <u>WATCH</u> bulletins are issued by the Kansas City Center with consultation with local forecasters. The <u>WATCH</u> is called when atmospheric conditions are such that tornadoes could occur. When a tornado is spotted in an area, a tornado <u>WARNING</u> is given by the local community or communities involved. When a <u>WARNING</u> is given, seek shelter.

"There is no radio station in Washtenaw County which maintains an ESSA weather wire to receive (quickly) severe storm warnings. Because of this, the U.S. Weather Bureau recommends that Washtenaw County residents tune to certain Detroit stations which have a weather wire. These are WJR, WCAR, WXYZ, WQTE, WWJ, WKNR, and WJBK.

"Concerning the tornado itself--rain and frequently hail precedes the tornado--a heavy downpour occurs after it has passed. Direction of travel is usually from a westerly direction--usually from the southwest. Length of path is usually 10 to 40 miles (average--16 miles), but they may move forward for 300 miles. Average width is about 400 yards, but they have cut swaths over a mile in width. Forward motion is 25 to 40 miles per hour, but has varied from 5 to 139 miles per hour. Wind speed inside the tornado is estimated to be as high as 500 miles per hour."

The University Security Office maintains a staff on duty 24 hours a day and notifies the reactor operating staff and administration when a Tornado Watch and/or a Tornado Warning is in effect for the area. A weather monitor FM receiver is also operating in the control room and gives weather alerts for all damaging types of weather storms.

2.6 Site Geology

The geological aspect of the reactor site which is most pertinent to safe operation of the FNR is an evaluation of any earthquake hazard that may exist.

Figure 6 shows the location of the epicenter of all recorded earthquakes in the region of the FNR site since approximately 1800. This information was obtained from Reference 4. The roman numeral in the circles represents the intensity of the quake on the Modified Mercalli Intensity Scale of 1931. The main features of the relation of the scale number to observed damage is given in Table 2-3.

A review of this document (Reference 4) shows that the majority of the earthquakes of record for the central region of the United States have occurred to the southwest of the reactor site in the general area of the Mississippi, Ohio, Illinois, and Wabash River Junctions. These areas are 300 to 400 miles from the reactor site.

Earthquakes of intensity VIII or larger have not been recorded within 300 miles of the FNR site except for the 1937 quake in southwestern Ohio.

Earthquakes with intensities of less than V are not indicated since they produce an awareness of the quake in observers but do no physical damage.

TABLE 2-3

MODIFIED MERCALLI INTENSITY SCALE

IV. During the day felt indoors by many, outdoors by few. At night some awakened. Dishes, windows, doors disturbed; walls make creaking sound. Sensation like heavy truck striking building. Standing motorcars rocked noticeably. (IV to V Rossi-Forel scale.)

V. Felt by nearly everyone, many awakened. Some dishes, windows, etc., broken; a few instances of cracked plaster; unstable objects overturned. Disturbances of trees, poles, and other tall objects sometimes noticed. Pendulum clocks may stop. (V to VI Rossi-Forel scale.)

VI. Felt by all, many frightened and run outdoors. Some heavy furniture moved; a few instances of fallen plaster or damaged chimneys. Damage slight. (VI to VII Rossi-Forel scale.)

VII. Everybody runs outdoors. Damage negligible in buildings of good design and construction; slight to moderate in well-built ordinary structures; considerable in poorly built or badly designed structures; some chimneys broken. Noticed by persons driving motorcars. (VIII Rossi-Forel scale.)

VIII. Damage slight in specially designed structures; considerable in ordinary substantial buildings with partial collapse; great in poorly built structures. Panel walls thrown out of frame structures. Fall of chimneys, factory stacks, columns, monuments, walls. Heavy furniture overturned. Sand and mud ejected in small amounts. Changes in well water. Persons driving motorcars disturbed. (VIII + to IX Rossi-Forel scale.)

IX. Damage considerable in specially designed structures; well-designed frame structures thrown out of plumb; great in substantial buildings, with partial collapse. Buildings shifted off foundations. Ground cracked conspicuously. Underground pipes broken. (IX + Rossi-Forel scale.)

X. Some well-built wooden structures destroyed; most masonry and frame structures destroyed with foundations; ground badly cracked. Rails bent. Landslides considerable from riverbanks and steep slopes. Shifted sand and mud. Water splashed (slopped) over banks. (X Rossi-Forel scale.)

XI. Few, if any, (masonry) structures remain standing. Bridges destroyed. Broad fissures in ground. Underground pipelines completely out of service. Earth slumps and land slips in soft ground. Rails bent greatly.

XII. Damage total. Waves seen on ground surfaces. Lines of sight and level distorted. Objects thrown upward into air.

Based on the low rate of occurrence and the moderate intensity of earthquakes in the area of the reactor site, it is concluded that a specific analysis of the adequacy of the building design to withstand ground shocks produced by quakes is not indicated at this time.

2.7 Site Meteorology

The general meteorological conditions prevailing at the reactor site have been reviewed in detail elsewhere. References 5 and 6 were the documents which supported the request for Amendment 17 (September 17, 1967) to the FNR Operating License (R-28). This amendment permitted the use of a stack dilution factor of 400 for the FNR when averaging stack exhaust activity releases over a one-year period.

This dilution factor of 400 is composed of an instantaneous dilution factor of 100 and an average wind direction frequency factor of 4.

The instantaneous stack dilution factor was determined by observing ground concentrations of Krypton-85 released from the exhaust stacks during various meteorological conditions. A minimum value of 100 was measured for this factor and persisted for only a short time. This study (Reference 5) showed that local weather conditions were not the controlling influences on the ground concentrations at the FNR site; rather, the aerodynamic turbulence caused by the flow of air around the building was found to control ground level concentrations of radioactivity released from the stack.

Moreover, the conditions which could produce a dilution factor as low as 100 could only occur when the wind speed was approximately 10 MPH or more. Based on data accumulated in 1962-1963 (Reference 6), this "critical" wind speed occurs between 27-29%

of the time. The fraction of time that the wind speed is not 10 MPH was not included in the stack dilution factor in order to retain a degree of conservatism.

Reference 6, however, showed that the variations in the directions of the prevailing winds were such that the stack exhaust gases would not be directed towards any specific populated area more than 25% of the time. This information is shown on Figures 2, 3, and 4 in terms of the Plume Direction Frequency (PDF) which is defined as:

Fraction of time exhaust stack plume will
be directed towards a particular area.

The Safety Analysis section of this report bases its conclusions on the use of a dilution factor of 100 (the instantaneous value) since the prevailing wind at the time of such an occurrence could remain fixed for a major fraction of duration of any radioactivity release.

For the analysis of any routine operational conditions which result in the release of radioactivity, the dilution factor of 400 is used.

3. BUILDINGS

3.1 General Comments

The Ford Nuclear Reactor and the Phoenix Memorial Laboratory have been in existence for a number of years. The Phoenix Memorial Laboratory was completed in 1955, while the Ford Nuclear Reactor went into operation in 1957. These buildings as they now appear are shown in Figure 7. The Ford Nuclear Reactor is the brick faced concrete structure in the left of the photograph while the Phoenix Memorial Laboratory is to the right. The floor plan of the existing facilities is shown in Figure 8.

The effect of the masonry building modifications for 10 Mw operation on the exterior appearance of the building is shown in Figure 9. Figure 9 is an isometric view of the Ford Nuclear Reactor and the Phoenix Memorial Laboratory after these modifications have been completed. As can be seen from Figure 9, the primary architectural modifications result from the installation of larger cooling towers on the Reactor Building roof and the addition of a small Auxiliary Building to house additional plant services to the north face of the Reactor Building.

The cooling towers used for 2 Mw operation are located on the existing FNR roof and are housed behind the air inlet screens seen in Figure 7. The size of the cooling towers needed to support 10 Mw operation are too large to be housed behind those screens. The screens, as well as the concrete and brick wall at the northeast corner of the Reactor Building will have to be removed. The new cooling towers will completely fill the roof area and extend over the brick building extension as seen in Figure 7. This wall is not part of the containment line of the reactor facility and its removal does not structurally affect the building.

The Auxiliary Building seen in Figure 9 should not be confused with the existing building extension seen in Figure 7. The building extension which is obvious in Figure 7 is not seen in Figure 9 because it serves as part of the support for the 10 Mw cooling towers. The placement of the cooling towers on the building roof and the existence of the Auxiliary Building is shown in a series of architectural elevations of the Ford Nuclear Reactor and the Phoenix Memorial Laboratory on Figure 10.

3.2 Reactor Building Structural Design

Table 3-I on the next page summarizes the design basis upon which the Reactor Building structural design was originally based.

As was discussed in Section 2 of this Report, the state of Michigan lies at the northeastern edge of the nation's maximum frequency belt for tornadoes and the state experiences an average of about nine tornadoes per year based on the last 10 years. However, as was discussed in Section 2 and shown on Figure 5, the FNR site, based on tornado accumulation information from 1916 to 1955, has not experienced a greater number of tornadoes than regions of the United States not normally considered tornado areas. These "non-tornado areas" would include Delaware, West Virginia, western New Jersey, northern and western Pennsylvania, and the coastal portions of the Atlantic seaboard states.

A tornado can be characterized as a vortex possessing both tangential and translational velocities. According to the information contained in Reference 11, tornadoes usually move at a speed of about 40 MPH, but forward speeds, ranging from stationary to as high as 68 MPH, have been observed. This is a very slow rate compared with the rotary

TABLE 3-1

BUILDING DESIGN PARAMETERS

1. Footings: Placed on soil with minimum bearing capacity of 10,000 lbs./sq. ft.

2. Design Live Loads:
   - 30 lbs./sq. ft. - Roof
   - 200 lbs./sq. ft. - Laboratories and mechanical equipment areas
   - 500 lbs./sq. ft. - Operating floor and first floor
   - 100 lbs./sq. ft. - Others
   - 30 lbs./sq. ft. - External wind load
   - 70 lbs./sq. ft. - Internal building pressure

3. Concrete:

   A. Compressive strength = 3000 P.S.I.
   B. Lightweight concrete for roof = Maximum of 80 lbs./cu. ft.
   C. Lightweight concrete capable of supporting 125 lbs./sq. in.
   D. Roof Slab = 1' - 0" regular concrete + cover of lightweight concrete pitched from 4" to 1" thickness.
   E. Minimum heavy concrete weight = 220 lbs./cu. ft.

4. Reinforcing:

   A. All reinforcing = intermediate grade, high bond bars, except ties and spirals in columns which may be plain bars.
   B. Reinforcing for concrete walls:
      - 8" wall
        - Vertical Bar - No. 4 @ 10" O.C.
        - Horiz. Bar - No. 4 @ 10" O.C.
      - 10" and 12" wall:
        - Vertical Bar - No. 4 @ 12" E.F.
        - Horiz. Bar - No. 4 @ 12" E.F.
   C. Vertical wall bars 2" clear of both faces.
   D. Corner bars are bent 2'-0" around corners.
   E. Horizontal bars lapped 24 bar diameters at splices.
   F. Wire mesh has minimum tensile strength of 70,000 lbs./sq. in.

5. Stairs:

   Live load of 100 lbs./sq. ft. with a safety factor of 4.

6. Drainage:

   All underground piping and joints underground tested to withstand pressure of 10 lbs./sq. in. sustained for 1 hour.

7. Process piping:

   Retention tank in pool water outlet designed to withstand external pressure of 20 P.S.I.; internal working pressure = 20 P.S.I.

speed of winds within the tornado which have been estimated to exceed 200 MPH and possibly may be as high as 500 MPH.

For the purpose of a tornado model and based on information accumulated by the Arkansas Power and Light Co. (Reference 7), the following characteristics were chosen as the tornado-resistant design criteria for their Russellville nuclear unit:

1. A tangential velocity of 300 MPH and a translational velocity of 60 MPH.
2. A pressure drop of 3 psi in three seconds.
3. A missile equivalent to a 4" x 12" x 12' plank traveling end on with a velocity of 300 MPH at any height.
4. A 4,000 lb. automobile traveling through the air at 50 MPH and not more than 25' off the ground.

For a planor wind velocity of 300 MPH there is a corresponding dynamic pressure of 230 lb./sq. ft. which, in turn, would correspond to an average design pressure on a square flat plate of 260 lb./sq. ft. This is not valid for a tornado vortex wind but the magnitude is similar. As can be seen in Table 3-1 by the external wind load design criteria of 30 lb./sq. ft. for the Reactor Building, the building was not designed specifically for tornado resistance. The building walls and roof are 12" reinforced concrete with the exception of one portion of the wall which is located adjacent to the wall of the Phoenix Memorial Laboratory. That wall is 8" of reinforced concrete.

As regards the consequences of the low-pressure region of the vortex enveloping the Reactor Building and causing a high pressure differential across the building walls, the use of the Arkansas Power and Light criterion of 3 psi in 3 seconds would result in a load application on the building walls of approximately 430 lbs./sq. ft. The internal pressure design for the Reactor Building was 70 lbs./sq. ft. Thus, the consideration of internal positive pressure caused by the existence of the tornado, quite apart from the results of the loading on the structure as a consequence of the high winds associated with the tornado, would result in overloading the building walls approximately a factor of six above the design basis for the building.

It therefore appears that an extensive analysis of the structural stability of the present FNR building must be performed. The ability of this building to withstand the tornado resistance criteria of the AEC must be evaluated at an early date.

If this criteria cannot be met, further design and planning efforts may be wasted. It is not, however, altogether clear that the tornado resistance criteria used by the AEC in power reactor evaluations will be imposed as stringently on research reactor facilities, and a clarification of this issue must be obtained from the AEC.

3.3 Auxiliary Building

The new Auxiliary Building which is to be added to the north face of the Reactor Building is shown in greater detail in Figures 11 and 12. This Auxiliary Building shall provide storage and operating space for that support equipment necessary to the routine operation of the reactor facility. The equipment which would be located in the Auxiliary Building would include:

1. The compressor and condensers for the central air conditioning for the facility

2. The additional emergency generator protection that may be required

3. Air compressor

4. The cold water deionizer system for make-up water to the primary water system

5. A 15,000 gallon demineralized water storage tank to permit occasional storage of portions of the primary water system

6. The nitrogen gas bottles which will serve as the dry nitrogen supply system for use as an emergency back-up to the critical portions of the compressed air system (damper control equipment operation, inflatable door seals, etc.)

With the exception of the demineralized water storage tank, the components operating in the Auxiliary Building are not directly connected to any of the process fluids, or contaminated air handling systems of the Reactor Building. The Auxiliary Building will be located outside the containment line of the Reactor Building, enabling the use of normal construction techniques for this building. Shut-off valves will be placed in the lines which connect the process lines from the Auxiliary Building to the process lines in the Reactor Building. Operating personnel will have free movement between the Reactor Building and the Auxiliary Building since the doorway, located at the second level in the Reactor Building, as shown in Figure 12, will be a containment door. An audio-paging system will be placed within the Auxiliary Building so that any operator making checks or performing other duties there can contact the reactor control room and be made available for special duties that may arise.

The elevation of the demineralized water storage tank relative to the reactor pool is shown in Figure 11. This tank elevation makes it impossible to inadvertently drain the reactor pool by opening a valve to the demineralized water storage tank. Should the valve be inadvertently opened, the pool water level would drop approximately four feet. At that point the water levels in the storage tank and in the reactor pool would be equalized and no further drainage would occur. Further drainage of the primary water system into the storage tank requires the use of the primary system pumps. When the pool system has been drained of the 15,000 gallon capacity of the storage tank, the pool water level will be approximately 7-1/2 feet below the pool gutters. This lower water level will be used to facilitate certain maintenance procedures on reactor equipment.

The ability to drain the reactor pool to this lower level would result in reducing the shielding water over the fuel storage racks located around the walls of the reactor pool. Accordingly, new spent fuel storage racks will be designed which are kept on the reactor pool floor. Spent fuel storage in the racks on the pool walls will be eliminated. These racks will be used for storing other lower radiation level items, such as sample holders and other contaminated and partially activated equipment. Designs of racks which can store 30-40 fuel elements at one time are available. Thus, the use of floor storage racks in place of the existing wall storage racks will not reduce the storage ability for "hot" fuel at the facility.

Since much of the equipment in the Auxiliary Building will be maintained by the Plant Department of the University, the building will be equipped with an exterior roll-up door and a truck-loading dock to permit access and movement of repair equipment. Operation of this exterior door will be under the control of the operating staff and the ability to open the door will be restricted to a location inside the Auxiliary Building.

The design criteria to be applied to the construction specifications for the Auxiliary Building are those general criteria which are used throughout the University in the construction of permanent storage areas since the Auxiliary Building is outside the containment line of the reactor facility.

4. REACTOR FACILITY CONTAINMENT

4.1 General Features of Containment System

The containment philosophy to be used for the FNR at a power level of 10 Mw is the "pressure venting" containment as generally described in Reference 8. It is also referred to as the "controlled release" form of containment. The general features of this type of containment philosophy are as follows:

1. During normal operation, all building personnel and equipment doors will be closed except during the passage of equipment and/or personnel into the building. The normal ventilation system for the facility would be a one-pass system with discharge of building ventilation air to the cooling tower bay on the roof. This is the manner in which the existing building ventilation system is operated.

2. In the event of an accident within the reactor facility, the building ventilation system would be isolated from the outside atmosphere by closing large dampers in both the supply and exhaust ducts.

3. An Emergency Ventilation System would then go into operation and exhaust a fraction of the building volume continuously through various filter media (activated carbon, absolute filters, etc.) to remove radioactive material suspended in the building air system as a result of the accident.

4. Operation of the Emergency Ventilation System would maintain a negative pressure within the Reactor Building relative to the outside atmosphere. Reactor Building air will be released after passing through the filtering system and out the building exhaust stacks rather than as an uncontrolled and unfiltered discharge through leaks in the building.

The "containment line" which will be referred to throughout this section of the report refers to that combination of walls, floors, ceilings, and doors which encircle the volume from which the Emergency Ventilation System will draw air in the event of an accident.

4.2 Containment Line Penetrations

4.2.1 General

The containment line is penetrated by a number of systems necessary to the operation of the Ford Nuclear Reactor at 10 Mw. These containment line penetrations include:

1. Piping and conduit for:

   a. Water, steam, and gas supply.
   b. Electrical power supply.
   c. Process water systems.
   d. Building drains.

2. Experimental facilities.

3. Ventilation ductwork.

4. Doorways and hatches for personnel and equipment access.

The steps to be taken to eliminate these containment line penetrations as a source of building air contents leakage in the event of an accident within the facility will be discussed below.

4.2.2 Piping and Conduit

Figures 13 and 14 point out the existing electrical conduit and service piping which penetrates the containment line. It also shows the location of the large "sight" drain on the first floor which is the discharge point for the "cold waste sump," the secondary water system (including "blowdown"), and the primary water system to the storm sewer outside the facility. Each of these containment line penetrations will be checked to assure that the seals originally installed continue to be in good condition. When necessary, seals will be replaced or rebuilt. The water line from the primary water system which permits drainage of the primary to the storm sewer will be removed and blanked-off. The storage tank in the Auxiliary Building eliminates the need for this drainage capability.

The sanitary sewer and storm drain lines which connect to the main drains in the east wall of the first floor elevation will be modified to include a two foot deep trap, as will the large "sight" drain.

Another piping penetration through the containment line is shown on Figure 13, where the secondary water piping system connects to the cooling tower sump and discharge header. These seals are continually in use and the lack of leakage into the facility demonstrates their suitability at the present time. The only time at which there is a possibility of a containment violation is when the secondary water system has been drained and the heads removed from the heat exchangers for the purpose of inspecting the condition of the exchangers. If all main valves in the secondary lines were to be left open, this would constitute violation of the containment line. It will be established as an operating procedure that whenever the heat exchangers are dismantled for service work, a main line valve in both the supply and return secondary water system lines shall be sealed; alternatively, plugs can be installed in these lines as may be necessary.

Through the reactor roof areas, both over the main reactor bay and over the office and laboratory areas, there are a number of sumps which drain rainwater from the roof into the storm sewer system. The collected water is channeled through storm drain piping located within the Reactor Building. The suitability of the joints on these piping systems to remain sealed under the 70 psf positive building design pressure will have to be verified.

The Auxiliary Building introduces additional piping connections passing through the north basement wall. Each of these pipelines will pass through a sealed and gasketed joint. Also, the water line between the main storage tank and the reactor primary water system will have an automatic shut-off valve which is controllable from the reactor control room, and closure will be initiated manually as may be desired or automatically when the Emergency Sequence is initiated. Our experience with the ability to seal such pipelines to the Reactor Building has been quite satisfactory, and no particular problems are expected in sealing the piping and electrical conduit which will be necessary for the equipment in the Auxiliary Building.

4.2.3 Experimental Facilities

At the present time there is only one experimental facility which regularly violates the reactor containment line. This is the pneumatic tube system which enables the transfer of samples from laboratories throughout the Ford Nuclear Reactor and the Phoenix Memorial Laboratory to the reactor core area and their return to the originating laboratory. At the present time there are six 1-1/2 inch diameter aluminum tubes which penetrate the reactor containment line along the south wall of the reactor basement.

At this time there are no specific plans to attempt to change the overall manner of operation of this system. If, when the Emergency Ventilation System is initially tested, it is found that the leak rate through the pneumatic tube system lines is sufficiently large as to be a limiting factor on the performance of that system, then pneumatically or electrically operated shut-off ball valves could be installed in those six aluminum lines. Upon initiation of the Emergency Sequence, the valves would shut and eliminate the p-tube lines as a source of air leakage.

A capsule transfer chute now exists which delivers samples from the reactor pool into Hot Cell 2, the Hot Cell which is located adjacent to the south wall of the reactor pool. While this is not an experimental facility as such, some comment regarding its function with regard to containment line penetration should be made. Basic operation of this transfer port is as a double-valved pipe in which the opening of one valve allows the introduction of a sample into the region between the valves. Closure of the first valve and subsequent opening of the second valve delivers the sample into the Hot Cave area. The opening of both of these valves simultaneously would introduce a major leak into the reactor pool system and corrective action would be required immediately. Consideration will be given to an electrical interlock system for the valves to prevent this possibility from occurring.

4.2.4 Ventilation Ductwork

Air handling system ductwork penetrates the reactor containment line at five points in the 10 Mw design for the Ford Nuclear Reactor.

The main supply air inlet is located immediately adjacent to the exhaust air exit. Both are located in the northwest corner of the Reactor Building. These two main air systems pass through the building containment line at that location. Because of the conclusions reached about the potential Design Basis Accident (Section 15) and the possible over-pressure of the Reactor Building, it will be necessary to re-design these dampers. Re-design of the main dampers in these lines is desirable also from the standpoint achieving a faster damper closing rate. At the present time, a closing time of one second is believed desirable, but this requirement will have to undergo further study. The mechanical strength of the existing design raises questions regarding the reliability of operation and potential leak rate under accident conditions which would have an effect on the performance of the Emergency Ventilation System. The present intent is to replace the existing ventilation system dampers with 30-inch diameter, pneumatically operated butterfly valves which would seat in soft gasket material such as neoprene or rubber. These are commercially available valves and have been suitably used for this purpose in other reactor installations.

Two other ductwork penetrations are made in the southeast corner of the Reactor Building at the second floor elevation. These two ducts are the exhaust air system from the hood located in the third floor laboratory and the "Hot Off-Gas" system which will be described in some detail in a later portion of this section. Both of these ducts pass through the containment line and use pneumatically operated dampers for closure in the event of an emergency. Based on the experience accumulated with these units to date, the existing dampers should be quite satisfactory for 10 Mw operation. A fifth penetration of the containment line will be made to provide an air inlet to the Emergency Ventilation System.

4.2.5 Personnel and Equipment Access

Figures 13 and 14 show the location of the personnel doors and equipment access hatches into the Reactor Building. There are five doors which violate the building containment line used by personnel. These are the three corridors on each of the three floors of the building, the door which will lead into the Auxiliary Building, and the door which leads up to the cooling tower bay. These five doors will use fixed seal (as opposed to inflatable) gaskets and use two point latching hardware similar to that used on industrial freezers. Three of the Reactor Building doors have already been modified to use this type of hardware, and the performance of the doors and the hardware has been quite satisfactory. The type of hardware which is in use on the existing doors is shown in Figure 15. The striking bolt which reaches through the door to allow operation of the panic bar will be sealed by using a compressible sealing boot between the surface of the door and the outside of the bolt. The door is 2-1/2 inches thick, fabricated by plug welding 1/4 inch thick aluminum plates to 2 inch aluminum channels. The channels are placed at the edges of the door and along all hinge and latch hardware bolt lines. This heavy door construction is adequate to withstand the 70 psf requirement for the building containment line.

There are two access doors for moving equipment onto the first floor level of the Reactor Building. These are both located in the south wall which joins the Reactor Building to the Phoenix Memorial Laboratory.

The access door, located on the east side of the reactor pool, as shown in Figure 14, will be sealed and clamped shut. There is no intention to continue to use this door since the location of experimental equipment in that area of the first floor restricts the use of that door.

The vertical sliding equipment access door located on the west side of the reactor pool will be retained in the 10 Mw facility design. At the current time, this door uses fixed, non-inflatable seals. These will be replaced with inflatable seals of the general type surveyed in Reference 34. The control system for the unlocking of the main equipment access door and the deflating of the door seals is shown in Figure 16, while the door modifications are generally shown on Figure 17. The following comments apply:

1. All door controls will be located inside of the Reactor Building so that personnel inside the building must operate the door. This assures that the inadvertent opening of the door will not occur during a building emergency.

2. An electrically operated locking latch will prevent the door from being opened whenever the seal is pressurized. Opening of the door is achieved manually.

3. The time delay relay in the control circuits assures that sufficient time has elapsed for the de-pressurization of the seal before opening of the door can be achieved. This prohibits seal damage from premature door openings.

4. A local audible alarm is initiated whenever the door seals are de-pressurized to alert personnel in the area. An alarm light will also be initiated on the door status panel, which will be located in the reactor control room (and the PML lobby) informing the operator on console duty of de-pressurization of the access door seal.

5. The electrical supply for the control system at the door will be fed from the building Emergency Power Distribution Panel (Y panel).

6. The air supply for the door seals will be fed from the emergency air supply system.

These last two steps will be taken to assure closure and sealing of the door during any extended supply power interruption at the reactor facility.

The removable hatch shown on Figure 14, which connects between the concrete apron outside the Reactor Building and the first floor level, to permit the movement of heavy equipment into the Reactor Building, will be sealed with a fixed gasket seal as presently is the case for 2 Mw operation. Moreover, it will be clamped into place and reinforced, if necessary, to assure that the plate will not be over-stressed, nor will the seals be compromised in the event of the design internal pressure of 70 psf occurring as a consequence of the DBA described in Section 15.

Once 10 Mw operation has begun, this hatch will seldom, if ever, be opened. At the present time, the hatch is completely sealed over and tar-papered to prevent seepage of rainwater into the building. After the 10 Mw conversion is completed and the change to the different type of seal and clamp is initiated, this tar-paper covering will no longer be necessary.

The use of the facility personnel doors will not be restricted during reactor operation -- free movement of personnel through personnel doors will be permitted. However, a Door Status Panel, which indicates to the control room operator the condition of all doors (including the equipment access door), will be provided. Whenever personnel doors are opened into the reactor facility, a local buzzer and light will sound to remind people passing through the doors that the doors must be closed securely. With the opening of any door, the Status Panel will indicate an alarm light, showing the opening of the door. Fifteen seconds after the door is opened, an alarm will sound on the Status Panel if the door has not been resealed. This will alert the control room operator that the door should be checked.

Opening of the equipment access door will also be permitted during reactor operation. However, it will be an operational requirement that the approval of the Reactor Supervisor, Associate Reactor Supervisor, or the Chief Operator must be obtained before the equipment door can be opened. General public tours of the reactor facility, which will still be permitted during 10 Mw operation, will always be directed through the second floor corridor personnel door. This door will be equipped with a T.V. monitor to the reactor control room to indicate to the operator on duty the cause for any prolonged alarm as would occur with the passage of tours into the reactor facility. At this time, T.V. monitoring of the other personnel doors and equipment accesses into the Reactor Building is not intended.

The local alarm and Door Status Panel Control circuits are shown on Figures 18 and 19.

These access controls are subject to change depending upon revisions to AEC procedures.

4.3 Containment Ventilation Systems

4.3.1 Normal Ventilation System

The building ventilation system is a one-pass air handling system with conventional inlet filters and ductwork distribution to various areas of the reactor facility. Central air-conditioning of the reactor facility is recommended for the 10 Mw conversion, and this should involve extensive modification to the supply ductwork system. However, no final decisions have been arrived at regarding the manner by which air-conditioning of facility will be handled.

Additional cold weather heating capacity for the building containment is provided by the means of baseboard hot water radiators supplied by a steam-water heat exchanger located in the equipment room on the second floor of the reactor facility.

The exhaust air system for the reactor facility consists of three general area pick-up ducts which return air to the plenum of the exhaust fan for the facility. The three pick-up points are in the ceiling of the first floor area; the north walls of the reactor bay area; and the additional ductwork to be added to carry air from under the plexiglas covers which will be installed over the reactor pool. This ductwork connection is shown in Figure 12 along with the general arrangement of the service catwalk and the plexiglas panels over the reactor pool.

At the present time there are no individual air return ducts located in the offices and laboratories in the Reactor Building. The exhaust air from offices carries to the aforementioned open area collection points since each office has a fresh air supply and exhaust louvers in the doors. Until decisions are made regarding the manner of central air-conditioning of the facility, there is no intention to change this general manner of supply air connections to the individual offices and laboratories within the Reactor Building.

Since the Emergency Ventilation System, which will be described in the next portion of this section, is used to remove any significant amount of airborne particulate material which might arise in the building air as a consequence of an accident, filtering the normal ventilation system exhaust is not planned.

4.3.2 Emergency Ventilation System

The Emergency Ventilation System will be a series of "absolute" filters and activated charcoal filters for the removal of fission product particulates and iodine which might be released into the building as a consequence of an accident. The magnitude of the accidents and the amount of release to be expected is summarized in detail in Section 15 (Safety Analysis) of this Report. The schematic arrangement of the components as to their physical location in the plant and the general arrangement of the internal filter units within the overall system is shown in Figure 12. The filtering system and its radiation shielding will be located in the exhaust fan room on the third floor of the Phoenix Memorial Laboratory immediately adjacent to the south wall of the Reactor Building. An inlet to the Emergency Ventilation System will be provided through the south wall of the building. The flow rate for the Emergency Ventilation System has not yet been decided upon.

This flow rate will have to be sufficient to maintain the Reactor Building at a measurable negative pressure relative to the outside air and to the air pressure in the Phoenix Memorial Laboratory when all normal ventilation systems have been shut down and all containment line openings sealed. The Emergency Ventilation System shall be capable of meeting the following general specifications:

1. The system will be designed to exhaust air from the Reactor Building at a rate not in excess of 300 CFM. Figure 18 shows the results of some preliminary studies that were done which indicated that a measurable negative pressure in the Reactor Building relative to the outside air and to the Phoenix Memorial Laboratory can be achieved with a flow rate of 300 CFM. The data on this Figure was taken without special precautions taken to eliminate

leaks into the Reactor Building through doors and the pneumatic tube system. This was also done before the new door design was installed on the personnel doors into the facility. Therefore, an Emergency Ventilation System flow rate of less than 300 CFM should be suitable.

2. Air will be drawn through the Emergency Ventilation System by either the existing PML exhaust fans as shown on Figure 12 or a separate Emergency Ventilation Exhaust Fan. The pressure drop through the final filter bank design will determine if there is a need for a separate fan. If a separate Emergency Ventilation Exhaust Fan is required, it shall be connected to the emergency electrical power supply for the reactor facility.

3. Two filter bank units shall be provided, and each unit shall be independent of the other. Both units will not be in operation at the same time. The use for dual units arises from the possibility of high heat loads on the filter media from the gamma and beta decay of deposited fission products. The question of filter media ignition from that heat load will have to be considered in the final design of the Emergency Ventilation System. Figures 19 to 22 present the anticipated heat loads on the particulate and iodine removal filters of the filter system as a function of the type of accident which causes the release of fission products into the Reactor Building. The definition of these accidents are all contained in Section 15 of this Report. For the Design Basis Accident, a maximum heat load of 2000 BTU/hr. is calculated for the iodine filter and 400 BTU/hr. for the particulate filter.

4. Controls for the system shall provide the following functions:

   a. Automatic transfer of the air stream from the unit in operation to the standby filter bank unit in the event that excessively high temperatures are monitored within the filter media of the first unit and automatic closure of isolating valves on either side of the first unit to prevent further combustion of the contents.

   b. Automatic control of the air flow rate through the Emergency Vent System by means of a damper in the inlet ductwork of the Reactor Building south wall.

c. Automatic initiation of operation of the Emergency Ventilation Exhaust Fan, if one is required, upon starting of the emergency sequence.

d. Shutdown of the emergency unit shall be accomplished manually once the system is put into operation.

5. Shielding shall be provided around the filter units of the Emergency Ventilation System to be suitable to reduce the radiation levels from deposited radioactivity on the filters following the Design Basis Accident to a level of not more than 1 R/hr. on the external surface of the shielding. This should be satisfactory for any emergency operations. Figure 25 presents the radiation anticipated from the filter system following the Design Basis Accident and separates each of the gamma ray energies into intervals for convenience in shielding calculations. The total radiation level on the filter unit from all gamma emitters is expected to be 50,000 R/hr. at a distance of 10 ft. from the emergency filters.

6. Equipment shall be provided in the inlet ductwork to the Emergency Ventilation System to provide that the relative humidity of the air entering the emergency filter bank shall not exceed 85% at 120°F. This is necessary to insure the proper performance of the iodine removal filters, especially when one considers the possible formation of methyl iodide (see Section 15.4.7).

7. The filter bank shall contain suitable filters to give the following system performance:

   a. Roughing filters shall be provided to remove most large particulate material which could result in "fouling" of the absolute and iodine filters.

   b. "Absolute" particulate filters shall be provided capable of removing 99.9% of particulate material as determined by DOP smoke test.

c. Iodine removal filters of the activated charcoal type shall be provided capable of removing more than 99.9% of elemental iodine and 99.9% of iodine in the form of methyl iodide (see Section 15.4.7 of this Report).

An Emergency Ventilation System which is capable of meeting the above general specifications will satisfy the safety criteria as specified in Section 15 (Safety Analysis) of this Report.

4.3.3 Hot Off-Gas System

The Hot Off-Gas System is a ventilation ductwork system which provides exhaust air connections to a number of areas within the reactor facility and draws this air off through absolute filters past a monitoring station for ultimate discharge of the facility stacks. This system is used to vent contaminated air systems and other gaseous systems used in reactor facility process equipment or experimental arrangements. At the current time, the Hot Off-Gas System services the beam port storage ports on the west wall of the reactor's first floor level, the reactor beam port area, and the exhaust from the pneumatic tube system. The following units will be added to the Hot Off-Gas System:

1. The vent line from the primary water degasifier system.
2. The vent line from the primary and secondary delay tanks in the reactor primary water system.
3. The vent lines from the air eliminators on the primary heat exchangers.
4. The vent line from the air eliminator on the demineralizer units for the primary water system.
5. The vent line from the air cushion in the pool hot water layer make-up tank.

While the load on the Hot Off-Gas System from these additional components is not expected to be appreciable, it may be necessary to consider the installation of a booster fan in the Hot Off-Gas System duct to supplement the existing PML fans now in operation. There is sufficient floor space in the second floor equipment room in the Phoenix Memorial Laboratory, through which the duct passes, to install such a booster fan.

5. REACTOR CORE AND CORE EXPERIMENTAL FACILITIES

5.1 General Arrangement

Figure 26 shows the general arrangement of core components at the north end of the reactor pool for operation of the FNR at 10 Mw. The core will no longer be supported from the bridge by a tower as is currently done for 2 Mw operation. Instead, a support structure will be fabricated which will mount on the pool floor.

The shim-safety rod positioning mechanisms will still be located on the reactor bridge and aligned with the control core components on the support structure. The reactor bridge will still be movable for maintenance but will be rigidly clamped to the bridge rails during periods of operation. Supported by the core support structure will be two large, aluminum encased lead or bismuth shields which will house the reactor control system ion chambers. The use of the shields will reduce the gamma signal to the ion chambers. Moreover, it is believed that positioning of the chambers below the reactor center line will result in a lesser variation of the chamber output currents as a function of shim rod position. Shim rod movement during operation distorts the core leakage flux and results in a variation in the output current of the ion chambers for the same thermal power level. The reactor will still be fitted with a Heavy Water Reflector Tank in the beam port area of the reactor. This Heavy Water Tank will also be supported by the core support structure.

Because of the heat load which will be generated in the Heavy Water Reflector Tank, a heat exchanger will be located in the reactor pool as shown on Figure 26. This will transfer the heat generated in the Heavy Water Tank to the water which is used in the

Hot Water Layer in the pool system. The reason for this Hot Layer is developed in Section 15 (Safety Analysis) and is discussed in Section 7 (Process Water Systems) of this report.

Because of the potential tritium problem associated with the circulation of large quantities of tritiated heavy water, the totally enclosed circulating pump for the heavy water heat exchanger will be located in a submerged enclosure off the north end of the reactor pool. This is also shown in Figure 26. That submerged enclosure will be vented to the Hot Off-Gas System and periodically monitored to check for the presence of tritium.

5.2 Core Support Structure

Figure 27 shows the core support structure general arrangement in more detail. The present conceptual design of the core support structure should permit installation in the reactor without having to drain the pool. The structure is capable of such underwater loading provided that all of the reactor beam port extension tubes are removed. The primary piping which connects the plenum beneath the reactor core to the water line exit into the delay tank below the reactor pool floor can also be assembled under water.

Since recent observations of the condition of the tile on the inside surface of the reactor pool show the start of tile failure, it may be necessary to install a stainless steel liner inside the reactor pool. In that case, drainage of the pool will be necessary, and a different core support structure design can be considered. In any event, it is recommended that any new core support structure rest on the pool floor rather than a tower type arrangement as is now in use.

There are two specific features of the current core support structure design which should be retained in any redesign which may occur. The first is that the core will be completely housed in a welded aluminum tank. This is to assure that if the pool should drain the action of the anti-siphon valve in connection with the absence of any header mechanism in the plenum assures that while the pool may drain, the core itself will be immersed in a tank full of water. This will provide a moderate amount of emergency cooling for the hot fuel on the loss of the primary water system.

Second, Figure 27 also indicates the presence of a thermal shield around the reactor core tank. This thermal shield will be installed if it is determined that the dose rates at the reactor pool wall might be excessive at a power level of 10 Mw. Ion chamber measurements made at the 2 Mw level indicate that there is a dose of 10^6R/hr. at the corner of the thermal column closest to the core and a dose rate of 10^5R/hr. along the inside of the north wall of the reactor pool. Since the beam port extensions are gasketed to the beam port embedments by a soft rubber gasket, it may be necessary to consider the installation of a thermal shield sufficient to reduce the pool wall gamma ray intensities to the 2 Mw values. The performance and operating lifetime of the gaskets in the reactor pool system have been satisfactory at 2 Mw.

For those familiar with the FNR at the current 2 Mw power level, Figures 26 and 27 do not show the two thru ports (Ports B and C) currently in position in the reactor pool. These beam ports will be removed since they are not utilized effectively at the 2 Mw level. The availability of four shielded access ports from the beam hole floor area into the reactor pool would provide useful access for loop experiments, and hydraulic and/or pneumatic shuttle systems into the core.

5.3 Core Irradiation Facilities

The reactor core will be fitted with a number of core radiation facilities which consist of either cylindrical tubes located in core positions or control type fuel elements. There will never be a core location internal to the core which does not contain a holder, partial fuel element, or some other device to prevent the inadvertent insertion of a fuel element into that location during operation. There will be no open core location in the reactor grid plate. Moreover, the current use of grid plate sealing plugs will not be continued at the 10 Mw level. The presence of a plug in the grid is not considered adequate assurance that a fuel element could not be brought up against the face of the reactor core.

During the preliminary start-up testing of the reactor core, it will be necessary to ascertain the reactivity effect of a singular fuel element brought up against the outer face of the reactor tank and separated from the core by the thermal shield arrangement.

It is anticipated that the fluxes interior to the reactor core in the core irradiations facilities will approach 2×10^{14} nv.

The grid plate construction outlined on Figure 27 allows for a total of 42 core locations within the reactor core tank. It is expected that not more than 35 of these will be in use for the fuel of the operating core. A maximum of 38 has been observed at some reactor facilities but we do not anticipate operating cores that large. This will enable the use of 7 other core locations within the tank for irradiation and experimental facilities.

The design of new in-core hydraulic and pneumatic irradiation facilities has been under consideration for some time and this consideration will continue if the decision for

10 Mw conversion is made. At this point it is only possible to say that there will be sufficient core locations available within the core tank to permit the installation of such irradiation systems.

6. EXPERIMENTAL FACILITIES EXTERNAL TO CORE

6.1 Pneumatic Tube System

Figure 27 shows the pneumatic tube system which consists of four dual pipe sets which rise from the reactor pool floor near the southwest corner of the core and terminate at a location near the west face of the Heavy Water Reflector Tank. This is the pneumatic tube system currently in operation at 2 Mw and will be retained for 10 Mw operation. The following laboratories in FNR and PML are connected to these pneumatic tubes:

1. The laboratory in room 3103 (FNR).
2. The laboratory in room 1066 (two locations - PML).
3. The laboratories in rooms 1044 and 1054 (one location shared by both rooms - PML).

Pneumatic tube system is a 1-1/2 inch diameter system which currently uses commercially manufactured polyethylene rabbits of the type shown on Figure 28. These rabbits are currently obtained from the Japanese firm of Atom Kyogo K. K. Corporation. These rabbits have functioned quite satisfactorily at the 2 Mw level and total rabbit exposure times of up to 10 hours have been recorded without rabbit capsule failure. Although the rabbit exposure time at 10 Mw will have to be reevaluated, it is estimated that at least 2 hours of irradiation time should be available for each of these rabbit units before they must be disposed of.

The pneumatic tube system operation uses one of two vacuum blowers located in the basement of the Reactor Building. At the present time they are located in the northwest corner of the reactor basement. Because of changes which will be necessary to the basement equipment locations for 10 Mw operation, these blowers will be moved to a different basement location which is shown in Figure 47 (southwest corner near Hot Sump).

Because of the gamma heating which will be experienced at 10 Mw, it may be necessary to modify the present operational sequence for the pneumatic tube system blowers. Personnel now turn off the pneumatic blower during long irradiation periods. The blower is used only for insertion and removal of the sample. It will be necessary to determine whether the 10 Mw gamma heating rate of the rabbit is sufficiently high to require that the blower operates at all times to prevent softening and deformation of the rabbits. A number of 5 Mw installations which have similar pneumatic tube systems maintain their blowers in continuous operation for just this purpose.

Section 15 (Safety Analysis) will discuss the production of argon-41 in the reactor system and its release to the outside environment. Whether or not the pneumatic tube blowers are in continuous operation does not affect argon-41 releases from the facility because of the amount of leakage in the pneumatic tube systems.

6.2 Radial Beam Ports

Figure 27 shows the eight radial beam ports of the reactor. No modifications to the in-pool sections of the radial beam ports is intended at this time, nor are any necessary to accommodate the 10 Mw core support structure. The radial beam ports will continue to operate against the face of the Heavy Water Reflector Tank. Nitrogen operated metal bellows will still be used between the face of the beam ports and the Heavy Water Tank on those ports which can receive such equipment.

The following modifications will be made to the radial beam port equipment for 10 Mw operation:

1. All collimators and shield plugs used in the beam ports used at the 10 Mw level will be provided with additional, larger vent holes to prevent pressure build-up from dissociation effects on the shielding concrete from prolonged gamma ray and fast neutron exposure. This will be done to prevent the pressure build-up induced plug failures which have been reported at the ORR, The Puerto Rican Nuclear Center, and the Union Carbide Corporation Research Reactor. In all these cases, completely sealed collimator and shield plug assemblies were used, and the pressure build-up inside those assemblies was sufficient to cause failures ranging from cracking of welds to explosive failure of the end cap of the collimator plug.

2. Only metals and concrete will be used at the core end of any collimator construction. The use of paraffin, polyethylene, and other organic-type materials will be restricted to the far end of the shield apparatus to prevent breakdown of that material from prolonged exposure to radiation.

3. Wherever possible, collimators will be of the "short" variety in that the collimators will be located within the concrete shield walls rather than extending to the port face. Collimator designs which require a portion of the collimator be installed up to the end of the reactor beam port against the Heavy Water Reflector Tank must be analyzed to determine if sufficient cooling will exist to eliminate any potential overheating of the end of the collimator or other radiation effects such as swelling and other distortions. The reactor radial beams using full-length collimators will have the first twelve inches of that collimator unit exposed to a gamma intensity from the core of not more than 10^9 R/hr. This will be the limiting case of gamma ray intensities at the reactor beam ports. These values apply to the A and J ports. All other port locations will experience lower radiation levels at 10 Mw.

   At the present time, there are 2 "long" collimators in use in the FNR. These are located in the A and J ports. It will be necessary to reevaluate their suitability for 10 Mw operation.

4. A port water circulation system will be provided to permit circulation of water used in flooded beam ports to minimize the effect of internal corrosion of the beam port liners, the build-up of gas pressure from the dissociation of the water under the higher radiation levels at 10 Mw, and the production of N^{16}. This water system could also be used for any needed cooling of collimators.

5. A retaining ring will be provided to the outer face of reactor beam port shield plugs and collimators so that any core pressure transient as might follow a SPERT type accident, as discussed in Section 15, would not be able to effect ejection of the beam port contents. This retaining ring is shown in Figure 29 and will reduce the probability of rapid drainage of the reactor pool following such a transient. This modification should eliminate the beam ports as a potential source of pool leakage.

6.3 Access Ports

Figure 26 shows a set of "pass-through" ball valves located at those points which used to be the connection flanges for the B and C thru-ports in the reactor pool. As was mentioned in the previous section of this report, these thru-ports will be eliminated for 10 Mw operation and ball valves will be installed in their place. By means of the in-pool ball valve and another seal closure on the external face of the pool walls, it will be possible to provide access to core irradiation facilities for experimental apparatus installed on the beam hole floor. Such experiments might consist of circulating fluid loops (gas, high pressure water, etc.), and pneumatic or hydraulic shuttle systems for sample irradiations in the core whose counting and handling apparatus is located on the beam hole floor.

6.4 Heavy Water Reflector Tank

It was previously stated that the Heavy Water Tank currently in use in the FNR at 2 Mw was designed to accommodate the 10 Mw power level. Because of restrictions posed by the existing tower structure, the present design only enables four of the eight radial beam ports to be fitted with water displacing bellows assemblies. If it is decided that the other ports should be fitted with bellows to make maximum use of the neutron intensity available at 10 Mw, a new tank can be built to provide bellows installations at all eight ports.

Irradiation facilities are available in the Heavy Water Reflector Tank for the exposure of small capsules (1 inch maximum diameter). A 3 inch diameter facility is available as well, and is currently occupied by the neutron radiography vertical exposure tube.

6.5 Support Structure Irradiation Facilities

The reactor core support structure will also be fitted with a number of irradiation facilities external to the core tank. These are shown as holes in the grid plate structure extension on Figure 27 along the south and east faces of the thermal shield structure. Preliminary estimates of the fluxes available in this location suggest that the grid plate extension fluxes will be in excess of the maximum flux available at the 2 Mw level. This will markedly increase the over-all experimental utilization of the core in the ability to conduct sample irradiations. This grid plate extension can also serve as the mating point for large experiments.

6.6 Reactor Pools

In a pool-type reactor, the pool is an additional radiation facility for large objects. No modifications of the reactor pool are presently contemplated for 10 Mw operation. The installation of large experiments should still be possible using techniques developed to suit individual experiment needs.

It has been noted previously that a pool cover will be installed. Referring to Figure 12, it can be seen that the service catwalk arrangement for operational personnel will not be aligned with the center line of the reactor pool core. This has been done intentionally so that removal of the plexiglas panels from the west side of the catwalk will permit the insertion of large experimental facilities into the pool.

The installation of the service catwalk and the plexiglas panels over the top of the pool will be done in such a way that it would be possible to install a support mechanism and tower (often called an Instrument Bridge) on the existing bridge rails from which experimental apparatus could be suspended and moved periodically up to and away from the operating reactor core. The planned pool cover system will not prevent the use of the bridge rails as a support point for experimental apparatus.

7. NUCLEAR DESIGN AND THERMAL ANALYSIS

7.1 Nuclear Design

7.1.1 Core Type

Section 1 of this report discussed the reasons for selecting the MTR type fuel element for 10 Mw operation of the FNR. This type of fuel element is used in the reactor at the 2 Mw level, and its design will be modified slightly for use at 10 Mw.

It may be recalled that a primary reason for the selection of the MTR type of fuel element is the extensive operational experience with this design in a number of operating research and test reactors in the United States. Its heat transfer characteristics are superior to other possible fuel element designs considered. The experience with this design ranges up to power levels of 40 Mw. Variations of this basic design have been used as high as 175 Mw.

For the 10 Mw operation of the FNR, it is planned that the fuel elements will be used with initial fuel loadings of 200 grams. Similar fuel elements with higher weight loadings (240 grams) have been successfully used in a number of test reactors (Plumbrook, ORR, AFTR, and MTR).

7.1.2 Fuel Element Design

Figures 30 and 31 show the two types of fuel elements which will be used in the FNR at 10 Mw. The upper end of the fuel elements will be modified to provide the elements with greater protection against the possibility of a loss in cooling flow to the individual fuel sections from debris. The modifications include providing a series of small diameter rods across the fuel element water inlet openings which would serve to support any large debris (gasket material, work gloves, etc.) similar to the types of things which have caused a fuel element blockage and the release of fission products as described in Section 15.2.3 of the Safety Analysis portion of this Report. In order to provide some degree of coolant flow to any element upon which debris has fallen, emergency cooling slots will be cut into the side plates of the fuel elements as shown in both figures. This will permit water from adjacent fuel elements and from the space between adjacent fuel elements to flow below the blocked portion into the fuel region of the elements. Although these slots will not provide as much coolant as is available through the normal top opening of the individual elements, they should reduce the possibility of damage to the element.

The fabrication specifications for the fuel elements have to be finalized. The heat transfer calculations discussed in a later portion of this section of the Report were computed based on the use of the specifications currently in existence for the FNR 2 Mw elements. It is the intention of the operating group to compare the existing fuel fabrication specifications with those which are in use for the Oak Ridge Research Reactor and the Plumbrook Test Reactor fuels. The FNR specifications will then be modified to take advantage of the higher power operating experience of those facilities. Fabrication specifications are also available for the Air Force Test Reactor (10 Mw). Fuel elements have been successfully fabricated for these facilities by all the United States fuel element manufacturers. Therefore, use of these specifications as a guideline in the formulation of specifications for the 10 Mw FNR would not lead to fabrication requirements which are either economically prohibitive or technically unfeasible.

Figure 32 summarizes the hydraulic characteristics of the 18 plate fuel element which would be used in the FNR at a power level of 10 Mw. The information in this figure was compiled from the data presented in References 37 and 38. The values presented in Figure 32 are average values. The effect on heat transfer calculations of variations in this flow rate for different elements and the flow variation within the individual fuel element channels is considered in Appendix E to this Report which summarizes the heat transfer calculations.

The element utilizes a uranium-aluminum alloy in the "meat" portion of each individual fuel plate. An alternate meat design is now available which will be considered during the final design studies for 10 Mw operation. Both the ETR (175 Mw) and the ATR (250 Mw) reactors utilize a powder metallurgy type meat construction in which the uranium is in the form of UAl_3. UAl_3 powder is homogeneously mixed with high purity aluminum powder to make up other necessary meat proportions. This meat section is then clad with high-purity aluminum by conventional rolling techniques, as is done in the present alloy-type system. One of the advantages of UAl_3 over the alloy system is that the melting point of UAl_3 is higher (2460° F) than the alloy (1220° F). Furthermore, its retention of fission products in the event of a cladding failure is superior to the alloy system, thereby reducing the consequences of any fuel element cladding failure.

Preliminary information from Atomics International, a division of Rockwell Corporation, which manufactures this type of fuel element, indicates that the heat transfer characteristic of the UAl_3 system is essentially identical to that of the uranium-aluminum alloy system and the fabrication costs are comparable.

7.1.3 Core Mass and Reactivity Coefficients

Because of the extensive experience available with this type of fuel element in research and test reactors, core physics calculations were not performed to predict core critical masses and the reactivity coefficients for the reactor.

A graduate student (Reference 39) did perform a series of preliminary calculations using codes available at The University of Michigan. These were done to estimate the applicability of these codes to the 10 Mw design of the FNR. To check the suitability of these codes for this application, calculations were performed to predict FNR core masses using the present 140 gram fuel elements of the FNR. The results were not encouraging. For example, a core containing 32 fuel elements with an average burnup of 18% was calculated to have sufficient excess reactivity for an operating cycle at 2 Mw. In practice, a 32 element core with an average burnup of greater than 10% will not provide sufficient excess reactivity for a one-cycle run. The existing code as it was used did not take into account the control elements used for shim-safety rods or irradiation facilities. Moreover, new fuel elements with no burnup are loaded in the center of the core as the highest burnup elements are moved to the outer edges of the core. This locating of control elements within the core and the non-homogeneity of the core burnup is believed to complicate the ability to predict core masses with the existing code.

Successful codes, however, have been generated at a number of the national laboratories which operate cores of this type in their research reactors (Oak Ridge National Laboratory, National Reactor Testing Station, etc.). If it is deemed worthwhile to have such codes available for calculational purposes, they could be obtained and modified for our use.

It is clear, based on the experience available from other facilities, that core physics computations for this reactor need not be independently done at this location for the purpose of a preliminary analysis of reactor behavior. There is sufficient experience available so that additional local core physics calculations, although useful and instructive to the operating group, are not necessary for a safe start up of the reactor to a power level of 10 Mw.

Based on this experience, reactor core sizes of between 25 and 40 fuel elements are anticipated for the FNR. The factors which determine this variation in core operating fuel mass include the element arrangements, experimental facilities in the core, and the degree of burnup of individual fuel elements.

Table 7-1 on the following page summarizes the core reactivity coefficients pertinent to the operation of the FNR at a power level of 10 Mw. These coefficients were deduced from published information from reactor facilities which utilize a similar fuel element design, operating core masses, and overall core component arrangements similar to those anticipated for the 10 Mw FNR. Specific values will be determined during the preliminary startup testing of the FNR after conversion to the 10 Mw design.

In order to crudely simulate the effect of fuel burnup on the initial core configuration, the initial reactor fuel loading will consist of fuel elements whose U^{235} content will be intentionally varied. Standard fuel elements will be fabricated for use in the initial core with U^{235} content of 200, 180, 160, and 140 grams. Control fuel elements with U^{235} content of 100, 80, and 60 grams will similarly be used in the initial core.

TABLE 7-1

CORE REACTIVITY COEFFICIENTS*

Core temperature coefficient	$-(5 \times 10^{-3} - 7 \times 10^{-3})\%\ \Delta K/K/^{o}F.$
Core void coefficient	$-(1 \times 3 \times 10^{-4})\%\ \Delta K/K/C.C.$
Central fuel element worth	$+(3.5 - 5\%)\ \Delta K/K$
Beam ports worth	0% (uncoupled by D_2O tank)
Fuel burn-up	$-(0.14 - .020)\%\ \Delta K/K/MWD$
Heavy water reflector tank	$-(5 \times 10^{-3} - 7 \times 10^{-3})\%\ \Delta K/K/^{o}F.$
Sample void coefficient (in core)	$+(0 - 3) \times 10^{-4}\%\ \Delta K/K/C.C.$

* Estimates based on operating experience from reactors using similar core components

7.1.4 Core Reactivity Accountability

Table 7-2 below indicates the estimated reactivity worth of various factors for sustained operation of the FNR at 10 Mw. Since the average worth of a shim-safety rod is not expected to exceed 2.5% $\Delta K/K$, this tabulation indicates why a shim rod operating complement of 6 shim-safety rods will be required for 10 Mw operation of the FNR. A more complete description of the shim-safety rods will be given in Section 9 (Instrumentation and Control) of this Report.

7.1.5 Core Arrangements

At this time it is not possible to indicate the specific operating core arrangement of fuel elements and experimental facilities which will be used in the core. However, certain general statements can be made which will apply to all operating cores for the facility. These are as follows:

1. The Heavy Water Reflector Tank will always be in position for all operating cores in order to couple the operating core to the reactor beam ports.

2. Six control type fuel elements with shim-safety rods in place will be present in every operating core.

3. Two or more control type fuel elements without a shim-safety rod (used as irradiation facilities) will be located in every operating core.

4. All new fuel elements loaded during a refueling procedure will be inserted into the center of the core with the higher burnup fuel elements located towards the outer periphery of the core.

5. No two control elements will be located in adjacent core positions along the north-south direction.

TABLE 7-2

CORE REACTIVITY ACCOUNTABILITY

<table>
<tr><th>Core Reactivity Accountability Parameter</th><th>Value</th><th>(% ΔK/K)</th></tr>
<tr><td>Temperature (isothermal increase from 80° F. to 115° F.)</td><td>.18</td><td>.25</td></tr>
<tr><td>Temperature (80° F. at zero power to 10 Mw)</td><td>0.25</td><td>.35</td></tr>
<tr><td>Core Experiments</td><td colspan="2">2.0</td></tr>
<tr><td>Heavy Water Reflector Tank (80° F. to 120° F.)</td><td>.20</td><td>.28</td></tr>
<tr><td>Xenon - 135</td><td>3.6</td><td>4.1</td></tr>
<tr><td>Samarium - 149</td><td>0.8</td><td>0.9</td></tr>
<tr><td>Low cross section fission products (added during operating cycle)</td><td>0.04</td><td>0.06</td></tr>
<tr><td>Operating allowance for fuel burn-up* (150 MWD)</td><td>2.1</td><td>3.0</td></tr>
<tr><td></td><td>9.2</td><td>11.0</td></tr>
</table>

* Continuation of existing 28-day cycle expected, but will be accomplished with one midcycle shutdown for refueling.

6. Control elements used as irradiation facilities will be the higher burnup control elements with the lower burnup control elements reserved for use in control locations.

With these preliminary restrictions in mind, Figures 33 - 35 show different core configurations which will be considered for use at the 10 Mw level.

7.2 Thermal Analysis

7.2.1 General

This portion of the Report will concern itself with the thermal analysis of the FNR core at a power level of 10 Mw. A "hot-channel" analysis was performed for steady state operation. The calculations and input data are summarized in Appendix E (Heat Transfer Data and Calculations) of this Report. The basis and results of the calculation will be summarized.

Specific computations of the transient thermal conditions that may result from either a reactor power transient or a loss of coolant flow type accident is not treated in this section. Information concerning these types of transient behavior are contained in Section 15 (Safety Analysis) of this Report.

The analysis was performed to assure that sufficient core coolant flow rate and system capacity will be provided so that the maximum fuel plate surface temperature in the "hot-channel" will be below the local boiling point of the coolant.

7.2.2 Coolant Mass Flow

Figure 36 shows the calculated maximum total coolant flow which can be obtained for reactor cores containing between 25 and 40 fuel elements.

From Figure 36 it can be seen that the maximum reactor flow rate available for a 25–element reactor core is approximately 6500 gallons per minute. This flow rate of 6500 gal./min. was chosen as the design specification for the reactor primary water system. The heat transfer analysis was based on a total core flow rate of 6500 gal./min. The calculation which was performed to determine this maximum flow rate took into account the following core conditions:

1. Each of the core sizes considered would contain six control type fuel elements fitted with shim–safety rods and two control type fuel elements used as irradiation facilities.

2. 20% of the water flow rate through the reactor core will bypass the reactor fuel elements by passing through the region between adjacent fuel elements.

3. The primary water system bulk temperatures will vary between 80 and 120° F. (This system temperature variation has a negligible effect on the pressure drop calculations.)

Figure 37 presents the values of the average mass flow rate of coolant through an average fuel element coolant channel as a function of the reactor system total flow rate and the number of fuel elements in the core. It can be seen from the figure that for the reactor core sizes of interest to this analysis,the average coolant channel mass flow rate varies from 4000 lbs./hr. (40 element core) to 6400 lbs./hr. (25 element core). The thermal analysis assumed that the "hot channel" would have a mass flow rate which is 70% of the average value. This was based on the results of hydraulic studies presented in Reference 38. It is interesting to note that under the specifications by which the ORR fuel elements are fabricated, actual measurements performed at the ORR demonstrate a hot channel mass flow reduction of a maximum of 17%. Thus our calculation of the steady state thermal results for the reactor core may have additional conservatism over that originally intended for the calculation.

7.2.3 Coolant Characteristics

While the FNR 10 Mw design does not involve the use of a pressurized core in the sense of a tank-type research reactor, the immersion of the reactor core under a pool water depth of 20 feet does result in a slight pressurization of the reactor primary water system as regards the coolant fluid characteristics in the area of the reactor core. Figure 38 shows the variation of the boiling point of the reactor water as a function of the coolant pressure in the reactor core.

For example, a reactor core with no flow, therefore, no core pressure drop, would have a coolant pressure of 20 feet of water and a boiling point of 238° F. Correspondingly, a core pressure drop of 20 feet of water would place the coolant in the reactor core at ambient pressure with a boiling point of 212° F. The second graph on Figure 38 shows the pressure drop through the reactor core region as a function of the mass flow rate in an individual coolant channel. This is the same data which was presented in Figure 32, but in terms of coolant channel mass flow rate rather than total element flow rate.

Figure 39 presents the data of Figure 38 in a more useful form for the computation of the "hot-channel" conditions. It shows the boiling temperature of water within a particular coolant channel as a function of the mass flow rate through that channel.

Figure 39 shows the minimum coolant boiling point channel. This minimum occurs at the point of maximum pressure drop in the channel. This point of maximum pressure drop is at the discharge point from the channel at the bottom of the fuel element. This is not the point of maximum surface temperature of the fuel elements since that occurs at the point of maximum heat flux (center of the element). When determining whether or not the non-boiling criteria for the "hot-channel" had been met, this discrepancy between the minimum boiling point temperature and the maximum fuel plate surface temperature was not taken into account to add a further degree of conservatism into the calculation. This approach adds a safety margin of at least 5° F. to the calculated difference between the maximum fuel plate surface temperature and the boiling temperature of the coolant fluid at that point.

7.2.4 Heat Generation Rate

In order to estimate the heat flux which will be generated in the "hot-channel" of the FNR, the following assumptions were made regarding parameters pertinent to heat distribution:

1. The fuel in the fuel plates bordering the hottest channel are uniformly distributed in the vertical direction.

2. The core heat flux distribution was assumed to be a clipped cosinusoidal distribution in the vertical and transverse axes.

In order to account for the neutron flux and fuel mass distribution variations in the core, the cosinusoidal distributions used are shown on Figure 40. The parameters for the five curves shown represent the ratio of peak core flux to the flux at an outside face of the reactor. The magnitude of this ratio depends upon how the reactor core is loaded and the physical size of the core including the effectiveness of the reflector used (heavy water, light water, steel). The ordinate in Figure 40 is the ratio of the heat flux in the hottest channel to the heat flux present in the average channel of a core.

Figure 41 shows the maximum heat flux which will be experienced by cores of various sizes (using core restrictions enumerated above) for FNR elements at a power level of 10 Mw. A 25-element core with a ϕ max./ϕ min. ratio of 4 will have a maximum heat flux of approximately 270,000 BTU/hr./ft.^{2}

7.2.5 "Hot-Channel" Considerations

As used in this Report, the "hot-channel" factor is that factor which adjusts the maximum calculated heat flux to compensate for uncertainties in thermal, hydraulic, fabrication, and operational factors which would introduce errors into a calculation. The uncertainties which were considered in the heat transfer analysis are summarized in Appendix E. They are briefly summarized below:

1. Minimum equivalent hydraulic diameter for fuel channel = .017 ft. (nominal dimension = .019 ft.).
2. Maximum/minimum channel coolant flow velocity variation = 1.5.
3. Reactor power measurement error = 5%.
4. Flow velocity variations within channel = 5%.
5. Fuel core mass variation over length = 2%.
6. Uncertainties in heat transfer coefficient for channel = 20%.

7. Fuel core thickness variation = 13%.

8. All fission energy assumed to be deposited locally in fuel plate (conservative by approximately 10%).

The heat flux distribution described above included variations caused by neutron flux differences in the cores. A heat flux variation of 4 (ϕ max./ϕ ratio) is not observed at the FNR at a power level of 2 Mw.

7.2.6 Results of Calculations

Figure 42 shows the calculated temperature increase of the fuel element cladding in the "hot-channel" necessary to cause local boiling. The results are presented as a function of primary flow rate through the reactor, the variation in neutron flux distribution in the core, the total reactor power level, and the size of the reactor core (25 and 35-element cores shown).

The effect of neutron flux distribution on the maximum surface temperature of the fuel elements is seen to be of less importance than the total power level of the reactor or variations in the primary coolant flow rate through the reactor. The information for power levels in excess of 10 Mw is of interest since boiling in the "hot-channel" is undesirable at the power level which initiates automatic shutdown (scram level) of the reactor. The 25-element core at 10 Mw and with a flow rate of 6500 gal./min. has between 35 and 40° F. margin before boiling can begin. The margin for the 35-element core with the same flow rate is in excess of 50° F. All further calculations were based on a flux ratio of 4.

Figure 43 presents similar information to that contained on Figure 42 in that it presents the calculated cladding temperature increase necessary to cause boiling in the "hot-channel" for reactor cores of various sizes as a function of the total core flow rate. In each case, the banded region of temperature for each power level shown is determined by a lower line representing the data for a 25-element core and banded by an upper line representing the data for the 40 element core. It is apparent that the 25-element core is always the limiting case as regards the onset of boiling in the "hot-channel."

At the full flow rate of 6500 gal./min., one could raise to 14 Mw and still have some 8° margin available to avoid boiling. If the reactor flow rate was allowed to drop as low as 5500 gal./min., boiling in the hot channels still should not occur.

Figure 44 presents the operating limitations which would be imposed upon the reactor at 10 Mw. The temperature data shown is appropriate to the 25-element core. The normal operating region for flow at 10 Mw will be 5500 - 6500 gal./min. The cladding temperature for an inlet pool water temperature of 110° F. (the maximum pool level temperature permissible) will be more than 20° F. below the boiling point of water in the "hot-channel" everywhere within that flow range. At 115% of full power (11.5 Mw), the auto-control system will initiate an automatic insertion of rods for a reduction of power. The high power scram point will be set at 125% of full power (12.5 Mw). If the high power scram level should be reached when the primary flow rate is just slightly above the low-flow scram point of 5500 gal./min., then the "hot-channel" will be approximately 5° below the temperature at which boiling will occur for a 110° F. inlet water temperature. The margin will be larger for all larger cores (greater than 25 elements).

7.2.7 Comparison With Measured Results

In order to determine the amount of conservatism present in the calculations, a calculation was performed using input data appropriate to the average hydraulic flow characteristics of the Air Force Nuclear Test Facility Reactor, a 10 Mw facility located at Wright-Patterson Air Force Base in Dayton, Ohio. The results of this calculation are shown on Figure 45 along with the "hot-channel" measurements made by Air Force personnel and reported in Reference 35. The method used in the FNR calculation predicts consistently higher maximum fuel cladding temperatures with a difference between the calculated and measured "hot-channel" fuel temperature being 28° at a power level of 10 Mw. Thus the calculations which have been performed should be suitably conservative to insure that operation of the FNR at 10 Mw and 5500 - 6500 gal./min. should be safe from a heat removal standpoint.

Data was also available from the startup tests which were performed on the Kyoto University 5 Mw reactor. Experiments made at that facility show that the hottest fuel element surface temperature was observed to be approximately 54° F. greater than the coolant temperature in that region. A calculation was performed for the Kyoto conditions and a maximum fuel plate surface temperature of 90° above the local coolant temperature was calculated. Thus our calculation technique, when compared to measurements at two other reactor facilities, showed consistently conservative predictions.

7.2.8 Thermal-Hydraulic Induced Problems

The mechanical strains and stresses induced in reactor fuel elements because of the hydraulic forces from high flow rate and the thermal expansion caused by the high heat load from the fission energy deposition within the fuel is discussed in some detail in Reference 40. That reference summarizes fuel element design work and performance tests conducted on the ORR elements for 30 Mw operation. Their satisfactory results in the operation of that facility, and the fact that we will utilize their fabrication experience for the FNR fuel, should assure the FNR elements are suitable to withstand the thermal-hydraulic stresses which will occur at 10 Mw.

8. PROCESS WATER SYSTEMS

8.1 General

Figure 46 shows the process water systems necessary for 10 Mw operation. Figure 47 shows the equipment lay-out for the basement of the Reactor Building necessary to accommodate the additional equipment required for 10 Mw operation.

A primary consideration in the design of the process water systems for 10 Mw operation was that the breakdown of a singular component should not completely interrupt reactor operation. With this in mind, the process water systems were designed, where possible, to provide parallel paths so that reduced power level operation (approximately 5 Mw) would be possible after a primary system component failure. The items which could not be provided as parallel systems to permit lower power operation of the reactor include:

1. The primary water system components between the core support structure and the first delay tank.
2. The existing 1400 gallon delay tank.
3. The reactor pool structure.

However, the components provided as parallel systems include:

1. Primary Pumps.
2. Secondary Pumps.
3. Heat Exchangers.
4. Additional Delay Tanks.
5. Hot Dionizer Units.

Malfunction of the other systems would either not seriously hamper operation at 10 Mw, or are not required at the 5 Mw level.

The following portions of this report will summarize the preliminary design decisions and the areas yet to be evaluated for the following process water systems:

1. Reactor Coolant System including the primary water system, the shutdown cooling and circulation system, and the secondary water system.

2. The Heavy Water Reflector Tank Coolant System.

3. The Hot Water Layer System.

4. Auxiliary Water Systems including:

    a. Hot and cold demineralizer water purifications systems
    b. Water Degasifier System
    c. Make-up Water System
    d. Experimental Facility Cooling System

5. The Emergency Water Systems including:

    a. City water make-up to pool
    b. Emergency Core Spray System
    c. Anti-siphon Valve

8.2 Reactor Coolant System

8.2.1 Reactor Primary Water Systems

The general circulation pattern for the coolant water in the primary system is as follows:

1. Pool water will flow (6500 gpm) down through the reactor core components, be collected below the core in the core support structure plenum, and through the anti-siphon line which

consists of a piping rise of ten feet above core level as shown in Figures 26 and 27 past the anti-siphon valve.

2. After passing the anti-siphon valve the primary water passes through the existing 1400 gallon delay tank which is located below the reactor pool floor.

3. The water is drawn out of the first delay tank by the two primary pumps to a common discharge header to the two delay tanks located in the northeast corner of the reactor basement.

4. After passing through the two delay tanks the primary coolant passes through the two shell and tube heat exchangers into the return header and past a swing check valve located immediately below the reactor pool. This check valve will be installed to prevent rapid back-drainage of the reactor pool to any pipeline or tank which might occur during an accident in the basement. Such accidents have been treated in Section 15 of this Report, and drainage of the pool is prohibited by:

   a. Action of the anti-siphon valve to prohibit drainage of the pool below the ten foot level.
   b. The existence of the return line check valve which prohibits water from flowing through the return header to the fracture location.

5. After passing through the check valve, the primary water returns to a distribution header located on the reactor pool floor. This header must be designed to minimize water turbulence in the reactor pool system which could adversely affect operation of the hot water layer system which will be described.

At the present time no controls will be provided to insure the even distribution of water flow between the two delay tanks and the two shell and tube heat exchangers. Manual throttling of the gate valves on the discharger side of these units to produce equal unit pressure drops should be a suitable technique for insuring balance flows between the units. If preliminary testing of the manual proves unsatisfactory, a pneumatically operated control valve could be installed.

These following general specifications have been arrived at for some of the components of the primary water system:

Primary Pumps:

The primary pumps shall have a total output flow rate of 6500 gpm when operating in parallel. Each unit shall deliver 3500 gpm at 100 feet TDH. They shall be fitted with stainless steel or aluminum pump casings, shafts and impellors with mechanical shaft seals. They shall be in line coupled motor/pump combinations to permit motor replacement in the event of motor burn-out. (Lack of ease of motor replacement is one of the major drawbacks of the present Monobloc construction of pump used in the FNR.) Pump motor shall be 100 hp, three phase, 440 volt open drip-proof motors. (440 volt service will have to be provided to the reactor facility because of these motor requirements. Use of 220 volt motors would result in exceptionally high starting currents which would cause line voltage variations deleterious to reactor operating instrumentation.)

Delay Tanks:

The two new delay tanks installed on the discharge side of the primary pumps shall be 1800 gallon capacity stainless steel welded tanks 42 inches in diameter by 24 feet long when assembled. These tanks are too large to be brought into the reactor basement in one piece. They will have to be shop fabricated with a bolted flange joint which can be field erected in the basement work area. The unit shall be designed for 75 psig internal working pressure in accordance with the latest edition of the ASME Unfired Pressure Vessel Code. Because of the low pressure service and the non-corrosive nature of the working fluid, it should not be necessary to design these components in accordance with the conditions of the nuclear vessel interpretations of the code.

A limiting consideration in the selection of the tank sizes which can be moved into the reactor basement is the dimensions of the floor hatch and the concrete beams cast into the roof of the basement between the first floor and the basement.

The following tank sizes are the maximum combinations of tank shell diameter and length which can be accommodated through the existing hatch:

1. 4' - 6" diameter by 16' long
2. 4' diameter by 18' long
3. 3' - 6" diameter by 20' long
4. 3' diameter by 22' long

Any tank shell which meets these dimentional limits can be rigged into the basement area through the hatchway without having to remove portions of the first floor itself.

Heat Exchangers:

Heat exchangers shall be dual five Mw units to operate in parallel. They shall be shell and tube, fixed tube sheet, removable head (both ends) construction with stainless steel in contact with primary water; mild steel may contact the secondary water. Units shall be designed for secondary water flow through the heat exchanger tubes (2 pass) with the primary flow rate through the tube sheet (1 or 2 pass). Layout on Figure 47 is based on single pass unit. Units shall be designed for 75 psig shell tube pressures and 100 psig internal tube pressures in accordance with the latest version of the ASME Unfired Pressure Vessel Code. Again the nuclear interpretations of the code need not be followed for these units. Heat exchangers shall work under the following operating conditions:

Primary water flow rate 3500 gpm
Primary inlet temperature 112° F
Primary exit temperature 100° F
Secondary inlet temperature 82° F
Secondary exit temperature 94° F
Secondary flow rate 2900 gpm

Primary Piping:

Primary piping shall be schedule 40 steel pipe, field fabricated to convenient lengths with flange joints between fabricated sections. After field fabrication is completed, pipe shall be removed and fitted with rubber lining to serve as protection of the steel piping against corrosion by the primary water. Piping then to be reinstalled. This fabrication technique chosen to minimize cost. This type of pipe is in use at the FNR now and has performed satisfactorily.

Reactor Pool Lining:

The need to have a suitable pool lining to minimize the dissolved mineral content of the primary water system is well known for pool type reactors. What is not well known, at this point, is whether or not the condition of the existing ceramic tile pool lining is satisfactory enough to allow its continued use at 10 Mw. There have been some instances of individual tiles falling off the walls. The extent of the tile lining deterioration will have to be established. If it is sufficiently extensive to warrant replacement, then it is recommended that a stainless steel pool liner be installed in the reactor pool over the existing tile, rather than replacing the tile.

8.2.2 Reactor Shutdown Cooling System

Section 15.2.4 in the Safety Analysis portion of this Report shows that there is not a safety need for a shutdown cooling system for the reactor core. This was concluded on the basis of analyses performed at similar reactor facilities which showed that interruption of the primary pumps and the loss of reactor flow rate would not result in excessively high core temperatures. Moreover, the anti-siphon valve described previously opens on loss of primary flow. This valve opening establishes natural connection flow through the reactor core, and this flow is sufficient to keep the surface temperature of the fuel elements below the boiling point of the coolant. The anti-siphon valve will be designed to open whenever the reactor flow rate drops below 500 - 1000 gpm in addition to opening whenever the pool level is less than 8 feet above the reactor core.

Although a shutdown cooling system is not necessary from a safety standpoint, the value of such a system to the operations performed during periods of reactor maintenance cannot be discounted. When a reactor of the FNR type has been shut down and maintenance work is performed on core components (fuel element transfers, sample loading and unloading, reactor core component rearrangement, etc.), the natural convection cooling of the shutdown core results in thermal currents which distort the core visibility available to the operator. Thus for the sake of operational convenience, a shutdown circulation pump will be provided which will circulate 1000 gpm down through the reactor core when the main pumps are not in operation. This pump will start upon shutdown of the primary pumps.

8.2.3 Reactor Secondary Water System

The schematic arrangement of the components of the secondary water system are shown on Figure 46. It has been mentioned previously in this Report that a four cell cooling tower located on the roof of the Reactor Building will transfer the 10 Mw power generated in the reactor core to the outside atmosphere. The secondary pumps and the heat exchanger for the system are located in the reactor basement, the layout of which is shown on Figure 47. Secondary system water, after passing over the cells of the cooling tower, will be collected in the cooling tower sump located in the building roof. The water will be carried by a 16 inch water line (existing) to the basement where the two secondary pumps operating in parallel will pump the water through the tube side of the main heat exchangers. The use of the tube side of the heat exchangers for the secondary water, while causing an increase in cost for the heat exchanger units, enables the cleaning of the inside of the tubes of any fouling material deposited on the tubes by the secondary system. Since the primary water system is basically a filtered and purified water system with a very low solids content, the passage of the primary water over the less accessible shell side of the heat exchanger is a better system arrangement despite the increased costs.

After the secondary water is heated by the reactor primary water system in the heat exchangers, the secondary leaves the heat exchangers and passes through a 16 inch pneumatically operated throttling valve before returning to the distribution headers located in the four cell cooling tower on the building roof. This control valve which throttles flow in the main secondary water line to the cooling tower is used to control the discharge temperature of the primary water from the heat exchanger. By this method the bulk pool water temperature will be controlled at 100° F. This type of system is installed on the FNR for use at 2 Mw and operates satisfactorily. A number of other reactor facilities use a similar system for controlling their bulk pool water temperature.

The following specifications have been arrived at for certain components of the secondary water system:

Secondary Water Pumps:

The secondary water pumps shall have an output capacity of 3,000 gpm each with a 120 foot TDH. The pumps may be iron body steel shaft and bronze impeller construction similar to the unit currently in use but fitted with mechanical seals. They should be of the in-line mounted, motor/pump combination rather than monobloc construction. The motors for the pumps shall be 100 hp, open, drip-proof construction, 440 volt, and three phase.

Cooling Tower:

The cooling towers will have to be units which require air inlet from one side only since the cooling tower units will be located against the reactor bay area wall on the building roof as shown in Figures 9, 10, and 11. Towers suitable for this operation would be 3 Marley Cooling Tower Model 473-703 plus 1 cell which is Model 473-701. The 3 similar units would be mounted along the east wall of the reactor bay with the fourth unit mounted along the north wall of the bay area. The general arrangement of this type of cooling tower is shown in Figure 48. The units will each have 30 hp, 2 speed, reversible (low speed only) fans.

Cooling Tower Makeup Water System:

The makeup water system which provides makeup water to the secondary system to compensate for the evaporative water loss from cooling tower operation shall be controlled as is currently done for the 2 Mw case. A sump level controller will monitor the level in the cooling tower sump and control a pneumatically operated throttling valve in a city water line connection to the secondary water system in the basement as shown in Figures 46 and 47. The makeup water system for 10 Mw operation should be capable of delivering 600 gpm with the city water main pressure of 60 psig when the cooling tower basin is full and the throttling valve is wide open.

Main Control Valve:

The main throttling valve for controlling the flow rate of the secondary water system shall be a butterfly construction valve with a pneumatic operator. The valve body need not be fitted with a soft seat material since tight closure of the valve will not be desired. Moreover, the valve wafer shall be modified so that when the valve is fully closed as determined by full extension of the pneumatic operator shaft, the total flow rate in the secondary water system shall be reduced to approximately half of its maximum value. Total shut-off of the secondary water system flow rate is undesirable. The pneumatic operator shall be constructed as a diaphram type operator and mounted so that on loss of air, the secondary water valve will move to the fully open position. Thus temporary air failure to the valve operator will not restrict reactor operation while valve operator or air line repairs are made.

8.3 Heavy Water Tank Cooling System

During operation of the FNR at a power level of 10 Mw the heat input to the

Heavy Water Reflector Tank immediately adjacent to the reactor core will be approximately 50 kilowatts. The present heat input into the existing 2 Mw Heavy Water Tank is approximately 10 kilowatts. Temperatures have been measured in the tank and show that the heavy water in the tank reaches a maximum temperature approximately 40° F above the bulk pool temperature. From this it has been inferred some cooling of the heavy water will be required to prevent boiling in the Heavy Water Tank. Moreover, the heat generated in the tank can be used as a source of heat for the Hot Water Layer System which will be installed in the reactor pool and will be described in Section 8.4.

A further desirable effect of cooling the heavy water in the tank will be to eliminate the need to compensate for the negative temperature effect on the reflector effectiveness of the Heavy Water Tank. (This temperature effect will be determined during the early full power testing of the reactor.)

Heavy water will be drawn out of one standpipe of the tank by a totally enclosed pump located in a submerged enclosure which was shown on Figure 26. The submerged housing will be used as a means of monitoring for the presence of any heavy water leakage from the pump and control of any subsequent release of tritium into the building atmosphere. The heavy water will be pumped through the shell side of a stainless steel shell and tube heat exchanger mounted four feet below the pool surface on the pool wall. This heat exchanger will serve as the heat source for the Hot Water Layer System. The cooled heavy water will then return to the other standpipe of the

Heavy Water Tank. A general conceptual arrangement of the components of this system are shown in Figure 49. Their operation will be discussed in greater detail in Section 8.4 below.

Specifications for the components to be used in Heavy Water Tank Cooling System have not yet been determined. A heat transfer analysis of the Heavy Water Reflector Tank to estimate the expected temperatures and total heat input into the tank is required. The heavy water in the tank should be cooled to a point as close to the reactor core exit temperature as possible so that there is a minimum thermal gradient between the two systems while retaining sufficient thermal energy so that the heavy water can serve as the heat source for the Hot Water Layer System.

8.4 Hot Water Layer and Pool Makeup Water Systems

Section 15.1 of the safety analysis portion of this Report discusses the dose rates from the reactor pool surface which will be experienced during operation at 10 Mw. That section of the report concludes that it is advisable to establish a 2 foot thick, clean (radioactive free), hot water layer on the pool surface to act as a shield against the gamma activities from the contamination in the primary water system. This technique of using a clean water layer as a biological shield is in use at other reactor facilities. For example, the French Siloe' reactor, which operates at 30 Mw, periodically establishes a hot water layer on its pool system by directing the reactor core exit water directly to the pool surface for a short period of time without passing that water through the system heat exchanger. For several hours at a time, this serves as an effective biological shield in the Siloe' system.

For the FNR at a power level of 10 Mw, we are attempting to establish a continuously maintained Hot Water Layer System. Work on this system is presently underway with the piping being completed and the effort now being directed towards the establishment of a suitable heat source. Evidence collected to date at the 2 Mw level indicates that we have occasionally reduced the gamma ray levels from the contaminated water by a factor of 4 but have not been able to maintain this layer for lack of suitable heating source.

The layer system will also operate in conjunction with a makeup water system to maintain the reactor pool level at the desired overflow rate.

Basic elements of the Make-up Water System and the Hot Water Layer System are shown in Figures 46 and 49. The system will operate in the following way:

> Water overflowing the pool gutters will be brought to the reactor basement through a header system to the hot layer make-up tank. This tank will be an air-cushion tank whose tank level is monitored by a controller. A decreased make-up tank level will indicate that sufficient evaporation has occurred to reduce the overflow rate in the reactor pool gutters. At this point the controller will open a throttling valve connected to the main line from the 15,000 gallon storage tank in the Auxiliary Building to permit water from the storage tank to enter the make-up tank and thus compensate for evaporative losses in the pool. The water passing through this make-up control will be metered to provide a measurement of the pool water loss rates. The hot layer pump will pump water to the hot layer deionizer (DI) and filter system which will purify the water before returning it to the reactor pool area. The hot layer DI will not function as a full stream de-ionizer unit since we expect the hot layer circulation rate to be in excess of 100 gallons/min. The de-ionizer unit will act as a bypass system with a regulating valve as shown in Figure 46 to maintain a constant flow rate and to automatically operate the de-ionizer unit and filter as a bypass system. After leaving the purification section of the system, the flow will be diverted through

a portion of the tubes in the Heavy Water Tank heat exchanger as shown in Figure 49. From there it will be distributed to the distribution header located 2 feet below the water level.

There will be no throttling of the water flow in the hot water layer circuit. The other portion of the tubes in the two section heat exchanger used for the cooling of the heavy water will be fed by a throttled water line from the pool system hot de-ionizer and filter system. The throttling valve in this circuit will be controlled by a temperature measuring system which will monitor the temperature of the hot layer. Should that temperature become excessive, the valve will open and direct pool de-ionized (DI) water to the other tube section of the heat exchanger. The water which passes through the heat exchanger from the pool DI circuit will be discharged directly into the reactor pool. The heat exchanger is to be mounted at a point four feet below the pool surface so that the direct discharge of this water does not interfere with the stability of hot water layer.

Specifications for the system components for the hot water layer and pool make-up water system have not been established.

8.5 Auxiliary Water Systems

8.5.1 Water Purifications Systems

Purification of the water used in the reactor pool (other than the hot water layer) shall be accomplished by the use of dual, manually operated de-ionizer units similar to the existing hot and cold de-ionizers used at 2 Mw. The existing units will be converted for use as this "hot DI" system and mounted behind a shielded enclosure in a basement location as shown on Figure 47. The radiation dose rate from each of these units is expected to be between 20 and 50 R/hr. (at contact). The majority of the radiation is expected to be from Na-24. The units will not be in operation at the same time. One unit will operate until its charge is depleted. Then the other unit, whose contained radioactivity should have been decaying for approximately 20 days, will be recharged and put into service. The radioactivity of the first unit will then be allowed to decay before recharge. In this way it will be possible to minimize the amount of radioactivity sent to the waste water storage tank system (see Section 12) in PML for ultimate disposal to the outside. Recharging of the hot de-ionizer system provides the largest quantity of activity to the waste water collection tanks.

The water from the hot de-ionizer units will be used to supply:

a. The hot water layer heat exchanger cooling circuit described above,

b. The DI system discharge line into the main return water line to the reactor pool, and

c. The supply header for the beam port circulation system to be described.

The demineralized water needed to compensate for losses from the reactor water system (evaporation, pump leakage, etc.) will be supplied by a manually controlled, automatically regenerated make-up de-ionizer (cold DI). This DI will be located in the Auxiliary Building near the 15,000 gallon storage tank. (See Section 8.4.) Make-up water will be drawn from the storage tank rather than the de-ionizer itself. Therefore the de-ionizer, once it is regenerated, will be operated to depletion of its charge with its putput discharged directly into the water storage tank. When the water level in the storage tank is reduced to 500 gallons, an alarm will sound to notify the control room operator who can then send personnel to the Auxiliary Building to manually initiate the automatic recharge procedure.

A demineralizer with a capacity sufficient to demineralize 2000 gallons of Ann Arbor water to a purity of not less than 1,000,000 ohm-cm will be satisfactory.

8.5.2 Pool Water Degasifier

Section 15.1.4 of the safety analysis portion of this Report discusses the desirability of installing a degasifier circuit in the pool water system to reduce the evolution of argon-41 from the reactor pool surface when the operators must open the plexiglas panels over the pool. At this time no specifications have been developed for this particular piece of equipment. Preliminary work with commercial manufacturers indicate that suitable degasifiers are available but that the "head room" requirements are restrictive for installation of the equipment in the basement area. Accordingly, it may be necessary to design our own equipment or to modify the commercially available equipment to provide this degasifier function. Further development work on the degasifier design, its applicability to this type of system and its effectiveness in reducing the argon-41 concentrations evolved from the pool surface, must be completed before design specifications can be formulated.

8.5.3 Experimental Facility Cooling System

In the discussion of the "hot DI" system above it was mentioned that a portion of that system's return water will be diverted to the Experimental Facility Cooling System. The general conceptual arrangement of the system is shown in Figure 46 for a typical beam port.

The supply header will be a pipeline installed around the outer face of the reactor biological shield from which valved, individual lines will be provided to each of the horizontal beam ports. Connections can be made to these valve lines either for circulating water through the beam port liner in the case of a shielded and unused beam port or can be connected to collimator cooling systems as may be required for 10 Mw operation. Discharge water from an individual beam port will then be carried through a valved line to a port collection header mounted on the face of the reactor shield. This header will then rise to the third floor and return into the reactor pool over the pool parapet. The discharge point for this will be several feet below the surface so that the discharging water will not interfere with the operation of the hot water layer.

8.5.4 Thermal Shield Cooling

Consideration has to be given to the need to cool the steel plate thermal shield which is to be constructed around the reactor core tank on the core support structure. The general configuration of the shield was conceptually shown in Figure 27. A preliminary estimate of the heat load on the thermal shield suggests that, for a thermal shield constructed of half-inch thick stainless steel plate, a maximum heat flux of approximately

30,000 BTU/hr. will be generated in the plate closest to the reactor tank. It will be necessary to evaluate the heat flux more carefully and determine whether or not natural convective cooling of the thermal shield will be satisfactory. The present conceptual design of the system assumes that it will be. However, the heat flux to be experienced does appear to be marginally close to the boiling threshold and further analysis is needed. There are a number of ways by which cooling water could be directed to the thermal shield if such is found to be necessary.

8.6 Emergency Water Systems

8.6.1 General

Emergency water systems have as their function the providing of sufficient water to the reactor pools to either compensate for, or to mitigate the consequences of, large leaks in the reactor primary water system. The manner by which such leaks might occur and their consequences on a reactor operation is treated in some detail in Section 15 (Safety Analysis) of this Report. The various systems available in the 10 Mw design of the FNR will be described below.

8.6.2 Auxiliary Building Storage Tank

Since the 15,000 gallon storage tank to be located in the Auxiliary Building is to be used for water storage when the reactor pool level is lowered for maintenance purposes, it is not likely that there will be a very large supply of water in that tank whenever the reactor pools are full. It is, however, the point to which the cold DI system will discharge after regeneration whenever the auxiliary storage tank contains less than 500 gallons of demineralized water. Therefore the demineralized storage tank should contain, during normal operating periods, between 500 and 2500 gallons of demineralized water. This could be used to compensate for any minor leaks which might develop in the pool or primary water system and for which corrective measures can readily be taken. The water loss could then be compensated for by pumping water from the storage tank back into the pool system. In the event of any large leaks from the reactor primary system, the Auxiliary Building storage tank probably would not provide an adequate supply of make-up water.

8.6.3 City Water Line

Figure 46 shows that a four inch diameter water line will be provided from the main city water line into the Reactor Building to the inlet to the shutdown circulating pump with a branch line to the sixteen inch return water line to the reactor pools. Valves in these lines are manually operated and can be used to direct a flow rate of approximately 600 gpm into the reactor pools to compensate for major leaks. The control valves for these lines are located in the reactor basement and, as pointed out in Section 15 (Safety Analysis), it is possible a leak originating within the reactor basement could make the valves inaccessible. This emergency water line system is in existence at the present time and is the technique by which city water could be directed into the reactor pool system. At the present time there are no plans to change the control valve locations to another point in the building.

8.6.4 Core Spray System

A fire nozzle will be mounted at the north end of the reactor pool and aimed in such a way that it can direct a stream of water onto the reactor core tank at a rate of not less than 300 gpm. Control of the water through this line will be achieved by a manual, quick-opening block valve located in an area close to or within the reactor control room so that personnel may have ready access to it in the event of an emergency. In addition to the manual valve, there will be an automatic valve which will open the spray nozzle line whenever the reactor pool level has dropped to a point 8 feet above the reactor core. This is the point at which the anti-siphon valve should have opened and would only be reached in the event of a major pool leak from the reactor system. The use of the parallel mounted automatic valve is justified in the event that the accident which causes the leak to occur was of a nature to render the operational crews unable to actuate the manual valve. By initiating the core sprays prior to exposure of the core the remote possibility of a metal-water reaction from the core spray water with an overheated fuel element should be avoided.

8.6.5 Anti-Siphon Valve

The anti-siphon valve operates, in a sense, as part of the emergency water systems in that one of its functions is to prevent the complete loss of pool water from the reactor core whether the system leak which requires use of any emergency water system occurs in the reactor pool walls, through the beam ports, or in any of the reactor water systems located in the Reactor Building basement.

9. INSTRUCTIONMENTATION AND CONTROL

9.1 General

Table 7-2 in Section 7 of this Report discussed the range of core reactivity accountability which has been estimated for operating the FNR core for 15 days (longest duration between mid-cycle shutdowns) at a power level of 10 Mw. This operational capability of 15 full power days will be discussed in Section 13 (Administration) of this Report. Based on the information in Table 7-2, an overall excess reactivity requirement of between 9 and 11% $\Delta K/K$ seems indicated for the intended operational schedules at 10 Mw. Section 8 of this Report discussed the process water systems necessary to 10 Mw operation.

This section will concern itself with the instrumentation and control features necessary to achieve safe control of the core excess reactivity requirements and to assure satisfactory operation of the process water systems necessary to full power operation.

9.2 Nuclear Instrumentation

9.2.1 Types of Systems

The reactor will be equipped with three types of nuclear instrumentation systems for operation at 10 Mw:

1. Start-up systems with detectors which are used as indicators of relative power in order to observe neutron level behavior in the subcritical to supercritical neutron level ranges.

2. Wide range systems to enable determination of reactor power level from low power region (approx. 1 Kw) to full power (10 Mw).

3. Safety systems to provide a fixed full power level reference and to permit determination of power level up to and beyond the scram level safety point.

9.2.2 Start-up System

The start-up system for the FNR will be essentially the same as the one now in use. Two fission chambers will be used as detectors feeding two independent logarithmic count rate circuits. The fission chamber count rate will be displayed on meters and recorders. The channels will also differentiate the count rate in order to display the count rate period on the same recorder which displays the count rate. A count rate range of 4 decades will be satisfactory.

Drive mechanisms will be provided for the waterproof housings which contain the fission chambers. A total stroke length of 8 feet should be available. The drive mechanism controls will be interlocked to prohibit both drives being in operation at the same time.

While this is the system in use at the present time, new solid state design count rate instrumentation is being ordered to replace the existing tube type circuitry, and new drives will have to be designed which:

1. Provide the desired greater stroke.

2. Occupy less space on the bridge structure than the current units.

The fission chamber waterproof housings will operate with guide tubes which will be located in grid positions within the reactor core tank. The grid positions used will vary dependent upon the core configurations in use.

A block diagram of this system is shown in Figure 50.

9.2.3 Wide Range System

The reactor will be provided with three wide range instrumentation systems. Two of these will have logarithmic displays and one will have a linear display. The general arrangement and features of this system are shown on Figure 51. The detectors for all three of these systems will be located in the chamber shields on the core support structure.

The Log N Channel will consist of a compensated neutron sensitive ionization chamber with associated power supply feeding into a solid state logarithmic amplifier with level and period outputs. The level output will be displayed on a panel meter located on the amplifier and on a single pen recorder. The period output will also feed a local panel meter, one pen of a dual pen recorder, and the Neutron Period Safety Amplifier.

The Log Gamma Channel will consist of a gamma ionization chamber, power supply, and a logarithmic amplifier identical to that used in the Log N Channel. The level output will be displayed as for the Log N Channel but the period output will be fed to the other pen of the dual pen period recorder. The gamma period output will be fed to the Gamma Period Safety Amplifier.

The use of a wide range, gamma sensitive channel is to provide back-up to the neutron sensitive channels at higher power levels and to provide additional diagnostic information when anomalous behavior of the neutron sensitive chambers occurs.

The Linear N Channel will consist of a compensated neutron sensitive ionization chamber with chamber power supply and a micro-microammeter with output from the meter fed to a panel meter and a recorder. This recorder will work in conjunction with the servo control system for auto power level control. This is the system currently in use at 2 Mw and will be quite satisfactory for 10 Mw operation.

The control interlocks taken from each of the channels of this system are shown on Figure 51.

9.2.4 Safety System

The general arrangement of the components of the Safety System is shown on Figure 52.

The Safety System as shown, which is designed to provide rapid shutdown of the reactor in the event of the core reaching 125% of full power or increasing power at a period of 5 seconds or less, is presently being fabricated, and component testing should begin in September, 1974.

The basic system design philosophy requires that if any input signal reaches its scram point, reactor shutdown will occur. Coincident or "2 of 3" type logic will not be used.

The system inputs will consist of:

1. The output current from 3 uncompensated neutron sensitive ion chambers (UIC).
2. The output current from a gamma ionization chamber.
3. The period output from the Log N channel amplifier.
4. The period output from the Log Gamma channel amplifier.

Chamber power supplies will be provided within the Safety System for the four ionization chambers. A visual panel mounted alarm light will indicate an output voltage reduction of more than 5%. A test button will be on each power supply chassis to test this alarm point before and during operation.

The various input signals will be fed into individual Safety Amplifiers which will provide a linear display of reactor power (or period) on a panel meter on each chassis. Each Safety Amplifier will also have a recorder output. Two of these amplifiers (1 neutron and the gamma level) will be fed to linear scale Safety Level Recorders. Each of these recorders will have interlock switches at 110% and 120% of full power as will the Linear N channel of the wide range system.

The Safety Amplifiers have two alarm annunciator set points:

1. 125% full power to indicate which of the amplifiers initiated the reactor shutdown.

2. 80% full power to inform the operator on duty that a specific amplifier is either malfunctioning or the detector has suddenly reduced its output current. This alarm interlock will be of greatest use after the reactor has reached full power.

Since the Safety System is composed of Solid State circuitry, a separate Safety System Power Supply for the Safety Amplifiers and portions of the other components of the system was assembled separate from the individual amplifiers. This power supply provides the 12 Vdc supply (positive and negative) for the entire Safety System. It is provided with alarm annunciators to alarm whenever either of its outputs (± 12 Vdc) vary by more than a fixed amount. The set points of these alarms will be decided upon after the evaluation program for the Safety System is concluded.

The magnet current to each of the individual electromagnets holding a shim-safety rod is controlled through a Magnet Current Control chassis. Each control unit receives a separate voltage supply from the Magnet Current Power Supply to enable producing individual magnet currents of up to 100 ma d.c. Each control is fitted with the following features:

1. Scram signal from any Safety Amplifier results in complete interruption of magnet current for duration of signal.

2. Recessed magnet test button to permit manual release of an individual magnet as in rod release tests.

3. Current control potentiometer (0-100 ma range).

4. Fuse protection on magnet leads set at 125-150 ma to prevent high magnet currents which could interfere with shim-safety rod releases.

5. Alarm light which indicates when fuse has blown.

Figure 52 also shows the general arrangement of a shim-safety rod, its drive mechanism, and some of the other components of that portion of the system. These will be described in greater detail in another part of this section.

Another feature of the reactor Safety System will be the functions provided by the Safety System Test Panel. These functions include:

1. Simulation of output current of ion chambers to calibrate the Safety Amplifiers used to monitor power level.

2. Exponentially increasing output current to Log N and Log Gamma channels to calibrate Safety Amplifiers used as period safety channels.

3. Once every 5 seconds, the current input to one of the Safety Amplifiers will be increased to a point above the

> scram level for 20 microseconds. The output current from the six Magnet Current Controls will go to zero for 50 microseconds. If any control fails to shut off the current to the electromagnet, an alarm will sound to notify the operator on duty. Each of the amplifiers will be so tested in sequence.

This last feature will provide a functional test of the system's ability to achieve a reactor scram while at full power and without interrupting reactor operation. The failure to interrupt reactor operation arises from the fact that the magnet current must be depressed for approximately 3-5 milliseconds before release of a shim-safety rod will occur. The short duration of the test sequence will not result in the release of a rod.

The electromagnets used in the Safety System will be that design which is currently being used at 2 Mw. These have proven to be quite satisfactory.

9.3 Process System Instrumentation

9.3.1 General

The instrumentation described in the following section refers to the functions which should be monitored rather than the specific devices to be used. The selection of specific commercial devices suitable for the application intended will be deferred until the final design stages of the 10 Mw conversion.

Throughout the following sections, reference will be made to four locations at which the instrument making a measurement can be "read." These include:

1. Local (L): A device which has a read-out (dial, recorder, etc.) directly mounted at the point of installation of the device; e.g., a pressure gage.

2. Local-remote (LR): A device whose read-out is slightly removed from the installation point but is in the general area of the installation point; i.e., a pressure gage on a primary pump discharge line but located on the face of the primary pump radiation shield to permit observation during reactor operation.

3. Process Panel (PP): The device read-out is located on a separate Process Equipment Read-out Panel located at the entrance to the reactor basement as shown on Figure 47.

4. Control Room (CR): The device read-out is located in the Reactor Control Room to permit observation by the operator on duty.

9.3.2 Reactor Primary Water System

Table 9-1 on the following page summarizes the function for which instrumentation is to be provided for the Reactor Primary Water System. Those parameters which will be displayed in the reactor control room will be handled as follows:

1. All system temperatures will be displayed on a multipoint recorder with a recorder interlock set at 128° F.

2. The ΔT for the reactor core will be displayed on a separate recorder with interlocks at 16° F and 18° F.

3. Primary (and secondary) water flow rate will be displayed on a dual pen recorder.

4. System pressures or Δp displays will be on panel meters - not recorded.

9.3.3 Reactor Shutdown Cooling System

No additional instrumentation is required by this system over that provided for the Primary Water System. Low flow (2000 gpm) or main pump shutdown will start the shutdown pump.

TABLE 9-1

REACTOR PRIMARY WATER SYSTEM INSTRUMENTATION

FUNCTION	READ-OUT LOCATIONS	INTERLOCKS
Pressure Measurements		
1. Δp between core exit water line and pool at anti-siphon valve	None	$\Delta p < 6"$ H_2O
2. First delay tank pressure	LR, PP	$p > 20'$ H_2O
3. Inlet and outlet pressure on old pumps	LR, PP	None
4. Δp across each of the new delay tanks	LR, PP	None
5. Inlet, exit pressures on both heat exchangers	LR, CR	Inlet pressure
6. Δp across each heat exchanger	LR, PP, CR	None
Temperature Measurements		
1. Core exit temperature (2 units) (Located in basement before first delay tank)	CR, PP	NOTE: All temperature interlocks on recorders in Control Room
2. Inlet and exit temperature at each heat exchanger	CR, PP	
3. Temperature sensor in primary return header to control valve in secondary	None	
4. Return water line temperature	CR, PP	
5. Pool Water Temperature at: a. 1' below surface; b. 3' below surface; c. 20' below surface (for Δt); d. 20' below surface (for recorder)	CR	

TABLE 9-1 (Continued)

REACTOR PRIMARY WATER SYSTEM INSTRUMENTATION

FUNCTION	READ-OUT LOCATIONS	INTERLOCKS
Miscellaneous		
1. Pool level	None	1' below gutters 12' below gutters
2. Primary system water flow	PP, CR	Flow < 5500 gpm Flow < 2000 gpm

9.3.4 Reactor Secondary Water System

Table 9-2 on the following page lists those Secondary Water Systems parameters for which instrumentation must be provided. The temperatures to be displayed in the control room will be recorded on the multipoint recorder used for the primary water temperatures.

9.3.5 Reflector Tank Cooling and Pool Hot Layer Systems

The controls required for these systems were discussed in Sections 8.3 and 8.4 of this Report and will not be duplicated here. Instrumentation needed to measure critical system parameters is summarized on Table 9-3.

The temperature measurements will be displayed on a multipoint recorder in the control room along with the temperatures from the auxiliary systems described below. This recorder will be fitted with an interlock at 140° F.

9.3.6 Auxiliary Systems

Table 9-4 lists the instrumentation requirements for the Auxiliary Process Water Systems. All temperature displays will be on the multipoint recorder (140° F interlock) noted in 9.3.5 above.

Specific instrumentation for the degasifier equipment is not included in Table 9-4 because the final design of this equipment is still in question.

The automatic regeneration controls and instrumentation for the Cold Deionizer will be provided with the equipment and need not be stipulated here.

TABLE 9-2

REACTOR SECONDARY WATER SYSTEM INSTRUMENTATION

FUNCTION	READ-OUT LOCATION	INTERLOCKS
1. Throttling valve controlled by temperature sensor in primary water return header (3000-5000 gpm)	None	Valve full open
2. Inlet and exit heat exchanger pressure	LR, PP, CR	Exit pressure
3. Inlet and exit heat exchanger water temperature	CR	None
4. Δp across each heat exchanger	PP	None
5. Sump level control in use at 2 Mw satisfactory, but must increase water line size for fill line	CR (exist.)	None
6. Existing chemical treatment system and controls satisfactory. Output rates will have to be adjusted by Plant Department personnel	None	None
7. Inlet and exit pressures at secondary water pump	L	None

TABLE 9-3

REFLECTOR TANK COOLING AND HOT LAYER INSTRUMENTATION

FUNCTION	READ-OUT LOCATIONS	INTERLOCKS
Pressure Measurements		
1. Inlet and exit pressure on hot layer pump	L	None
2. Inlet and exit pressure on hot layer deionizer	L	None
3. D_2O pump exit pressure	LR	None
4. D_2O pressure in return line	L	None
5. D_2O tank static pressure	L (D_2O panel)	$p > 2$ psig
6. Inlet and exit pressure on hot layer filter	L	None
Temperature Measurement		
1. D_2O inlet and exit temperatures	CR	NOTE: All temperature interlocks on recorders in Control Room
2. Pool temperature 1' and 3' below pool surface	CR	
3. Hot layer deionizer inlet temperature	CR	
4. Hot layer heat exchanger exit temperature to distribution header	CR	
Miscellaneous		
1. D_2O flow rate	CR } Panel Meters Only	No flow
2. Hot layer deionizer flow rate	CR } Panel Meters Only	
3. Hot layer make-up tank water addition (integrating water meter)	L, PP	
4. Hot layer deionizer inlet and exit conductivity	PP, CR	

TABLE 9-4

AUXILIARY WATER SYSTEM INSTRUMENTATION

FUNCTION	READ-OUT LOCATION	INTERLOCKS
Hot Deionizer System		
1. Inlet and exit pressure on hot deionizer pump, DI tank, and DI filter	L	None
2. Inlet water temperature to hot deionizer	CR	None
3. Hot deionizer flow rate (ribbon meter)	PP, CR	None
4. Hot deionizer inlet and exit conductivity	PP, CR	None
Cold Deionizer System		
1. Exit water conductivity	CR, L	Resist. < 250,000
2. Storage tank level	CR, L	Vol. < 500 gallons
Experimental Facility Cooling System		
1. Flow meters at each beam port	L	None
2. Cooling system header pressure	L	None

9.3.7 Emergency System

Table 9-5 reviews the instrumentation and interlocks to be provided for the Emergency Water Systems.

9.4 Radiation Detection

This section will review the permanently installed radiation detection instrumentation to be used in the FNR. The existing area radiation monitors (NMC) and moving tape air particulate monitors (Tracerlab) have been quite satisfactory. When the conversion to 10 Mw occurs, it may be worthwhile to continue using these manufacturer's items. The final decision will be reached with the assistance of the University's Radiation Control Service.

Table 9-6 reviews the areas and monitor types to be provided for 10 Mw operation.

Four emergency monitors should also be installed to provide information in the event of a major reactor accident. They should be located on the east walkway of the pool floor, the control room, the second floor corridor, and the beam hole floor office area. These monitors are fixed units which can be removed and analyzed at a later date (e.g., "criticality balls").

9.5 Ventilation System Instrumentation

Table 9-7 lists those ventilation system parameters for which instrumentation must be provided. The systems for which instrumentation is needed include:

TABLE 9-5

EMERGENCY WATER SYSTEM INSTRUMENTATION

FUNCTION	READ-OUT LOCATION	INTERLOCKS
1. City water line pressure	PP, L (2 places), CR	$p < 50$ psig
2. Core spray system header	L, CR	$p < 40.$ psig
3. Core spray flow	CR, Lobby	When manual or auto valve open
4. Anti-siphon valve open	CR, Lobby	When valve not fully closed

TABLE 9-6

RADIATION MONITORING INSTRUMENTATION

CONDITION	TYPE INSTRUMENT	MONITOR LOCATIONS
Direct radiation levels	Wide range units (1 mr/hr - 5 R/hr) with local read-out and alarm and signal output to recorder in control room and remote interlock connection	1. Hot DI work area 2. Beam hole floor areas 3. Both sumps 4. Exhaust air plenum 5. Hot exhaust duct 6. Heat exchanger area 7. Second delay tank area 8. Primary pump area 9. Reactor bridge 10. South end of reactor pool 11. Hot layer DI 12. Reactor control room 13. 2 units on pool floor walls
Direct radiation levels	High range unit (1 - 100 R/hr) or above - to separate recorder in control room	1. Core exit water line
Airborne activity	Units presently monitoring beam hole floor area and pool floor area but with recorders located in the reactor control room and the PML Lobby.	

TABLE 9-7

VENTILATION SYSTEM INSTRUMENTATION

FUNCTION	READ-OUT LOCATION	INTERLOCKS
Normal Ventilation System		
1. Supply and exhaust fans operating	L, CR	Whenever either is not in operation
2. Supply filter bank Δp	L	None
3. Main Building dampers	L, CR, Lobby	Status lights
Hot Off Gas System		
1. Lab exhaust fan operating	L, CR	Whenever not in operation
2. Standby lab exhaust fan operating	L, CR	When in operation
3. Exhaust filter bank Δp	L, CR	Alarm setpoint to be determined
4. Exhaust filter bank radiation level	L, CR	To be determined
5. Status of isolation damper	L, CR, Lobby	Status lights to show open or closed
Emergency Ventilation System		
1. Emergency exhaust fan operating	L, Lobby	Interlock
2. Unit Δp	L, Lobby	setpoints
3. High temperature alarm	L, Lobby	are yet
4. System radiation level	L, Lobby	to be
5. Exit air activity	L, Lobby	determined

1. Normal ventilation system.

2. Hot off-gas system.

3. Emergency ventilation system.

Those controls and instruments which are necessary to the operation of the heating and ventilation system with no special safety or operational significance are left to the discretion of the architect-engineer with final design responsibility and are not included here.

Figure 53 shows the conditions which will initiate the Reactor Building Emergency Sequence which was discussed in Section 4 of this Report.

9.6 Reactivity Control

9.6.1 General Mode of Operation

As mentioned previously, the core excess reactivity requirements have been estimated to be between 9 and 11% $\Delta K/K$. For the type of shim-safety rod (SSR) to be used on the FNR, an average rod reactivity worth of approximately 2.5% $\Delta K/K$ per rod is expected. (This is based on current FNR experience as well as the experience of other facilities.)

The FNR will use a total of 6 shim-safety rod assemblies (total worth approximately 15% $\Delta K/K$). With this arrangement, it is estimated that the fully loaded, cold, clean reactor core will achieve criticality with the SSR's withdrawn between 3 - 45% from their fully inserted position. An operational core (equilibrium Samarium, higher pool temperature, equilibrium low cross section fission products, and less than 2% ΔK in-core experiments) will achieve criticality with the SSR's withdrawn even further. As a general rule, the reactor will be operated with balanced SSR positions except for the shim-safety rod which is used for power level control.

The reactor will be able to use any one of the six SSR's as a control rod. Whenever a SSR is so used, limit switches will restrict the region of travel of the rod under automatic control to that rod movement with a reactivity worth of less than 0.6% $\Delta K/K$. By the use of these dual-purpose SSR assemblies, a valuable core position (in terms of reactivity worth) need not be wasted on a low worth, non-scrammable control rod as is currently the case at 2 Mw.

Reactivity control during reactor start-up will be achieved by limiting the reactivity addition rate from the shim-safety rods to less than 0.1% $\Delta K/K$ per second. This criterion has been factored into the rod drive mechanism design so that gang withdrawal of all 6 shim-safety rods will not exceed this rate limit for any rod position.

Reactivity control during start-up and other manipulations will also be limited by rod withdrawal inhibit interlocks which will prevent rod withdrawal when certain core, process equipment, or reactor instrumentation conditions exist.

Four methods of reactor power reduction will be available:

1. Set back: auto control system reduced power level to 5 Mw.
2. Slow run-in: Shim-safety rods insert at slow speed.

3. Fast run-in: rods insert at high speed.

4. Scram: the release of the shim-safety rods from their electromagnets.

The mechanisms, controls, setpoints, and logic involved in achieving each of these types of reactivity control will be treated in the following paragraphs.

9.6.2 Reactivity Control Devices

9.6.2.1 Shim-Safety Rod

The shim-safety rod design to be used in the FNR is shown in Figure 54. The evaluation work done thus far for this rod design is summarized in Reference 37.

If this design does not prove suitable, a nickel plated Ag-In-Cd rod design is commercially available for use. One 5 Mw reactor (Union Carbide) has recently converted to this design and, thus far, finds it satisfactory. The two other available designs (boron-stainless steel and B_4C powder filled units) are not recommended by this author for 10 Mw operation.

9.6.2.2 Rod Drive Mechanism

Figures 55 through 59 are assembly drawings of the two speed, dual purpose (safety and control) rod drive mechanism to be used at 10 Mw. These units are currently being tested (out-of-core) by the operations group.

The rod drive mechanism design provides the following design features:

1. Electromagnet, magnet contact switch, and rod seat switch all out of water.

2. Two rod insertion speeds (only 1 withdrawal speed) for increased safety and operational flexibility).

3. Easier placement of shim-safety rod assemblies in various core positions.

4. Any one of the drive mechanisms can be chosen to act as a regulating rod drive.

If testing of this unit proves satisfactory, one or more units may be used on the FNR at 2 Mw to obtain operating reliability data.

The mechanism also has, as a safety feature, the insertion of the shim-safety rod in the event that the flexible shaft shears, or a shearing of any power shaft or power transmission gear or worm. Under these conditions, the shim-safety rod will be inserted into the core within 10 seconds of the time of the failure.

Testing of various shock absorber designs is presently underway to reduce the forces experienced by the system during free-fall of the shim-safety rod following a scram. This work on shock absorbers is considered an integral part of the evaluation of the drive mechanism.

At the present time, the following general performance is expected from the drive mechanism arrangement shown on Figure 55.

Rod drive withdrawal rate:	3"/min.
Slow insert rate:	3"/min.
Fast insert rate:	9"/min.
Magnet release time:	< 10 ms.
Magnet drop time:	< 400 ms.

9.6.3 Rod Withdrawal Inhibit Functions

A rod withdrawal inhibit is an interlock which prevents the application of power to the withdraw motor of the rod drive mechanism. The conditions which will actuate this interlock are shown on Figure 60. Note that the control arrangement is such that only one workable LCR circuit is required for rod movement. Moreover, once the Log N channel registers a power level in excess of 1 KW ("Log N Confidence"), the LCR channel interlocks are bypassed.

When the rod selector switch is in the "unload" position, a slow scram will occur. This position is used for maintenance on the drive mechanisms and the scram prevents shim-safety rod withdrawal.

Note that as shown in the diagram, the primary function of the inhibit circuit is to limit the rate at which reactor power level is changed.

The reason for inhibiting rod movement for a reactor period shorter than 30 seconds is so that an operator cannot compensate quickly for any sudden, unexplained reduction in reactor power by withdrawing the six shim-safety rods.

9.6.4 Reactor Setback Functions

The "Reactor Setback" is accomplished by inserting the shim-safety rod used as a control rod until a power level of 50% of the servo setpoint on the Linear N channel is reached. The reactor control system will then maintain reactor power at that level. For normal, full power operation, this will cause a reduction in power from 10 Mw to 5 Mw.

The conditions for which this setback will occur are shown on Figure 61.

The "Experiment Panel" indicated on Figure 61 refers to an interlock system which will be provided for those experiments which, if experiment operation difficulties arose, could be damaged or pose personnel or reactor hazards if 10 Mw operation were to continue.

9.6.5 Rod Run-in Functions

9.6.5.1 General

A rod run-in (also referred to as a "reverse" or an "auto-rundown") is an automatically initiated insertion of all shim-safety rods by energizing the insertion windings of the rod drive mechanism motors. In a run-in, the rods are not released from their shim-safety rods.

There will be an "auto-rundown" feature in the FNR, and this is used to describe the automatic insertion of a drive mechanism at high speed whenever one of the shim-safety rods has released. This is an operational convenience and is not included as a safety feature.

Rod run-in can be accomplished at two rates. The low speed run-in is used to correct relatively minor operating parameter anomalies and the rod insertion and attendant reduction in power is quickly correctable once the condition which initiated the run-in is corrected.

The high-speed run-in will be used to correct more serious difficulties and the higher speed of insertion will require a longer time to achieve power level recovery. The insertion speed for the high-speed run-in is three times the slow speed run-in rate.

Any run-in condition automatically takes precedence over a rod withdrawal request.

9.6.5.2 Low-Speed Run-in

The conditions which will actuate a slow-speed rod run-in are shown on Figure 62. It should be noted that the slow-speed run-in is the same rod speed as the withdrawal rate.

9.6.5.3 High-Speed Run-In

The high-speed rod run-in is used as a back-up to the low-speed run-in interlock for those abnormal nuclear conditions which were not suitably corrected by the low-speed function. Those conditions are shown on Figure 63.

9.6.6 Rod Scram Functions

9.6.6.1 General

Rod scram functions are those conditions which cause the release of all of the shim-safety rods from their electromagnets and, consequently, result in rapid shutdown of the reactor. In the FNR, there will be two types of scrams:

1. Slow Scram: Process system parameters which cause interruption of the supply A.C. power to the Magnet Current Power Supply shown in Figure 52. The shim-safety rods will be released from their electromagnets within 50 milliseconds of the time of the power interruptions.

2. Fast Scram: Nuclear parameters which cause rod release by acting directly through the Safety System components as shown on Figure 52. The shim-safety rods will be released within 20 milliseconds of the time any of the trip levels are reached.

9.6.6.2 Slow Scrams

The various process conditions which will initiate a slow scram are shown on Figure 64.

9.6.6.3 Fast Scrams

Fast scrams will occur for any of the following conditions:

1. A power level of 125% or more as shown by any of the three (3) neutron sensitive level ionization chambers.

2. A power of 125% or more as shown by the gamma level ionization chamber.

3. A Log N Period indication of 5 seconds or less.

4. A Log Gamma Period indication of 5 seconds or less.

9.6.7 Rod Drive Mechanism Control Circuit

The interrelation of the various safety and control functions described above as they apply to the control circuits for one of the shim-safety rod drive mechanisms is shown in Figure 65.

The following system constraints should be noted:

1. The reactor can be operated in AUTO or MANUAL mode.

2. Only one of the six shim-safety rods can be selected to operate under automatic control from the servo control system.

3. All six shim-safety rods can be moved simultaneously when the reactor is in MANUAL operating mode. Selection of one rod for auto control and operation of the reactor in AUTO mode disconnects that selected rod from control by the gang rod switch.

4. A rod selector switch determines which of the six shim-safety rods can be moved individually by the gang switch. The options are all rods (see 3 above) or each rod individually. Other combinations will not be available.

5. High speed insertion of a singular drive mechanism occurs whenever a shim-safety rod releases from its magnet and reaches its fully inserted position (SEAT). This automatic, singular drive rundown feature can be bypassed, if desired, for certain operational circumstances.

6. The AUTO mode of operation for the reactor once established, will remain in effect providing the following conditions are met:

   a. No slow or fast scrams occur.
   b. No rod is released from its magnet.
   c. No run-ins (low or high speed).

   The set back is accomplished by the auto control system and therefore is not one of the disabling mechanisms for AUTO operation.

10. SHIELDING REQUIREMENTS

10.1 General

This section of the Report will discuss the current estimates of radiation levels at various locations in the FNR process systems. This will serve as the basis for the shielding calculations which will be performed during the final detail design engineering work yet to be done for 10 Mw operation of the FNR. While shielding calculations have not been performed, the experience obtained at other reactor facilities of similar power level (5-40 Mw) indicates that there is sufficient space available in the reactor basement to build adequate equipment radiation shields. For most cases, concrete block (solid) walls less than 18 inches thick should be satisfactory. However, this will have to be verified.

10.2 Reactor Biological Shield

In discussing the adequacy of the existing FNR Biological Shield for operation at 10 Mw, it is convenient to treat three separate planes of reference. The biological shield consists of:

1. Horizontal plane: pool water plus reactor pool walls.
2. Upper vertical plane: pool water over core.
3. Lower vertical plane: pool water and reactor base.

Each of these should be considered separately, but based on the information currently available, it has been concluded that the biological shield will be adequate at 10 Mw.

The horizontal shielding arrangement consists of not less than 4 feet of pool water and 6 feet of high density concrete. In terms of conventional radiation protection instrumentation, it is not possible to detect any evidence of direct radiation penetration of this shield at 2 Mw. A Nuclear Engineering graduate student (Reference 43) performed some measurements which showed a shield penetration radiation level of less than 0.03 mr/hr. If this level is correct, the level at 10 Mw should be approximately 0.2 mr/hr. Based on current experience in the beam hole floor area, this rate (0.2 mr/hr.) is less than the background levels which are anticipated in that area from experiment operations.

Moreover, a comparison of the construction of the shield with reactors in the 10 Mw power region suggests that the FNR shield will be adequate at 10 Mw. The matter of heat energy input to the pool walls will be treated in another part of this section when discussing the need for thermal shields (10.4 below).

As regards the shield adequacy in the upper vertical plane, Section 15 of this Report indicates that the direct radiation level from the FNR core at 10 Mw will be approximately 10 mr/hr. to the head region when standing at the pool railing. While this is higher than might be desired, it will be possible to limit doses received by personnel. This is discussed in greater detail in Section 15.1 (Routine Operational Safety).

The lower vertical plane of the biological shield is not of any concern as regards higher power operation. The only region under the reactor pool walls which is presently accessible to personnel is where the first delay tank is located. In the 10 Mw design, this region will not be accessible. As shown in Figure 47, that area will be behind the shielding walls provided for the primary pumps. As was shown on Figure 64, opening of the access door to the primary pump area will initiate an automatic shutdown of the reactor. The direct radiation from the core through the pool floor into the region of the first delay tank is estimated to be approximately 5 R/hr. compared to 50 R/hr. from the core exit water line and the first delay tank.

10.3 Reactor Basement Shielding

The major radionuclide in the primary water system for which shielding will have to be provided is the N^{16} (N.1 second half-life, 7 Mev gamma) formed by the O^{16} (n,p) N^{16} reaction as the primary water passes through the core. The second isotope which contributes to the radiation level will be the Na^{24} (15 hr. half-life, 2.75, 1.37 Mev gammas) dispersed in the pool water. This is of much less consequence in terms of shielding requirements for delay tanks, pumps, and heat exchangers than the N^{16}. It is the major contributor to the radiation level from the deionizer tanks.

The equilibrium Na^{24} level in the primary water is expected to be less than 0.1 microcuries/cc. This estimate is based on the present experience at 2 Mw.

Figure 66 shows the expected radiation levels at various points in the primary water circuit in the reactor basement. The location of these points can be seen in Figure 47. This curve (Figure 66) was originally developed to estimate the delay tank requirements for 10 Mw operation.

Figure 67 shows the concrete thicknesses (normal concrete: density = 150 lbs./ft.^{3}) of the beam hole floor over various portions of the reactor basement. The adequacy of this floor to provide radiation protection to personnel on the beam hole floor will have to be verified as well.

Using the information on Figure 66, and noting the placement of equipment as shown on Figure 47, the following shielding requirements can be outlined:

1. Primary Pump Area Shield: Three pumps, each reading approximately 15 R/hr. (N^{16}) and the east end of the first delay tank reading approximately 20 R/hr. must be shielded by a wall sufficient to reduce the gamma field at the wall to less than 10 mr/hr.

2. Second Delay Tank Shield: Two tanks (42" dia. x 24' long) each reading a maximum of approximately 15 R/hr. (N^{16}) to be shielded to 10 mr/hr. on outside face of shield. Beam hole floor radiation level to be less than 0.1 mr/hr. from tanks.

3. The inlet water line to the heat exchanger is expected to be about 250 mr/hr. Shielding of the heat exchangers can be deferred until operational experience is obtained. However, the adequacy of the beam hole floor thickness of 33 inches to reduce the 250 mr/hr. (N^{16}) to 0.1 mr/hr. must be verified during the design period.

4. The shielding for the two "hot DI" units should be capable of reducing the dose rate of an 18 inch diameter by 3 foot high cylinder from 150 R/hr. (contact on cylinder) to 10 mr/hr. The major radionuclide will be Na^{24}.

10.4 Reactor Thermal Shields

As was mentioned in Sections 5, 7, and 8 of this Report, the need for a thermal shield mounted on the reactor support structure will have to be evaluated. There are two major points to consider:

1. If a shield is required to reduce the energy input to the reactor pool walls to a sufficiently low level to assure that no pool concrete damage will occur, and/or

2. If a shield is required to reduce the radiation levels at the joint between each beam tube embedment and the beam tube extension to a point such that the rubber gasket used to seal the joint will give a service life of not less than 5 years. Presently, a service life of 10 years has been experienced.

10.5 Emergency Ventilation System Shield

The shielding requirements for the Emergency Ventilation System filter units was outlined in Section 4.3.2 of this Report. Those requirements should be referred to and will not be repeated here.

11. FACILITY UTILITY SYSTEMS

11.1 General

This section will concern itself with requirements of the various facility utility systems. Specific layouts of these systems will not be attempted since these can best be done by the A/E during the final design stages. The purpose here is to delineate system requirements.

The systems to be covered include:

1. Electrical Supply Power and Distribution.
2. Water Supply.
3. Compressed Air Supply.
4. Steam Supply.
5. Communications and Alarm Systems.

11.2 Electrical Supply Power and Distribution

11.2.1 General Features

The Supply Electrical Power System will consist of a normal power supply provided by Detroit Edison Company through new switchgear and transformer to be located in the existing switchgear room in the Phoenix Memorial Laboratory. The power should be supplied as:

a. 440 Vac, 3 phase for all new rotating equipment motors which are larger than 1/2 horsepower.
b. 220 Vac, 3 phase for existing equipment which will continue to be used at 10 Mw (Crane, etc.).
c. 117 Vac, 3 phase for lighting circuits, etc., currently in existence.

The existing distribution panels located in the FNR basement should be replaced with new equipment located in the motor control center shown in Figure 47. All control circuits used in conjunction with this control center should be 117 Vac instead of the 220 Vac systems now in use.

As a further safety precaution, in the event of a large water loss from the reactor pools as described in Section 15, all the motor starters, distribution breakers, and fuses located in the control center should be not less than 30 inches off the floor. In this way, any large water loss would not result in a loss of electrical power to the facility until after the pool drained and the core sprays had been in operation for more than 20 minutes. This would prevent any delay, because of a loss of building lighting, of a building evacuation during an emergency.

11.2.2 Emergency Electrical Power

An emergency electrical power supply such as a gasoline engine/generator set should be provided to supply critical components with power in the event of a failure of the power supplied by Detroit Edison. The needs for emergency power are not so stringent for this 10 Mw design that "failure free" power is required. The emergency generator supply should be able to start-up and assume electrical load within 10 seconds after a power failure. However, the ability to start reliably is of greater concern than this 10 second delay period. The recommended starting time of 10 seconds is based on convenience considerations rather than specific need.

The following items must be supplied by the emergency power system:

1. The stack exhaust fans of the Phoenix Memorial Lab.

2. The reactor instrumentation systems but not the drive mechanisms.

3. The ventilation fan used for the Emergency Vent System if a separate one is used in addition to the PML exhaust fans.

4. The radiation monitoring systems including:

   a. All NMC radiation monitors
   b. All airborne radioactivity monitors
   c. The building evacuation alarm system

5. The existing "night light" circuits in the Reactor and PML Buildings.

6. The intercom and public address systems.

7. All electrical controls associated with the Emergency Ventilation System.

8. The controls for the core spray system.

9. Building Air Compressor.

10. The door status control circuits and Indicator Panel.

The following equipment need not be provided with emergency power, but there are circumstances where the ability to operate this equipment would be helpful. If generator capacity is available, the following equipment should receive emergency power:

1. 1000 gpm shutdown cooling pump.

2. Hot demineralizer pump.

3. A series of 20 ampere capacity emergency power outlets located at various locations throughout the building for use in emergencies.

11.3 Water Supply

The existing city water supply to the reactor facility will have to be checked for adequacy during the final design phases for the 10 Mw conversion. The main lines feeding the building are probably adequate but these, along with the distribution systems inside the building, will have to be checked.

The primary water system requirement is that 200 gpm be deliverable through the core spray system in the event of an accident involving loss of pool water. This capacity should be verified under the following conditions:

1. Full core spray flow of 200 gpm.
2. City water pressure at minimum value (use 10 year minimum value except for periods of complete pressure loss).
3. Cooling tower sprinkler system operating at full flow as for a fire.
4. Cooling tower make-up water line full open and delivering water at maximum flow rate.

If the core spray system or building water supply system cannot meet the above requirements, then the supply will have to be revised or additional control systems and isolation valves installed to limit the non-essential use of city water during emergencies. A simple line diagram of the major portions of the supply water system is shown on Figure 68.

11.4 Compressed Air Supply

The existing compressed air supply system for the facility should be revised to separate it from the system used by PML. The new compressor unit should be installed in the Auxiliary Building as was shown on Figure 12. The essential features of the system are shown on Figure 69.

The air system will be supplied primarily by the building compressor. A portion of the compressor output will be passed through dryer and oil separator equipment to provide dry, oil-free air for instrumentation. This instrument air will be distributed throughout the building at receiver pressure and at 30 psig. A portion of the low pressure system will be separated out for distribution to critical, pneumatically operated facility systems. That portion of the low pressure system will be "backed up" by a dry nitrogen system which will automatically maintain the header pressure at 25 psig in the event of a compressor breakdown.

11.5 Steam Supply

A steam supply system for the facility should be provided either by using the existing steam supply to the facility from the Cooley Laboratory boiler system or by installing a small packaged boiler in the PML equipment rooms.

Steam is required at three locations:

1. Pool floor area to supplement D_2O heat input to hot layer system during lower power operation (below 10 Mw).
2. Reactor basement area for steam cleaning of heat exchanger tubes and operation of the air eductors in the degasifier system.
3. Liquid waste handling area behind hot cells (see Section 12).

The quantity of steam which will be required is not known, nor the required steam temperature and working pressures. These will have to be determined after further design work is completed on the above mentioned systems.

11.6 Communications and Alarm Systems

The present arrangement of communication devices (telephones, intercoms, etc.) are satisfactory for the time being. The intercom units should be replaced by newer devices but location changes need not be made. The building alarm systems, except for those modifications which are recommended elsewhere in this Report, are satisfactory for 10 Mw operation. Replacement of the alarm system units is not necessary.

12. LIQUID RADIOACTIVE WASTE DISPOSAL

12.1 General

The matter of the handling and disposal of liquid radioactive wastes at the reactor facility requires some attention especially when a power level increase to 10 Mw is contemplated. The matters of gaseous and solid radioactive waste disposal are more straightforward than that of liquid waste disposal and will be summarized in Section 15.1.5, General Public Safety (Routine Reactor Operation).

The present method of handling the disposal of liquid wastes will be reviewed, some estimates of future expected levels will be discussed, and a possible revision to the liquid waste handling system will be presented. At this point, however, it must be stressed that possible waste handling system revisions should receive a comprehensive review by Radiation Control Service and the Radiation Policy Committee. The system to be described here is one believed to have certain operational advantages (high decontamination factor, reliability, ease of operation, etc.) but is not the only way to treat the problem.

12.2 Present Liquid Waste Handling System

The following tabulation shows the amount of activity discharged from the present liquid waste handling system. During this four year period, the reactor was operated on the continuous operating schedule.

The present liquid waste handling procedure consists of collecting all liquid wastes in a series of three stainless steel storage tanks (3000 gallons each). Once any tank is full and the contained activity has decayed sufficiently that the short-lived radionuclide

TABLE 12-1

LIQUID WASTE DISCHARGES 1967-1970

Year	Gallons of Liquid Waste Discharged	Millicuries of Tritium Released	Millicuries of Other Activity Released
1970	141,000	295	80
1969	94,000	343	82
1968	58,000	185	50
1967	80,000	192	55

activity such as Sodium-24 from regeneration of the hot deionizers is negligible, chemicals are added to that tank to produce a suspension that settles to the bottom of the tank as a sludge. The settling of this suspension removes between 60 and 80 percent of the suspended activity from the storage tank contents (not including the tritium activity in the water). This sludge is then removed from the tank, dried, and disposed of as solid waste.

The remaining activity in the liquid remaining in the storage tank is disposed of by diluting with city water as may be required to meet 10CFR20 limits, and releasing it to the building sanitary drain system.

The 15 inch sanitary sewer line which passes alongside the Phoenix Memorial Laboratory and FNR is one of the main sanitary sewer drain lines for North Campus. This line leaves the North Campus area at the Fuller Road - Bonisteel Boulevard intersection area as shown on Figure 70. It follows Fuller Road to a point near the Penn Central RR tracks as they pass over the Huron River. This section of the main has also been fed from other feeder drains.

The sewer line contents are then pumped up the hill on the south side of the river to the main 36 inch drain which carries the sanitary waste from the central part of the city to the city sewage treatment plant. The data on Figure 70 about the city sewage treatment plant was current as of January, 1971. However, plans are currently being considered by the city to expand the operations of that plant. During the final design phases of this 10 Mw project, the sewage treatment plant data will have to be updated.

12.3 Anticipated Liquid Wastes

During 1970, the total quantity of liquid wastes processed by the existing liquid waste system was 141,000 gallons as shown on Table 12-1. This was believed to be unusually high and efforts will be taken to reduce this quantity in future years. However, a figure of 100,000 gallons per year is not considered to be excessive. This is believed to be distributed as follows:

1.	Non-evaporative pool water loss	37,000 gal/year
2.	Hot deionizer recharges	10,000 gal/year
3.	Cold deionizer recharges	15,000 gal/year
4.	Packing leakage (pump, valve, etc.)	5,000 gal/year
5.	Miscellaneous other waste water	33,000 gal/year
	Total	100,000 gal/year

At a power level of 10 Mw, a number of these quantities may change; moreover, for those items of waste water which involve the reactor pool water an increase of the specific activity pool water contaminants of five times the present levels is expected.

The following is an estimate of the quantity of contaminated liquid wastes to be handled by the liquid waste system when the FNR operates at a power level of 10 Mw. This list assumes that a leakproof liner is not installed in the pool but that mechanical seals are used on the new primary and secondary circulation pumps.

1.	Non-evaporative pool water losses	37,000 gal/year
2.	Hot deionizer recharges	30,000 gal/year
3.	Packing leakage	10,000 gal/year
4.	Miscellaneous other waste water	50,000 gal/year
	Total	127,000 gal/year

Thus, at 10 Mw, we do not anticipate a marked increase in the quantity of liquid waste to be handled. However, we do expect a significant increase in the radionuclide concentration in the waste water. This increased quantity of radioactivity will have to be handled either by repeated applications of the techniques currently used to process waste or by installation of new waste handling equipment, as described in the following portion of this section.

12.4 Proposed Waste Handling System

Figure 71 is a simplified layout of a proposed liquid waste handling system. It is based on the use of a reduced pressure, low temperature evaporator to concentrate moderate level wastes for solidification and disposal as solid wastes.

Liquid wastes would be held in any of the three existing 3000 gallon stainless steel hold tanks. In order to improve sampling accuracy, 500 gallons of waste would be transferred to the Sample Tank. If the level were low enough to discharge, the waste water could be fed directly to the Condensate Discharge Pump for release to the sanitary sewer. City water could also be added if some minimal dilution was required.

If the water in the Sample Tank is sufficiently contaminated, it would be fed to the evaporator (evaporator capacity of 20-40 gallons per hour should be sufficient to handle that portion of waste water which needs to be processed).

Low temperature boiling would be induced in the evaporator by a steam or compressed air eductor. The eductor discharge would be condensed and collected in the Condensate Hold Tank for sampling and activity determination. If the activity is sufficiently low, the condensate will be discharged to the sanitary sewer. If not, it will be recycled to the Feed Tank for further treatment. This equipment should be capable of decontamination factors (for other than volatiles and tritium) of 10^3 - 10^6.

The evaporator contents would periodically be released to the Concentrate Hold Tank for eventual solidification using plaster of paris, cement, or other solidifying agents. The solid material would then be shipped to an approved burial site for ultimate disposal.

13. ADMINISTRATION

13.1 General

This section will discuss certain administrative aspects of operating the FNR at a power level of 10 Mw. This will include a discussion of changes which are recommended to the facility administrative structure, but will also deal with such administrative matters as:

1. Operational expenditures.
2. Reactor operational cycles.
3. Operations crew training.
4. Operations reviews.
5. Facility security and emergency planning.

The discussions of these topics will, in the main, be limited to summarizing those areas to be evaluated during the final design and construction periods prior to higher power operation.

13.2 Facility Administration and Staff

Figure 72 shows the Administrative Structure recommended for the Phoenix Project when the FNR power level is increased to 10 Mw.

The plan is based on administratively separating the operations/services functions from the grant program/project finances functions by having two Associate Project Directors, each of whom report in turn to the Director of the Project.

All business office functions (accounting, wage/salary records, expenses, invoicing, etc.) would be put under a Business Office Manager who would report as shown on Figure 72.

The Reactor and Laboratory Managers would report to the second Associate Director, as would a Staff Engineer. This new position of Staff Engineer will be needed to handle technical and experimental evaluation problems pertinent to 10 Mw operation of the type presently handled by the Reactor Manager. However, because of their increased complexity and the additional operational demands on the Reactor Manager created by the more complex 10 Mw facility design, he will no longer be able to perform this dual function.

The Associate Director for operations will have the dual responsibility of maintaining the operational status of all Project facilities as well as increasing the over-all utilization of the facility by faculty, staff, students, and outside universities and industries. It is recommended that this individual could also serve as the responsible Project staff member during the construction phases of the modifications to the FNR for 10 Mw operation.

Figure 72 also shows that the recommended operational staff (licensed reactor operators) consists of 15 people. Twelve of these would be assigned to routine shift duties (four shifts; three operators per shift) with the other three used for special duties including vacation and sickness shift relief for the operating crews.

The current crew size for an operating shift on the FNR is two men: a Lead Operator and a Reactor Operator. A survey of research reactors in the power level range of 4-10 Mw was conducted in 1968. It was found that the reactors operating at power levels in excess of 5 Mw had a minimum crew size of three men. It will be assumed that three men will be necessary for adequate staffing. It is believed that two licensed operators and one trainee would constitute a satisfactory "minimum" crew composition.

13.3 Operational Expenditures

Based on the organizational structure shown on Figure 72 and the wage, salary, and equipment expenditure rates currently experienced by the reactor and laboratory staffs, the following is an estimate of operational costs for the reactor only:

TABLE 13-1

ESTIMATED REACTOR OPERATING COSTS

Staff Salaries 12 shift operators, 3 day operators, 3 administrative personnel, 1 staff engineer, 1 secretary	$ 211,000
Supplies and Services	30,000
Capital Equipment	12,000
Insurance	39,000
Utilities	54,000
Staff Benefits	22,000
Building Maintenance	21,000
Est. Total Operating Costs	$389,000

In addition, the costs of operation of the Laboratory Areas would be expected to increase to some degree:

TABLE 13-2

ESTIMATED LABORATORY OPERATING COSTS

Staff Salaries	$ 116,000
2 administrative personnel, 2 lab technicians, 2 instrument makers, 1 electronics engineer, 1 electronics technician, 1 secretary, hourly assistance wages	
Supplies and Services	22,000
Capital Equipment	10,000
Staff Benefits	12,000
Building Maintenance	38,000
	$ 198,000

The Business Office activities should not cost more than $21,000 per year to maintain.

Thus, the overall annual expenditures for 10 Mw operation of the reactor should be, in terms of 1971 dollars:

Reactor Operation	$ 389,000
Laboratory Operation	198,000
Business Office	21,000
Total	$ 608,000

This total does not include depreciation or Phoenix Project administrative costs represented by the salaries, expenses, and clerical assistance required by the Director and two Associate Directors of the Project. However, the total does include those items (building maintenance, insurance, etc.) which would not normally be entries on unit operating budgets.

13.4 Reactor Operational Cycles

The present plans for reactor operation are based on maintaining the present operating cycle of 28 days. At the present time this consists of 25 days of power operation followed by 3 days of shutdown maintenance and calibration.

The expected excess reactivity which can be controlled satisfactorily in the 10 Mw design by the six shim-safety rod assemblies is such that a 28 day cycle should be still possible at 10 Mw, but it will be necessary to perform a mid-cycle shutdown (approximately 15 days after start of a cycle) to refuel the core for an additional 10 days of operation. This mid-cycle refueling shutdown should not require more than four hours.

The above estimate of 15 days of operation before the mid-cycle shutdown would be variable depending on the in-core experiment load (in terms of experiment reactivity worth).. Because of this possible variation in the reloading period, it is recommended that this reload period not be used by experimenters for experiment changes or low power tests. These should still be restricted to the shutdown period at the end of a cycle.

13.5 Operations Crew Training

Training of the crews which will operate the reactor at 10 Mw will require considerably more time than is presently experienced at 2 Mw. This increased time will be primarily concerned with the various reactor systems and operating procedures. It is estimated that an additional 16 weeks of training will be required for operators with no previous nuclear systems experience (presently 20 weeks) while 8 weeks additional will be needed for those with prior experience.

The facility crew training experience and needs are summarized in Table 13-3.

Since the facility has experienced an average operator turnover rate of approximately 25% over the past five years (1966 thru 1970), it is expected that training of approximately four new operators each year will be required. Since retraining is the administrative responsibility of the Assistant Reactor Manager, operator training will occupy a significant fraction of his available time (20% estimated).

13.6 Operations Reviews

Operations Reviews refers to the procedures to be followed to enable the review of the reactor operation over a period by knowledgeable people not directly associated with facility operation. At the present time, this is achieved through the Reactor Advisory Committee. This committee has a membership of faculty and staff who are knowledgeable in various technical disciplines. The committee acts as an advisory body to the Reactor Manager, who acts as its chairman, on all matters pertinent to the operation of the facility. The committee, if requested by the Reactor Manager, reviews and advises on limitations pertinent to experiments to be run in the reactor.

TABLE 13-3

CREW TRAINING PERIOD REQUIREMENTS

AREA	EXPERIENCED OPERATOR	INEXPERIENCED TRAINEE
Nuclear Theory	2 Weeks	8 Weeks
Systems Principles	1 Week	4 Weeks
System Procedures	3 Weeks	6 Weeks
Operating Experience		
Reactor Controls	3 Weeks	6 Weeks
Support Equipment	6 Weeks	10 Weeks
License Restrictions, Federal Regulations, Admin. Proc., etc.	2 Weeks	3 Weeks
Health Physics	1 Week	4 Weeks
Total	18 Weeks	41 Weeks

Notes: 1. Weeks required based on period from employment until examination for AEC License.

2. Training period consists of formal instruction, practice, and study for approximately 4 hours per day. Employee assigned to routine activities for remainder of day.

By its own action, the committee has agreed on a set of rules of practice which require that:

1. The Reactor Supervisor reviews reactor operation with them not less than twice per year.

2. Each month, one of the members takes an extended tour of the reactor with the Reactor Supervisor, the purpose of which is to critically review operations, procedures, equipment operation, facility work conditions, etc.

Each member of the committee is appointed by the Vice President for Research for a three year term. These terms are overlapping to permit continuity.

It is recommended that this committee continue to perform as described above when the reactor is operating at 10 Mw. However, it is also recommended that a reactor supervisor from another reactor of comparable power level be invited, at least once each year, to visit the facility for two or three days to prepare a critical review of FNR operations. This report would then be presented to, and discussed with, the Reactor Advisory Committee for any action deemed suitable.

13.7 Facility Security and Emergency Planning

Section 15 of this Report deals with the safety evaluation of the FNR operating at a power level of 10 Mw. The major conclusions reached are:

1. Routine operation at 10 Mw offers no hazard to the health and safety of the general public, including visitors to the reactor facility.

2. In the event of a Design Basis accident occurring, the emergency equipment provided in the facility is sufficient to assure that the North Campus of the University meets the criteria of 10CFR100.

Since these are the conclusions reached, and because the issues of facility security and emergency planning are periodically reviewed, it is recommended that:

1. The matter of facility security not be treated as an issue of special concern because of 10 Mw operation. Those areas of security precautions (unauthorized entry, student activism, pilfering, etc.) which are regularly reviewed because of the present 2 Mw power level and general university security needs will be suitable at the higher power level as well.

2. Existing Emergency Plans will be broad enough to provide the necessary nucleus of an emergency organization suitable for 10 Mw operation. When the documents requesting the necessary license amendments to permit modification of the facility are being prepared, then the specific details of the emergency plan which are affected by the higher FNR power level can be reviewed and modified. Section 15 of this Report discusses the potential need of an alternate control point for emergencies in the event of the Design Basis accident. A suitable location would be either the Cooley Laboratory or the Lobby of the IST Building. The Cooley Building would be preferable since it provides a tunnel connection to the Reactor Building. This is primarily an area of Emergency Plan re-evaluation to be dealt with during the period of time devoted to formulating the PSAR and FSAR documents submitted to the AEC.

14. CONSTRUCTION CONSIDERATIONS

14.1 General

This section is concerned with the planning, scheduling and construction aspects of the conversion of the FNR to 10 Mw. For the purposes of this section, the "construction period" will cover the time period from the decision to go ahead with power upgrading to the time the reactor operates one full cycle at 10 Mw. This section will cover such topics as:

1. Construction Costs.
2. Scheduling.
3. Manpower Assignments.
4. Special Safety Considerations.
5. Quality Assurance.
6. Performance Testing and Low Power Testing.
7. Approach to Full Power.

14.2 Construction Costs

The estimate of construction costs necessary to complete the modifications described in previous sections of this Report is presented in Table 14-1.

This estimate is based on:

1. The use of 1969 dollars to estimate costs.
2. The AEC not requiring piping, vessels, and other primary system components to meet requirements of the ASME Pressure Vessel Code for Nuclear Vessels. (This requirement would add approximately $150,000 to the overall construction costs.)

TABLE 14-1

10 Mw CONVERSION CAPITAL COSTS*

Additional process heat removal equipment	$ 295,000
Control system improvements	115,000
Operational equipment	140,000
Containment improvement and other engineering safeguards	80,000
Building improvements	265,000
Engineering costs	105,000
Estimated Capital Cost	$ 1,000,000

*This presumes the adequacy of the basic building. If major structural changes to the building are required, this total may increase by as much as $500,000.

Based on the above, the overall cost of conversion of the facility to 10 Mw operation will range from a minimum $800,000, if central air conditioning is excluded from the construction, to a maximum of $1,600,000 if major structural changes and the ASME Nuclear Code requirements are indicated.

14.3 Scheduling

Figure 73 is a preliminary estimate of an overall project schedule to achieve 10 Mw operation within 30-36 months of the time of a decision to go ahead with power uprating of the FNR. Approximately six months of preliminary additional work would be required to gather the final data upon which to base a decision. This final data is primarily updated cost estimates and any major facility design revisions prompted by the preliminary reactions of AEC staff to the conversion.

14.4 Manpower Assignments

Section 13.2 of this Report suggested that the Associate Director for Operations could serve as overall project manager for the 10 Mw conversion period. In this way he would serve as a focal point of document preparation, cooperating with the Architect/Engineer (A/E), and any consultants employed in addition to coordinating the efforts of the members of the operating organizations.

In this context; the reference to coordinating the efforts of the A/E implies that he will work with whatever organization is given final design responsibility. This might be the University Plant Extension group or an outside A/E firm retained by them. In any event, the designation A/E refers to whichever of these organizations is assigned design responsibility.

Because of the multi-faceted nature of the design problems associated with 10 Mw uprating and operation, it is recommended that the manpower assignments for various phases of the project be distributed as shown in Figure 74.

14.5 Special Safety Considerations

Since much of the construction schedule shown on Figure 73 will take place while the reactor is operating and, furthermore, a portion of the work needed to be done will take place within the reactor pools, some attention must be given to the safety controls which will have to be instituted.

These special controls will have to be used during the + 12 to + 24 month period shown on Figure 73. The + 12 to + 18 month period will consist, primarily, of routine verification that construction of the Auxiliary Building does not:

1. Compromise existing building containment integrity.
2. Interrupt supply services to the facility.
3. Result in workers performing work in areas where they may receive radiation exposures in excess of the limits in 10CFR20. (This is unlikely during Auxiliary Building construction but will have to be verified.)

The + 18 to + 24 month period is the one which requires the greatest surveillance effort. During this period, the Reactor Advisory Committee and the University's Radiation Control Service will be heavily involved in both planning and day-to-day considerations.

The Reactor Advisory Committee (RAC) will be asked to assist in the planning relative to:

1. The storage of irradiated fuel in areas other than the reactor pool.

2. The disposal of portions of the reactor system which will no longer be used:

   a. Reactor tower
   b. Reactor grid plate, plenum, and header
   c. Various handling equipment
   d. Reactor thru ports

3. The removal and temporary storage of components to be reused:

   a. Radial beam ports
   b. Heavy Water Reflector Tank
   c. Shim-Safety rods

4. The review of safety procedures, test programs, operations procedures, AEC documents and transmittals, equipment evaluations and performance tests, and other preparatory and evaluation matters.

Radiation Control Service (RCS), in addition to their involvement in the above matters both through and in addition to the considerations by the RAC, will have to provide periodic health physics surveillance of personnel and work areas, especially in the basement and reactor pool locations. This will include survey work as well as personnel monitoring as may be required.

All installation of core and other in-pool components will be left to the operations group, thus reducing, to a small extent, the workload on RCS personnel. However, as mentioned in Section 6.6 of this Report, it may be necessary to consider the installation of a stainless steel liner in the reactor pool to replace (cover) the existing ceramic tile liner. Such an installation would have to be performed by contractor personnel, and the radiation control measures increased proportionately.

14.6 Quality Assurance

It will be the responsibility of the Associate Director for Operations during the design phases to collate all areas of code requirements applicable to the FNR 10 Mw conversion, all performance criteria upon which AEC licensure was based, and all other matters for which quality assurance or performance was required or inferred.

It will then be his responsibility to collect the necessary supporting data, certifications, and/or test reports to support the assurance claims. It is not possible at this point to identify specific requirements since much of this will be based on the requirements imposed by the AEC during the preliminary evaluation phase and their evaluation of the PSAR. It will also be based, to a great extent, on any self-imposed restrictions arrived at during the final design phase (- 2 to + 9 month period on Figure 73).

14.7 Performance Testing and Low Power Testing

These periods are those concerned with evaluating the process and support system performance as well as determination of the basic nuclear characteristics of the modified system.

The performance testing phase is intended to verify that all systems are performing satisfactorily so that fuel loading can be initiated. These tests will include:

1. Process and Auxiliary Systems Testing:

   a. Core total flow and primary system force analysis.
   b. Core pressure drop vs. flow.
   c. Core flow distribution patterns.
   d. Total system flow rates and pressure drops.

      (1) Primary
      (2) Secondary
      (3) D_2O
      (4) Hot Layer
      (5) Hot DI
      (6) Shutdown

   e. Flow and system safety interlocks operative.
   f. Heat exchanger tube rattling tests.
   g. Purification system performances.
   h. Pool water makeup system performance.
   i. Experimental facility cooling system performance.
   j. Pool water degasifier performance.

2. Emergency Water Systems:

   a. Isolation valves to storage tank - reliability and closure time.
   b. Core spray reliability, opening time, and volume delivery.
   c. Anti-siphon valve operability.

3. Instrumentation and Control:

   a. Operability of individual components.
   b. System interlocks.

      (1) Process systems
      (2) Radiation detection systems
      (3) Ventilation systems
      (4) Reactivity control systems

   c. Calibration of nuclear and process instrumentation.

4. Ventilation and Containment:

    a. Normal operation flows, pressure drops, air distribution patterns, etc.
    b. Emergency vent system operability.

        (1) Initiating controls
        (2) Flow rate controls
        (3) Changeover controls (fire safety)
        (4) Containment pressure as a function of outside weather conditions

Each of these system performance tests will require a test procedure prior to test execution and an evaluation report upon completion of the tests.

The satisfactory conclusion of the test program generally outlined above along with an operating license from the AEC will permit the start of low power testing of the reactor. This will consist of the following general test areas:

1. Initial critical loading of the reactor.

2. Reactivity determinations relative to first loading:

    a. Rod worths.
    b. Reactivity coefficients (temperature, void, etc.).

3. Loading of first operational core.

4. Reactivity measurements on first operational core:

    a. Shim-safety rod worths and reactivity insertion rates.
    b. Fuel-position reactivity worth.
    c. Reactivity coefficients.

        (1) Moderator temperature
        (2) Reflector temperature
        (3) Moderator void
        (4) Reflector void
        (5) Experiment facility void

5. Instrumentation responses to power changes.

6. Core flux mapping and experimental facilities.

7. Fuel element hydraulic characteristics evaluation.

14.8 Approach to Full Power

With the conclusion of the low power testing program, the approach to full power operation will begin. This consists of the stepwise increase in operating power level up to the design power of 10 Mw. During each phase of this power escalation, it is necessary to check the following areas:

1. Verify conservatism of thermal/hydraulic evaluation of core.

2. Verify adequacy of cooling system for Heavy Water Reflector Tank.

3. Verify adequacy of biological shielding of reactor and shielding added in pump room.

4. Verify gaseous activity (e.g. A-41) evolution rates and degasifier performance.

5. Verify performance of all portions of reactor safety and power level monitoring systems.

6. Verify adequacy of thermal shield cooling.

Since some time will be required to perform these evaluations, it is recommended that after each power level escalation, the new power level be maintained for at least one operating cycle. Although this is not necessarily required, it might be worthwhile to use a cycle to collect fuel burn-up data, reactivity coefficients (Xenon, burn-up, etc.)

and to monitor generally the performance of the system over an extended period. If the system performance is satisfactory, power would be increased again for the next cycle. Such a program might be as follows:

Cycle	Power Level
1	1 Mw
2	2 Mw
3	4 Mw
4	6 Mw
5	8 Mw
6	9 Mw
7	10 Mw

Using a program such as outlined above, the escalation program might involve as much as six to seven months.

15. SAFETY ANALYSIS

15.1 Routine Operational Safety

15.1.1 General

The safety aspects of routine operation of the reactor at 10 Mw will be analyzed. The following items will be treated:

1. Facility visitor safety.
2. Facility experiment safety.
3. Operational personnel safety.
4. General public safety.

15.1.2 Facility Visitor Safety

In order that facility visitors can continue to view operation of the FNR at the pool floor level, after the operating power level is increased to 10 Mw, the sources of radiation to which visitors are exposed during a visit must be evaluated and any necessary steps taken to keep visitor exposures below established limits.

The radiation sources to which visitors to the pool floor area are routinely exposed are listed below:

1. Airborne Argon-41 produced activation of air dissolved in the pool water and its evolution from the pool surface into the building air.
2. Direct radiation from the reactor core through the pool water.
3. Direct radiation from the Sodium-24 dispersed in the pool water.

There are other sources of airborne activity as well as other radionuclides dispersed in the water system, but none of these are comparable in magnitude to those enumerated above.

Figure 75 is an estimate of the Argon-41 concentration in the pool floor area at a power level of 10 Mw. The pool floor area is the region to which visitors are brought to view the reactor when it is in operation. This figure also indicates the Argon-41 concentrations at the present power level of 2 Mw -- the information upon which the 10 Mw predictions are based.

AEC regulations (10CFR20) require that persons in restricted areas (areas to which access is controlled by a licensee) shall not be exposed to airborne Argon-41 concentration in excess of 2×10^{-6} microcuries per cc. Moreover, minors under 18 shall not be exposed to concentrations in excess of 10% of that level. From the data presented on Figure 75, it is clear that some provision must be made to reduce the radioargon concentrations on the pool floor at a power level of 10 Mw if visitors are to be allowed in the facility.

There is appreciable variation of the rate of argon evolution from the pool as a function pool temperature. This variation is also shown in Figure 75. The pool floor area argon concentration will exceed the permissible level for occupational exposure with the reactor operating at 10 Mw since the pool will be maintained at a bulk pool temperature of approximately 100° F.

Methods which can be used to reduce the argon concentration in the pool floor area include:

1. Installation of a degasifier in the pool water system.

2. Installation of a pool cover equipped with an air sweep system which will ventilate the area under that pool cover.

3. An increase in the building air turnover rate in the pool floor area.

Previous portions of this Report which describe the general facility modifications noted that both a degasifier and a pool cover would be installed with a pool air sweep system attached to the building exhaust duct work.

No final decisions have been made with regard to the manner in which the ventilation system will be changed to accommodate the air-conditioning equipment recommended for the facility. These ventilation systems changes may very well include modifications which would increase the air turnover rate in the pool floor area of the building. However, even if an increase in air turnover rate is not experienced, the installation of the degasifier and the pool cover should adequately control the Argon-41 problem and permit continued visits to the facility by the general public and university staff.

Visitors to the facility are also exposed to the direct radiation level from the operating core and the Sodium-24, dispersed throughout the primary water system, when they stand at the pool railing.

The following table summarizes the effect of increased power level on the radiation dose rates received by facility visitors standing at the pool rail. The table also shows the effect of increasing the size of the demineralizer system which removes reactor pool water impurities.

TABLE 15-1

MAXIMUM FACILITY VISITOR DOSE RATES

(At Pool Railing)

CONDITIONS	DOSE RATE FOR HEAD REGION	DOSE RATE FOR GONAD REGION
1. Power Level = 2 M (present conditions)		
a. Dose rate from core	1.4 mr/hr.	.8 mr/hr.
b. Dose rate from Na^{24}	5.6 mr/hr.	3.2 mr/hr.
Total	7.0 mr/hr.	4.0 mr/hr.
2. Power Level = 10 M (no changes in demineralizer rate; remains @ 20 gpm)		
a. Dose rate from core	9.5 mr/hr.	5.5 mr/hr.
b. Dose rate from Na^{24}	34.0 mr/hr.	19.0 mr/hr.
Total	43.5 mr/hr.	24.5 mr/hr.
3. Power Level = 10 M (demineralizer flow rate = 100 gpm)		
a. Dose rate from core	9.5 mr/hr.	5.5 mr/hr.
b. Dose rate from Na^{24}	27.0 mr/hr.	16.0 mr/hr.
Total	36.5 mr/hr.	21.5 mr/hr.

As can be seen from the table, the increase in demineralizer flow rate from the present value of 20 gpm to 100 gpm for 10 Mw operation would reduce the total dose rate in the head region from 43.5 mr/hr. to 36.5 mr/hr. In order to reduce the Sodium-24 concentration to the present 2 Mw values by means of a demineralizer, a unit with a capacity of approximately 4,000 gpm would be required. This is economically prohibitive and is not recommended.

The economically prohibitive nature of utilizing a large demineralizer to reduce the contaminant levels in the reactor pool was the basis for the installation of the hot water layer scheme described in Section 8.4 of this Report. Briefly, the hot water layer scheme involves floating a water layer which is several degrees warmer than the bulk pool water and free of radioactivity on the pool surface. This layer then acts as a shield against the radiation from the material dispersed in the pool water below the layer. A hot water layer approximately two feet thick will reduce the Sodium-24 dose rate at the head region at a power level of 2 Mw to 3.4 mr/hr. Visitors at the pool rail of the facility will then be exposed to total radiation as shown below for a reactor power of 10 Mw:

Head Region	13 mr/hr.
Gonad Region	7.5 mr/hr.

Thus the use of a hot water layer for a FNR power level of 10 Mw will limit dose rates to facility visitors to less than twice the present values. Thus restrictions on permitting the public to visit the facility during operation at 10 Mw should not be necessary.

15.1.3 Facility Experimenter Safety.

The effect of 10 Mw operation of the FNR on the safety of experimental personnel within the facility includes not only those areas considered in the discussion of the facility visitor, but also a treatment of the types of experimental programs being conducted by the research personnel. The major areas of research activity involve the use of beam ports in the beam port floor area, activation analysis programs, and the irradiation of miscellaneous types of sources within the core.

No specific recommendations are made regarding changes to the beam port facilities. It is believed that the current biological shields in use at these facilities will be adequate for 10 Mw operation with the addition of limited amounts of local shielding in high radiation areas. Considerations apart from personnel safety may well dictate a need to make major shield changes.

The present collimator designs at all of the ports will have to be re-evaluated for operation at 10 Mw but many of them should prove adequate for 10 Mw operation.

The effect of 10 Mw operation on personnel utilizing the pneumatic facilities and receiving samples which have been irradiated in the core will have to be evaluated separately. The following general comments apply:

1. Many of the sources currently produced in the reactor will, because of the higher flux, have to be removed from the pool by means other than the current practice. This will require greater use of the transfer chute to Hot Cave No. 2. Possibly, installation of a hydraulic or pneumatic shuttle system between Cave No. 2 and the reactor core with an intermediate loading station somewhere in the reactor pool will be required.

2. Until experience is gained by both experimenters and operational personnel, it will be necessary to make greater use of RCS personnel in the review and surveillance of experiments. As such experience is gained, it will be possible to institute appropriate operational and experimental procedures for the handling of reactor produced samples.

15.1.4 Operational Personnel Safety

Operations personnel routinely work in areas where they are exposed to radiation. When working in the area of reactor pool floor these personnel are exposed to the airborne activity released from the pool surface, the direct radiation from the reactor core through the reactor pool, and the Sodium-24 dispersed in the reactor pool water. These three sources of radiation were discussed previously when dealing with the safety of visitors to the facility. However, in many instances, operational personnel are exposed to higher dose rates than visitors since the handling of equipment requires that they not restrict themselves to the area of the pool railing. Because of the "hot water layer" to be used in the reactor pool, Operations personnel working immediately above the pool surface will be exposed to a gamma dose rate of less than 40 mr/hr. This level will not be restrictive as regards routine operation and further efforts to reduce this dose rate are not considered necessary. It should be noted that Operations personnel working in this area are now exposed to 15 mr/hr. with the reactor at 2 Mw.

The methods previously described for reducing the airborne activity level in the pool floor area are also considered adequate for Operations personnel safety.

Another situation which results in Operations personnel radiation exposures in the area of the pool floor is the handling of irradiated samples and their removal from the reactor pool. At this time, it is not possible to detail the precautionary measures which must be taken to reduce these exposures. In general, as experience is gained with handling the higher activity level targets, operational procedures will be instituted to minimize personnel exposures.

This same technique of instituting adequate operational procedures also applies to the work areas around the experimental equipment on the beam hole floor. When sufficient shield modifications are completed to make the experimental area satisfactory for research personnel, these areas will also be adequately controlled for operational personnel.

The largest contributor to Operations personnel radiation exposures is the work areas in the reactor basement. The present radiation levels (2 Mw operation) in this area are produced primarily by the bypass deionizer which maintains primary water coolant purity and the delay tank (1400 gal.) which is used to delay passage of the primary water to the coolant pumps. This delay allows for the decay of the reactor produced Nitrogen-16 (7 sec. half-life).

Figure 47 showed the equipment layout for the basement of the Reactor Building. Note that shielding is to be added around the primary pumps and delay tanks. If necessary, additional shielding can be located around the heat exchangers, should they prove to be a significant source of radiation. While it is not specifically indicated on the equipment plan, the deionizer equipment will also be shielded, but no decision has been reached as to whether to shield each tank individually or to enclose the entire system behind a shielding wall.

In addition to the use of shielding to reduce the radiation exposures of personnel entering the basement during reactor operation, it should be noted that a read-out panel indicating process equipment operating parameters (flows, temperatures, radiation levels, etc.) will be located at the base of the stairway to the reactor basement. This panel will reduce the number of times that an operator need actually enter the inner regions of the reactor basement.

It is necessary to locate the primary pumps behind shielding walls because the radiation level from the existing 1400 gallon delay tank located beneath the reactor foundation will be 15 to 20 R per hour. Thus the primary circulation pumps for the reactor system will be handling relatively high radiation level primary coolant as regards personnel exposure, and access to these pumps must be prohibited during periods of operation. Of course these pumps will be fully accessible during periods of shutdown through the motor operated lead shielded door, which will be located on rails mounted on the basement floor as shown on Figure 47. The additional delay tank volume which will be installed in the discharge lines from the primary pumps should be adequate to reduce radiation levels in the general room area to acceptable levels. As is shown on Figure 67, the thickness of the floor slab over the primary pump area is a minimum of 42 inches of normal concrete. Unfortunately, a portion of the floor thickness in the area of the two additional delay tanks which will be added to the system is only 12 inches. It is for this reason that removable shielding will be installed above the two delay tanks as shown in Figure 47. Based on this general arrangement of equipment and additional shielding in the reactor basement, personnel should have sufficient protection from process system radiation hazards so that these hazards are not a limiting consideration in the evaluation of higher power level operation for the Ford Nuclear Reactor.

15.1.5 General Public Safety

The effect of routine operation of the FNR at a power level of 10 Mw on the safety of the general public outside of the reactor facility is discussed below. The following four areas will be treated:

1. Direct radiation levels from the Reactor Building.
2. Airborne radioactive contamination from the Reactor Building air exhausts.
3. Radioactive contamination of the building liquid effluents.
4. The disposal of solid radioactive waste materials.

The problem of the control of the direct radiation levels to which personnel and facility visitors will be exposed during routine operation of the FNR has been treated in the appropriate portions of this Report. The control measures outlined in these sections (hot water layer, increased beam port shielding, and primary water system shielding, etc.) were shown to be adequate to reduce the radiation level within the Reactor Building to permissible values. These measures, combined with the shielding provided by the concrete walls of the Reactor Building, will limit radiation levels external to the Reactor Building to allowable values. Thus, the routine operation of the FNR at a power level of 10 Mw would pose no direct radiation hazard to the general public.

At the present time, the primary radioactive material in the airborne effluent from the Reactor Building is Argon-41. This material is produced by activation of the normal argon fraction of air present in the pneumatic tube systems as well as that dissolved in the primary water system. This Argon-41 is released from the Reactor Building by two mechanisms.

Any Argon-41 produced either in the pneumatic tube systems or in the reactor beam ports is vented from the building via the facility stacks. After passing through roughing and absolute filters, the dilution factor currently in use for the stacks is 400.

The Argon-41 which is released from the reactor pool into the building volume is exhausted along with the building exhaust ventilation air through the exhaust air plenum of the reactor building into the area of the building cooling towers. At this point, fresh air drawn in by the cooling towers dilutes the air from the reactor building before it is released through the cooling tower exhaust structure. The amount of diluting air provided by the operating of the cooling towers is, of course, a function of the number of towers in operation. In addition to this fresh air dilution, a dilution factor of 4 is used to determine the average concentration of radioactive nuclides released through the cooling tower exhaust to the outside environment.

At the present power level of 2 Mw, the average concentration of Argon-41 in the exhaust air from the facility stacks is less than 10% of the allowable concentration. Operating of the Ford Nuclear Reactor at a power level of 10 Mw would increase this concentration to approximately 50% of the allowable level. Therefore, no increase in stack dilution factor will be required.

Figure 75, which was used in the discussion of the Argon-41 concentrations in the pool floor area as regarded facility visitor and operational personnel safety, also indicated the anticipated building exhaust plenum Argon-41 concentration at a power level of 10 Mw as well as that value appropriate to the current 2 Mw operation. Those measures which will limit the release of Argon-41 into the Reactor Building itself have already been described. This Argon-41 collected by means of the pool air sweep would be channeled to the outside environment through the normal building ventilation system. The exhaust air will be sent to the intake air grills of the cooling towers mounted on the building roof where it again will be diluted by cooling air drawn in by the cooling tower fans during their operation. Because of the much larger size of the cooling towers needed for 10 Mw operation, it is estimated that approximately four times as much cooling air will be required by the new cooling tower units compared to each of the existing units. As a result of this, the net increase in Argon-41 concentration levels being released from the facility cooling tower exhausts at 10 Mw is estimated at 20% higher than the current 2 Mw values.

The foregoing assumes that no additional sources of Argon-41 would be produced as a result of higher power operation. This is not necessarily true. Neutron and gamma heating of experimental facilities such as the reactor beam ports or in-core experimental equipment may require the use of a coolant such as air. At this time, it is not possible to estimate the exact form that such a cooling system would take, and its effect on Argon-41 release rates.

However, in the event that such cooling systems are required, it is possible to take steps such as the use of recirculating systems or the use of gases other than air to limit the Argon-41 production rate. At this time it appears reasonable to assume that such measures could be successfully adopted and that 10 Mw operation of the FNR will not pose unsolvable problems as regards the release of airborne radioactivity. Therefore, an increase in the dilution factor for the facility stacks or from the ventilation system exhaust does not appear to be necessary.

The matter of disposing of liquid radioactive wastes was treated in some detail in Section 13 of this Report. It concluded that the matter of liquid waste disposal requires further study but that a satisfactory disposition of the problem is possible.

The disposal of solid wastes from the reactor facility has never posed any significant degree of difficulty at 2 Mw operation. We have at all times been able to package and adequately shield all solid radioactive materials and ship them to an approved burial site. During 1967, the low level solid waste amounted to approximately 220 cubic feet. The high level solid waste was about 8 curies and was shipped in disposable shields weighing approximately 4 tons. There is no reason to believe that this procedure cannot be continued successfully even at a power level of 10 Mw. Therefore, disposal of solid radioactive waste should pose no hazard to the general public as a consequence of higher power operation of the FNR.

It is reasonable to conclude that the routine operation of the FNR at a power level of up to 10 Mw would not pose any safety hazard to the general public.

It should be noted that operation of the reactor will have no thermal effects on the bodies of water in the vicinity of the reactor facility. This is because waste heat is rejected to the atmosphere by means of the cooling towers located on the roof of the building. None of the heat generated in the facility by reactor operation is returned directly to the Huron River or any other body of water in the vicinity of this site.

15.2 Credible Operational Accidents

15.2.1 General

The following paragraphs will discuss the various types of operational accidents which are believed to be credible for a reactor facility of the FNR type. Several of the types of operational accidents to be discussed have actually occurred at various facilities throughout the United States and elsewhere.

During the course of the discussion of the magnitude of the release of radioactive materials from the reactor core which might accompany any one of these various credible operational accidents, reference will be made to three accident classifications (I, II, III). This use of a classification of the magnitude of the accident will simplify the discussion of the radiological consequences of the various credible accidents.

The classification system used is as follows:

Class I Accident:

An operational accident which causes the release of 10% of the total gaseous fission product inventory of the reactor core with no fission iodine or particulate material released as a consequence of the accident.

Class II Accident:

An operational accident which causes the release of 20% of the gaseous fission product activity in the reactor core and 5% of the fission product iodine. None of the particulate fission products are released from the reactor core.

Class III Accident:

An operational accident which causes the release of 25% of the fission product iodine, 50% of the fission product gaseous activity, and 1% of the particulate fission product activity in the core.

The following general comment applies to all of the above accidents:

1. Reactor power level is considered to be 10 Mw.

2. Reactor core burn-up level is considered to be 20% for the core.

3. Reactor core mass is assumed to be 5 kilograms of U^{235}.

4. The dilution factor for the facility stacks is assumed to be a factor of 100.

5. No time delay is assumed between the time of the incident and the release of radioactive material.

Since the consequences of an accident in the release of radioactivity from the building may occur over a fairly short period of time, e.g., one to five days, the additional factor of 4, which is permitted in the calculation of average concentrations of effluents from the facility stacks and is based on wind direction variability, has not been utilized.

The emergency ventilation system, whose operation is critical to the evaluation of the radiological consequences of accidents of all types in the reactor facility, has been described elsewhere in this Report (Section 4). The basic operation of this equipment during the course of an incident within the facility will be briefly reviewed.

In the event of the release of radioactivity into the Reactor Building, the emergency ventilation system is started either manually by push buttons on the reactor console, or by automatic equipment located at various points through the reactor facility (exhaust air plenum, bridge radiation monitor, storage vault alarm, etc.). When the emergency ventilation system operates, the building is sealed off by means of closure of the main intake and exhaust air dampers of the building ventilation system, while all doors are kept closed and sealed (normal operational requirement). A portion of the building ventilation air is drawn through a series of high-efficiency particulate filters and filters designed for the removal of halogens. It is assumed that the filtration system has no effect on the concentration of gaseous fission products (krypton, xenon). After this fraction of the building air leaves the filters, it is exhausted out of the facility exhaust stacks for dispersal into the outside atmosphere.

15.2.2 Reactor Power Transients

15.2.2.1 SPERT IV Results

Analysis of the transient response of the FNR to reactivity inputs is primarily based upon the information which has been gathered during the SPERT tests. Most specifically, the data accumulated during the SPERT IV sequence was the basis for evaluation since these tests were conducted using a reactor core having nuclear characteristics very similar to the proposed FNR 10 Mw design and were conducted with a pool depth of 18 feet. Much of the SPERT IV data pertinent to this study is summarized in Reference 22. Several of the more significant points are summarized here.

The majority of the information available from the SPERT IV test programs is presented in the form of graphs plotted against reciprocal reactor period as the independent variable. The reciprocal reactor period is the inverse of the shortest reactor period experienced during the transient response and is related to the amount of reactivity added to the reactor as a "step increase" which initiates the reactor excursion. Figure 76 shows the expected range of inverse reactive periods for the FNR 10 Mw cores for various reactivity inputs to the core. This range of values exists because the inverse period is related not only to the reactivity input, but to the prompt neutron lifetime of the core as well as the effective delayed neutron fraction yield. Because the FNR at a power level of 10 Mw will use fuel of varying degrees of burnup (up to a maximum of 40% per element, 20% average for core), both the prompt neutron lifetime and the effective neutron delayed fraction will change as a function of burnup.

Figure 77 summarizes the SPERT IV results for the maximum power level and energy release obtained during a transient as a function of reciprocal period. The energy release information from the SPERT IV test results are the energy yield from the initiation of the transient to the time of the peak power level. This energy release is approximately 2/3 of the total energy release from the excursion. The results shown are those obtained with stagnant water conditions, i.e., no flow through the reactor core.

Figure 78 shows the effect of flow rate on the reactor power level stability following the transient. "Chugging" occurs as a result of flow rate through the reactor core. This "chugging" phenomenon suggests that reducing the primary coolant flow through the reactor core in the event of a transient to prevent "chugging" might be worthwhile. For the sake of comparison, a primary core flow rate of 6,000 gallons per minute will be equivalent to an average coolant channel water velocity of 8 to 13 feet per second, depending upon the number of elements in the FNR core.

This same figure also shows the energy release for various transients as a function of various head and flow rate conditions for the reactor core. It will be noted that the energy release is dependent upon flow rate much more so than the hydraulic head of the water above the core. The overall energy release is increased from 20 to 30% for flow rates pertinent to this evaluation.

Figure 79 shows the maximum observed fuel plate temperature in the SPERT IV cores as well as the maximum pressure pulse generated in the water immediately below the reactor core during various transients. The effect of flow rate on maximum fuel plate temperature is small, with maximum fuel plate temperatures being reduced less than 10% for a flow rate of 12 feet per second compared to stagnant conditions.

Reference 8, in the discussion on nuclear power excursions, indicated that the combined effects of pressure and temperature transients resulted in fuel plate buckling whenever the peak reactor power level during the transient exceeded 100 Mw. This is equivalent to an inverse period of approximately 40 reciprocal seconds. From Figure 76 for the FNR it can be seen the reactivity input to achieve this is 1 - 1-1/2% $\Delta K/K$.

In reviewing the more destructive tests of the SPERT D core in which a reciprocal period of 313 was achieved with a peak power level of 2,250 Mw and an energy release of 31 Mw seconds, approximately 35% of the core was melted. The core was destroyed along with its associated equipment and approximately 4% of the fission product inventory of the core was released. Analysis of the results indicated that 12% of the total energy during the excursion was released from a metal-water reaction with the molten aluminum components. Again, using Figure 76, it can be seen that a reactivity addition of 2-1/2 to 4% $\Delta K/K$ would be required to achieve this result in the FNR core. Furthermore, the amount of fission product release which would accompany such an excursion is approximately equivalent to the Class III accident which has been described above.

15.2.2.2 Building Overpressure From Transient

The analysis of the consequences of an excursion must consider whether or not the expulsion of this quantity of energy in the form of steam into the Reactor Building would sufficiently raise the building internal pressure to the point where the building structure would be affected or damage would occur to the isolating duct work, dampers, and emergency ventilation system. Figure 80 presents the results of a calculation to determine the maximum building overpressure which would occur as a consequence of a nuclear excursion inside the Reactor Building. It should be remembered that the building overpressure design is 0.5 psi. This calculation indicates that an excursion involving the release of 150 Mw-seconds of energy would increase the building overpressure to 0.5 psi which is the design capability of the building. This computation was based on the following conservative points of view:

1. Steam formation would instantaneously follow the nuclear excursion and all of the energy of the nuclear excursion would go into the formation of steam.

2. The steam would be rapidly ejected from the reactor core, rise to the pool's surface, and mix with the building atmosphere immediately condensing and transferring all of the energy of the excursion to the building air which would be heated adiabatically and increase the pressure within the building.

3. The fact that approximately one second will be required to affect closure of the building dampers, thus providing an inherent pressure relief system, has not been taken into account. It is presumed that the excursion occurs when the building is already sealed.

4. The effect of the operation of the emergency ventilation system in reducing building overpressure by removal of a fraction of the building contents has not been taken into account.

Thus, a nuclear excursion of up to 150 Mw-seconds should not be capable of damaging either the Reactor Building or the exhaust system components, both normal and emergency.

The remote possibility of an overpressure in the building of 0.5 psi is the basis for modifying the building drain systems to include deep traps of 2 feet of water at every main drain location. Thus, the main building drains will not be "blown dry" by any building overpressure resulting from an excursion.

15.2.2.3 Operator Error

Reactor excursion accidents can be the consequence of a number of combinations of operator error and mechanical/electrical system faults. In classifying the accident which could result from any of these causes, reference is again made to Figure 76 which indicated the range of reciprocal period responses which the reactor would evidence for various reactivity inputs.

It is possible that operator error or a sample failure could result in the introduction of positive reactivity to the reactor core. Since our present operating license limits the reactivity worth of any single experiment on one face of the core to 1.2%, it can be seen from Figure 76 that the mishandling of such an experiment would result in relatively minor damage to the reactor core; a Class I accident. While there are no plans to increase this individual experiment reactivity limit for experimental programs currently planned for the reactor facility, there is an overall limit of 2% for all experiments in the reactor core. If, by some unique set of circumstances, it were possible to remove, insert, or damage all experiments in the core in such a combination that the entire 2% limit is rapidly introduced to the reactor, it can be inferred that this combination of circumstances would yield a Class II accident in which fairly significant mechanical damage would be done to the reactor core components with some release of fission products. This 2% insertion, however, would not lead to complete destruction of the reactor core and a Class III accident.

Moreover, steps will be taken by means of operational procedures and regulations to prevent the introduction of positive reactivity to the core from sample mishandling. Because of the design of the reactor core components which have been described elsewhere in this Report (Sections 5 and 6), the reactor core is uncoupled from the horizontal beam ports so that experimental changes at the ports, including the flooding and draining of the ports, have no effect on the reactivity status of the reactor core. During 2 Mw operation with a similar heavy water tank, it was possible to flood and drain reactor beam ports routinely in order to permit maintenance on those units during periods of reactor operation.

Another means by which reactivity can be rapidly introduced to the reactor core would be the mishandling of fuel elements during loading or storage procedures. Measurements made at this reactor facility and elsewhere show that elements inadvertently added to the exterior surfaces of the reactor core would introduce a reactivity value of less than 2%. This reactivity addition, as mentioned before, would result in a Class II accident. If, however, a central core location was left open (no fuel element present), one could postulate the accidental dropping of a fuel element directly into the center location of the core. If one further postulated that such an accident could be credible during reactor operation, tests at the ORR using 200 gram fuel elements showed that the reactivity introduction for a central location could be as much as 5% $\Delta K/K$. From the information presented thus far, this would clearly initiate a Class III accident resulting in complete destruction of the reactor core and the release of fission products. Precautionary measures will be taken to preclude the possibility of such an accident, and these precautionary measures include:

1. No core location internal to the core outline will ever be left vacant. That is, any internal core experiment or radiation locations will contain an experimental holder or other device whose presence would prohibit the seating of a fuel element into that core location.

2. The handling of fuel in the area of the reactor tower will be specifically prohibited during periods of reactor operation. The accidental droppage of fuel element such that it falls across the top of the reactor core would be equivalent to the loading of an element against a reactor face. Thus, that possibility need not be treated separately. One should also remember that since the reactor pool will be covered with removable panels, the inadvertent handling of fuel is quite unlikely since fuel element handling operations will be complicated by the

need to remove such panels. A number of panels will have to be removed simultaneously in order to permit fuel to be moved from the storage rack to the reactor core location. In any event, that accident, however remote, if it were ever to occur, would still constitute a Class III accident which is of lesser consequence than the Design Basis Accident (DBA), to be discussed later.

Improper seating of a fuel element in its grid location could result in a reactor excursion. If an improperly seated fuel element were not detected prior to reactor startup, it is conceivable that the primary core flow rate and pressure drop would be sufficient during operation to dislodge the element and cause it to fully insert. This would be equivalent to introducing additional fuel into the reactor during operation. Depending upon the fuel element location and the degree with which it is not seated, a reactivity introduction of up to 3% is credible. Were the reactivity addition to be as high as 3%, a Class III accident would probably follow. For a 3% reactivity introduction to occur, an element would have to be in the central position of the core and protruding approximately 16 inches above the core. It is believed that this would be readily observable prior to startup and that this type of accident more reasonably is probably restricted to element misplacements of two or three inches. In those cases, the reactivity addition caused by their sudden seating would be less than 1/2% $\Delta K/K$. In order to limit the probability of such an accident, operational procedures will be instituted calling for routine inspection of the core both prior to startup and during operation. In spite of the existence of the pool cover panels, visibility for operational personnel to observe the conditions in the reactor core will continue to be sufficiently good that these procedures should successfully detect conditions of this type.

Another method by which reactivity could be rapidly introduced to the core through an operator error would be if an operator were to remove the drive mechanism power head assembly during reactor operation, grasp, by some means, the shim rod extension pole, and withdraw that extension pole high enough to remove the shim-safety rod from the reactor core region. In that instance, a reactivity addition of not more than 3.5% $\Delta K/K$ would occur, again resulting in a Class III accident. Since the drive mechanism equipment is never to be touched or maintained during periods of reactor operation, it would appear that such an incident would occur as a consequence of intentional sabotage. In any event, this intentional disruption of reactor facilities would produce a Class III accident.

If the reactor is operating in a convective flow mode, as during lower power experiments, the average water temperature within the reactor core from both fission product inventory decay heat and low power operation is higher than if the reactor were operating in a normal forced coolant flow mode. If an operator were to inadvertently start up the primary pump systems with the reactor operating this way, the warm water would be removed from the reactor core and be replaced rapidly with the cooler bulk water of the pool. Since the reactor has a negative temperature coefficient, this sudden cooling of the pool would result in an introduction of reactivity to the core. Based on some experience gained with the FNR, this reactivity introduction should be less than 0.3% $\Delta K/K$. Thus, this "cold water" accident would have trivial consequences compared to the reactivity accidents which have been discussed. It is recognized that this pump startup might also occur because of a fault in the electrical contactors of the starting controls for the pumps. In any event, the consequences would be the same.

It is possible to postulate a circumstance in which reactivity is added to the reactor core because of a fault in the control mechanism circuitry causing one or more of the shim-safety rods to start withdrawing in an uncontrolled manner. Since shim-safety rod drive mechanism speeds will be chosen to limit the maximum reactivity insertion rate from the shim-safety rods to a value not greater than .01% $\Delta K/K$ per second, it would take more than a minute and a half at this rate to introduce 1% reactivity to the reactor core. This should provide ample time for the facility operators to take corrective measures to stop rod withdrawal, to inititate a reactor scram, or to otherwise disable the improperly acting unit.

15.2.3 Reactor Core Coolant Blockage

Another credible accident which involves a release of radioactivity from the reactor core is the blockage of portions of the reactor core by foreign material resulting in flow starvation to one or more core components. This flow reduction may result in high core component temperatures, local boiling within the element, and cladding failure resulting in a fission product release. A number of such incidents have occurred at various reactor facilities. For example, the ETR experienced a fission product release when a lucite sight-box was left in the core tank and blocked flow to a number of assemblies. In another instance, gasket material was left within the core tank of the ORR, and the primary water flow brought the gasket down on top of a fuel element causing failure.

While these incidents often involve the complete blockage of flow through a given subassembly of the reactor core, the degree of blockage often is not sufficient to cause observable changes in the total flow rate through the reactor core. As a result, the steps taken to prevent such incidents must be procedural in nature since existing core instrumentation is not sufficiently sensitive to indicate the presence of flow restricting foreign matter. Admittedly, the use of temperature and coolant flow rate sensing devices in each core component would provide such information, but the installation of such equipment is financially prohibitive; moreover, based on the experience of other reactor facilities, the use of procedural controls should be adequate.

The preventive measures taken to preclude the possibility of a coolant flow blockage accident of the FNR will include:

1. Reactor core inspection prior to every startup.
2. Reactor core inspection periodically during operation.
3. All materials used in the area of the pool will be colored or marked in such a way that they are readily visible under water.
4. No maintenance will be permitted on the reactor bridge structure or over the core components during periods of reactor operation.
5. The existence of the pool cover system will preclude any major pool maintenance activities during periods of reactor operation.
6. The reactor pool will be cleaned regularly to prevent the build-up of debris.

In addition to these preventive measures, others will be instituted as needed to satisfy requirements of individual projects, work schedules, experiment modifications and other maintenance and/or experimental activities in the area of the reactor pools.

In discussing the probable consequences of an accident of the type described above, the published results of similar accidents at the ORR and Siloe' Reactors has been reviewed. In the cases of these two incidents, it was observed that not more than 50% of the fission product iodine in the damaged fuel element and approximately 1,000 curies of volatile radioactivity was released. This being the case, such a coolant blockage accident would not be more serious than the Class III accident which was described previously.

15.2.4 Loss of Primary Coolant Flow Rate

In discussion of this accident in which the primary coolant flow through the reactor core components is interrupted in some manner, we are not including the possibility of the drainage of the reactor pool since this will be treated in a subsequent paragraph.

A number of circumstances which could produce a loss of forced coolant through the reactor core include:

1. The simultaneous failure of both primary pumps in coincidence with a failure of the emergency pump to start.

2. Sudden closure of the main isolating valve between the reactor core and the first delay tank.

3. Sudden plugging of the primary coolant water line such as might occur by a collapsing failure of the rubber liner located within the primary piping system.

4. Failure of a primary gate valve stem which would allow the gate valve wedge to rapidly close, thus stopping main flow.

5. Electrical power failure shutting down main pumps and a simultaneous failure to transfer load to the emergency generator rendering the emergency circulating pump inoperable.

In discussing the probable consequences of a set of circumstances which would result in the loss of forced coolant flow rate for the reactor, it is necessary to ascertain whether or not the reactor would continue to be operating at its full power of 10 Mw at the time of the coolant loss.

As has been discussed in Section 9 (Instrumentation and Control), loss of the primary coolant pumps by electrical interruption, loss of primary coolant flow rate as detected by the orifice plate differential pressure transmitters by any mechanism, opening of the siphon-break valve in the primary coolant system, and/or loss of electric power to the facility should all result in the release of all shim-safety rods into the reactor core by separate actuating devices. In the event that all of these devices were to fail to shut down the reactor, then reactor operation might continue at full power until boiling was established in portions of the reactor core. At that time the temperature and void coefficients acting together would produce reactor shutdown. Any activity release which follows facility shutdown through core self-shutdown mechanisms should be adequately treated in the discussion of the Class III type of accident.

A more credible set of circumstances surrounding the loss of core primary coolant would be for reactor shutdown through control device actuation to follow the coolant flow loss. Then the reactor thermal behavior caused by the fission product heating during the period immediately following core coolant stoppage must be analyzed. An analysis of the transient thermal behavior of a shutdown reactor core following a loss of primary coolant has been conducted at a number of reactor facilities (Ground Test

Reactor, Lynchburg Test Reactor, Oak Ridge Research Reactor, and Air Force Test Reactor). All of these reactors are in the 5 - 40 Mw power range. The results of their analyses are summarized briefly below:

1. Tests with the Ground Test Reactor (Reference 24) included simulated loss of flow accidents with the GTR at 10 Mw. The experimental instrumentation did not detect the presence of boiling in any of the tests. In addition, the maximum fuel surface temperatures were not great enough to support boiling in the transient or level operation tests.

2. Tests at the Air Force Nuclear Engineering Test Facility (AFNETF) summarized in Reference 25 demonstrated that the prompt maximum fuel plate temperature during low flow scram tests would not exceed 240° when the steady state power level prior to loss of flow was 12 Mw. The AFNETF is a reactor which utilizes a coolant flow rate of 10 fps through the fuel plate region and an inlet water temperature of 100° F. The plans for the FNR include a range of coolant flow rates of 8 to 13 fps with a core inlet temperature under normal operation of not more than 100° F.

3. Reference 26 which summarizes test work performed on the Oak Ridge Research Reactor revealed that detectable boiling would not occur in the ORR following a shutdown from operation at power levels up to 17 Mw even if no forced convection cooling were supplied. This is equivalent to the loss of coolant accident. That report also made reference to experiments which predicted that the natural circulation burn out heat flux in the ORR would be approximately 125,000 BTU/ft.^{2}/hr. The shutdown heat flux of the FNR is estimated to be below 25,000 BTU/ft.^{2}/hr. and will decrease rapidly with time.

From the above information, it is reasonable to conclude that the loss of flow accident should not be expected to produce reactor fuel temperature transients which could result in detectable core coolant boiling, much less damage to reactor core components.

15.2.5 Reactor Pool Drainage

This accident involves a major leak of water out of the pool to other areas of the Reactor Building.

In the event of such major leakage, the water would ultimately be collected in the Reactor Building basement from which there are no gravity drains. Water in the basement is pumped by the sump pumps to the building retention tanks. The retention tanks might overflow into the tank storage area, but also there are no drains to the outside in the tank area. Moreover, a major rapid leakage into the reactor basement area is expected to short out the main process power distribution equipment panel, which would then render the sump pumps inoperative. Therefore, the majority of water from any major pool leakage would be retained within the reactor basement.

The reactor basement area has sufficient free volume to hold all the water from the reactor pool; thus a water level would not build up on the beam hole floor area of the Reactor Building. Therefore, the beam hole floor corridor doors and the main building equipment door do not have to be capable of resisting the damaging forces of a head of water against them.

It is not credible that the reactor pool walls would be a major source of reactor pool leakage. While experience with the reactor pool walls has shown some minor leaks which are evidenced by slow rates of seepage through the walls, this leakage has never caused a serious problem of pool water make-up. The pool wall thickness in the beam port area is six feet of high-density concrete, reinforced on both faces by No. 7 bars spaced on 12 inch vertical centers, while No. 8 bars on 8 inch horizontal spacing are used on the inner face and No. 7 bars on 12 inch horizontal spacing on the outer face. The remainder of the pool walls are conventional concrete, 42 inches thick, with alternating No. 6 and No. 7 bars at 12 inch centers (both ways). The pool floor has a minimum thickness of 27 inches of normal concrete, reinforced by No. 5 bars located on 12 inch centers (both ways). This mode of construction precludes the possibility of a major pool wall rupture being a source of water leakage.

Failure of the present waterproof cover plate on the face of the reactor pool thermal column would be a source of rapid pool leakage. However, present plans for 10 Mw operation of the FNR include the complete sealing of the thermal column structure by installing reinforcing rods within the column and completely filling the column with concrete. The column is no longer useful as an irradiation facility and the 10 Mw design plans for reactor operation in the beam port area only. Thus the sealing of the thermal column will preclude that unit as being a source of potential leakage.

The equipment necessary to 10 Mw operation whose failure could result in a major loss of pool water includes:

a. primary water piping system
b. delay tanks
c. heat exchanger shells
d. auxiliary piping systems (deionizers, filters, etc.)
e. reactor horizontal beam ports
f. reactor pneumatic tube systems

Figures 81 and 82 are estimates of the time it would take for various types of leaks to drain the reactor pool to the point where the core is no longer covered with water. The figures indicate the location of the pool level alarm and the anti-siphon valve. The leakage times indicated on these figures are based on the assumption that none of the safety equipment designed to mitigate leak rates functions properly.

Figure 82 shows the drainage time to uncover the pool for a main line rupture of the primary process pipe lines in the reactor basement. This estimate is based on a separation of the 14 inch diameter return line to the reactor pool and its displacement by one pipe diameter. At least 90 seconds would be required for the water level to drop to the anti-siphon valve location at which point further leakage should be terminated. The check valve located in the primary water return line at the base of the pool should prevent water from "back-flowing" through the return water header to the leak area. If, however, the anti-siphon valve and/or the check valve in the return water header fail to act, the pool would drain in about 150 seconds. This would provide sufficient time for the operators to put the Emergency Procedure into effect and safely evacuate the Reactor Building. The 150 second delay should provide sufficient shutdown cooling to prevent core melt-down. However, to assure this further, the reactor core spray system will have been actuated either manually in the time available to operations personnel before leaving the building, or automatically.

As can be seen by comparing the data presented in both of these figures (81 and 82), the rupture of the primary water line is the most serious potential accident as regards the speed with which the reactor pool can be drained.

While it is clear that the main line rupture is the most serious potential problem, it would appear that the probability of such an accident would be quite low. This is because the primary water system piping is not operating at high pressure nor is it exposed to serious thermal stresses. Consequently, some attention should be directed to other potential leaks in the reactor system.

As regards equipment located in the reactor basement, it is possible that some equipment breakage or missiles as might be generated by the failure of the motor pump coupling could result in the puncture of one or more of the delay tanks or heat exchanger shells. The time required for a 6 inch diameter hole in one of the delay tanks or the heat exchange shells to drain the reactor pool is shown on Figure 82. In this case, almost 900 seconds are available to put emergency procedures into operation before core is completely uncovered. Approximately 500 seconds are available to operations personnel before the water reaches the anti-siphon valve location which should terminate drainage.

Figure 81 shows the time required to drain the pool if a waterline in one of the auxiliary systems should fail, such as the supply line, to the demineralizers and primary system filters. At this time it is not expected that any of these lines will be larger than three inches in diameter which is the case used for the estimate shown in the figure. Again, almost an hour is required for a line of that size to drain the pool to the location of the anti-siphon valve where pool drainage should be stopped.

When one considers the possibility of beam port failures resulting in pool leakage, it should be pointed out that the anti-siphon valve is not effective against leaks of this sort, since the leakage path going through the beam ports and collimator completely bypasses the anti-siphon valve location. Figure 27, which shows the reactor core support structure, indicates that the core will be enclosed in a welded aluminum tank. In the event of a beam port leak which would bypass the anti-siphon valve, the water level in the pool would reduce down to the level of the beam ports; however, the reactor core tank would keep the reactor core covered with water; the anti-siphon valve would open, allow the water in the stand pipe to the anti-siphon valve to backflow down into the reactor core tank, and keep the core tank full of water. In the event that there was a leak or other type of damage to the core tank, the lower eight inches of the reactor fuel elements would still be submerged. This water would provide a heat sink and a supply of steam by which the fuel elements would receive some degree of coolant. Therefore, beam port leakage would not put the reactor core into an air-cooled condition. It should also be noted that for a beam port to drain the core at the rate shown on Figure 81, it would not only be necessary that the beam port rupture, but that sufficient forces be generated to eject the beam port collimator assembly or the beam port shielding plug, whichever is within the port. A more realistic set of circumstances would be for the end of a beam port to be damaged without ejection of the collimator. For water to be able to leak past the collimator inside the port, the water seal on the collimator assembly must also fail. Thus, failure of two separate water seals is required to achieve leakage from the pool through the beam ports. The case of a single collimator leaking (the largest unit presently in use in the FNR) is shown in Figure 81. Figure 82, on the other hand, shows the effect on pool level of all collimators in the FNR leaking. This result assumes that the collimators presently in six of the eight reactor beam ports will still be in use at 10 Mw, and that the other two beam ports will be fitted with 2-1/2 inch diameter units.

The following steps will be taken in order to mitigate the consequences of a pool drainage accident:

1. The Core Spray System will be located over the reactor core at the pool edge and will be initiated either manually or automatically in order to direct approximately 200 gpm of city water down onto the reactor core. The manual control valve will be located near the reactor control room and will contain a quick-acting valve so that opening can readily be accomplished in the time that operations personnel have available to them for evacuation of the facility. In the event that the operators on duty fail to operate this valve, automatic operation will take place when the pool level drops to 4 feet below its normal operating level.

2. The action of the anti-siphon valve will terminate basement system leaks at a position 8 feet above the reactor core.

3. The reactor will be fitted with a pool level alarm system which will:

   a. Indicate when the level has dropped 12 inches below normal operating point.

   b. Automatically initiate the Core Spray System when the level has dropped 4 feet below the normal operating point.

4. A radiation detector mounted on the pool bridge will:

   a. Alarm when the local radiation level reads 100 mr/hr.

   b. Initiate the building evacuation procedure when the radiation level reaches 100 mr/hr.

   c. Automatically initiate the Core Spray System if the radiation level reaches 10,000 mr/hr.

These precautionary measures, while they will not prevent a reactor pool leak, should be sufficient to provide ample time to operations personnel, facility visitors, and experimenters to evacuate the facility before receiving any serious radiation overexposure. They should also be capable of assuring that a reactor core meltdown will not take place.

Reference 31 summarizes calculations that were performed for the FNR at a power level of 2 Mw which treated the problem of radiation levels which would exist in the Reactor Building after exposure. It is reasonable to generally extrapolate these results in the 10 Mw case. After exposure of the core occurs, the radiation level in the area of the walkway around the pool floor region will reach a maximum of not more than 800 R/hr. While this dose rate is high, it must be remembered that evacuation of the facility will already have taken place (should be completed within 30 seconds of the alarm). During the evacuation, the maximum dose rate around the pool level walkway should not be in excess of 25 R/hr. Thus the 800 R/hr. will be generated after personnel have evacuated the facility. Emergency rescue operations to recover anyone injured in the facility on the third floor should be possible. Rescue operations on the first and second floors will be much simpler since the dose rates in those areas will be considerably lower.

At this time it is not believed credible that a fission product release would accompany an accident of the type being described here. However, should such a fission release be possible because of a failure of the Core Spray System to operate or a failure of the anti-siphon valve, it is believed that the over-all fission product release will not be more serious than the Class II accident.

15.2.6 Radioactivity Releases and High Radiation Levels

There are a number of accidents which could result in the release of radioactivity into the Reactor Building. These are conditions different than those already discussed for fission product activity releases from the reactor core. Some of these types of accidents are enumerated below:

1. The removal by an operator of an irradiated sample from the reactor core:

   This type of accident could result in radiation levels in the area of the sample of as much as several thousand R/hr. It should be prevented by the presence of the reactor bridge radiation monitor. This monitor will alarm when the general radiation level in the area of the reactor pool exceeds a predetermined value. This monitor will be in addition to that previously described which actuates the building emergency ventilation sequence.

2. The failure of an irradiation capsule:

   It is probable that a sample capsule failure might occur resulting in the release of the contents of that capsule into the reactor pool. Depending upon the constituents of the sample itself, the material may evolve from the pool into the Reactor Building itself. Suitable standards for irradiation of samples at the 10 Mw power level must be established. These standards must include the consideration of steps to be taken to reduce the probability of such capsule failures. It is recognized that the present standards used at the 2 Mw level, while providing much worthwhile experience in the evaluation of different types of samples, must be modified to take into account the thermal, hydraulic, and radiation damage conditions that will exist at the higher power level.

3. Fuel element leakage:

   Experience with MTR fuel elements has shown that the basic element design is quite reliable as regards the intended service conditions; however, it is not uncommon for an element to exhibit a release of portions of its contained activity. Leaks have occurred from a combination of manufacturing flaws, physical damage during handling operations, fuel element corrosion, and other processes which cause violation of the integrity of the fuel element cladding. The difficulties with these "leaking elements" fall well within the bounds of the Class I accident which has been described previously. No special precautions will be required over and above the continued application of the manufacturing specifications for the fuel elements and those operational procedures which govern the handling of fuel both in and out of the reactor pool.

4. Fuel element suspended on crane:

Since the building overhead crane is used in the transfer of irradiated fuel elements between the reactor grid plate (crane not used on actual insertion and removal from core) and the pool storage racks located throughout the pool area, it is possible that a fault in the building crane "raise" control circuit could cause the uncontrolled withdrawal of a fuel element suspended from the fuel handling tool. The fuel handling tool is short enough that if the crane reaches its upper limit a fuel element suspended from the tool would be above the pool surface. In this event the radiation levels in the area of the pool floor could reach hundreds of R/hr. The crane lift rate at high speed is 4 inches/second. The fuel when brought to the area of the storage racks for element placement has at least eight feet of water for shielding. If the crane were to malfunction and start uncontrolled withdrawal at that time, it would take 24 seconds before the fuel element broke the water surface. The disconnect switch for the reactor crane is located in the southeast corner of the pool floor area and the crane controls during fuel element operations are always manipulated from the east side of the reactor pool. Therefore the crane operator is never more than 30 feet from the disconnect switch. This should provide ample time for the operator to reach the disconnect switch and disconnect all power to the building crane. Should this fail, he still has time to evacuate the facility along with other personnel before receiving any serious overexposure to radiation.

5. Inadvertent fuel criticality:

The possibility of an inadvertent fuel criticality seems remote since all fuel must, by procedure, be stored either on the grid plate, the pool storage racks, or the fuel vault racks. The controls exercised in assuring safety of fuel loading on the grid plates have already been discussed. The design of the storage racks in both the reactor pool and the storage vault have been verified as regards their safety from a criticality standpoint. For an inadvertent fuel criticality to occur there would have to be an overt attempt at sabotage or suicide.

15.2.7 Facility Fires

The possibility of laboratory equipment or other limited area fires should be considered so that adequate fire protection procedures and equipment will be available. The following steps have been taken to provide adequate fire protection for the reactor facility.

1. Each floor area of the reactor facility is equipped with a hose cabinet containing a two-inch diameter fire hose fed from the fire water mains, and a CO_2 type extinguisher for electrical fires.

2. Each experimental area is provided with CO_2 extinguishers.

3. The staff of the Ann Arbor Fire Department Station located closest to the reactor facility regularly tours the reactor facility. Operations personnel point out potential fire and high radiation areas, and review those emergency procedures which must be followed in entering a facility of this type. These procedures are updated at least annually. The Fire Department is equipped with radiation monitors and self-contained breathing apparatus.

4. Periodic tours by the University Fire Marshall, Reactor Advisory Committee, Radiological Safety Officer, fire insurance inspectors and liability insurance representatives for the purpose of reviewing housekeeping and general safety attitudes of the facility management and the adequacy of watchmen, fire protection equipment and overall fire procedures.

These steps should be adequate to assure protection of the facility against the hazards of fires.

15.2.8 Facility Damage By External Causes

Consideration must be given to the possibility that external conditions could result in damaging the reactor facility to the point that operation of the facility would be ill advised. Circumstances which fall into this type of category would include tornadoes, flood, aircraft collision with building, and sabotage or vandalism.

As regards the possibility of flood, while the reactor facility is not on the highest ground in the area, it is located high enough above the Huron River so that should the river reach flood stage, the facility would not be in danger. Some difficulties have been observed in the past with minor seepage of water through foundation seams and around imbedded piping during periods of extremely heavy rainfall. This leakage has never posed any problem other than one of housekeeping.

On the subject of overt acts of sabotage, vandalism or malicious mischief, the security precautions taken around the reactor facility will be regularly reviewed with both the Reactor Advisory Committee and the University Security Officer, in order to provide a level of plant security appropriate to any potential threat to that security. When considering the issue of student unrest and disorders, it should be noted that with the reactor located on the North Campus of the University, it is separated from the University's main campus by a distance of approximately 1-1/2 miles. Since the North Campus area is used primarily for engineering research activities and student housing units, the amount of student disorder which has been centered on the North Campus area has been minimal. While student disruptions of North Campus activities are not inconceivable, the distance from the main campus area provides the operational staff with time to take further security measures which might be indicated.

Three possible sources of damage to the Reactor Building have not been treated in this Report. These include:

1. Tornado or high wind damage to the reactor facility.
2. Resistance of the facility to aircraft accidents.
3. Resistance of the facility to truck or other mobile land equipment collisions with building walls.

It is not possible to make definitive statements of the resistance of the facility to the above conditions. It will be necessary to re-perform the structural analysis of the existing reactor building in the light of criterion considered pertinent for reactor facilities. This has not been done in the preparation of this Report and must receive priority in any further evaluations of high power operation which may follow this Report.

15.3 Design Basis Accident (DBA)

15.3.1 General

The Design Basis Accident (DBA) is an arbitrary combination of various aspects of credible accidents such as were described above. This is done to create a postulated accidental situation of more serious consequence and with a complexity of circumstances beyond anything that could be considered credible for the facility. Such an accident will be postulated in the following paragraphs and its consequences analyzed in Section 15.4 as regards the safety of the general public around the reactor facility.

15.3.2 Accident Description

It is postulated that the initiating event for the DBA will be a large positive reactivity input to the reactor core. With this large reactivity input, the reactor will undergo a rapid power transient and, simultaneously, the radiation level above the reactor pool surface will rapidly (10 to 100 milliseconds) rise to a value sufficient to cause initiation of the Emergency Ventilation System sequence. Assurance that the Emergency Ventilation System will function properly under the DBA conditions will be a major design criterion for that system. Thus, for the sake of this discussion, it is assumed that this Emergency Ventilation System will be capable of functioning during the DBA.

Following the initial radiation burst from the reactor core power transient, the steam bubble formed in the reactor core is postulated to remain in the core region long enough to allow fuel temperatures to reach sufficiently high values such that cladding failure begins and some fission products are released out of the core region.

Subsequently, the steam bubble will move out of the core region and rise to the pool surface, releasing fission products and steam into the reactor building atmosphere. The sudden return of water into the reactor core region, with the expulsion of the steam bubble, results in metal-water reaction between the pool water and the partially molten cladding yielding sufficient energy to cause:

1. Further failure of fuel cladding.
2. Expulsion of additional fission products to the reactor water system.

3. Failure of the core container structure so that a water-filled core container is no longer assured.

It is further postulated that there will be a simultaneous failure of some portion of the reactor primary water system; for example, a failure equivalent to separation of the main water line for the reactor water system. While the proper functioning of the return water check valve and the anti-siphon valve, located in the reactor pool, should mitigate the consequences of such a piping failure (which is unlikely in the first place), it is further postulated that the check valve and the anti-siphon valve, just mentioned, simultaneously malfunction and do not prohibit the drainage of the reactor pool. It is further postulated that the shutoff valves, which would serve as manual overrides to the double failure of the protective equipment just mentioned, are no longer accessible either due to the incapacity of operations personnel, or that the water level in the basement is rising so quickly that operations personnel do not attempt to affect manual isolation of the leaking water system.

It is then presumed that drainage of the reactor pool will go unchecked and that the reactor core will be fully exposed in somewhere between two and five minutes, as shown on Figure 82 for a major pipeline rupture.

It has been stated previously that when the water level reaches some low point in the reactor pool system, the emergency core cooling sprays would have been turned on to prevent further melting of the pool and to slow down the water loss rate from the reactor pool system. For the DBA, it is further postulated that the core sprays fail to function, either because there is a failure of the water supply system, the emergency power system providing power for the electric controls for the core sprays, or that the actuating valves themselves do not function properly.

Under the conditions just outlined, the reactor core, in addition to the damage experienced during the power transient, is now uncovered in a period of minutes after the initiation of the transient. It is presumed that fission product heating of the core under this rapid water loss will be sufficient to provide the means for a further metal-water reaction with any water in the vicinity in the core and a further fission product release both from the additional metal-water reaction as well as the local overheating of the cladding and fuel elements structure caused by the lack of cooling water from either the reactor pool, the core container, or the core spray system.

There will not be a circumstance involving blow-down forces acting on the reactor system. This is because the primary water system is neither high temperature nor high pressure, therefore the basic water system itself does not have any contained energy within it which must be dealt with. All energy during the accident must come from the reactor core and the fission product heating of that core.

15.3.3 Net Effects of Accident

Section 15.2.2.1 of this Report summarized the overall energy release experience accumulated during the Spert testing program and mentioned that in the destructive experiment of the Spert IV D core, the metal-water reaction for the aluminum fraction of the core provided only 12% of the total energy released during the excursion. Reference 20, in discussing transient test experience, indicated that the TREAT studies showed little metal-water reaction occurring in aluminum systems until the energy input from the excursion was greater than 630 BTU/lb. (equivalent to 2/3 Mw-second/lb.). The FNR operating core mass at 10 Mw, including aluminum components, will be approximately 200 pounds. Therefore, a transient of approximately 130 Mw seconds is needed to yield a metal-water reaction for which there will be a significant energy contribution to the total energy of the excursion. It will therefore be assumed that the reactor transient will be one which has an energy yield of 130 Mw-seconds.

On the basis of this information, it is assumed that approximately 5% of the reactor core will react with the pool water in a metal-water reaction which will release an additional 80 Mw-seconds of energy. It is to be assumed that all of this energy will be released simultaneously with the energy coming from the nuclear excursion. This is conservative since we have already presumed that some amount of the metal-water reaction may very well take place after the core has been exposed by virtue of the pool drainage.

The energy required to heat the reactor core structure up to temperatures near the melting point of the aluminum cladding so that the metal-water reaction can take place will be approximately 50 Mw-seconds. Thus the building pressure increase will be caused by the rapid formation of steam from a net energy input to the water system of 160 Mw-seconds. Figure 80, which discussed the building pressure response from nuclear power excursions, was plotted as a function of the energy release to the time of the peak power of the excursion. Approximately two-thirds of the total energy release occurs in the period from the initiation of the excursion to the time of peak power.

Therefore, the net energy input to time of peak power available to cause steam formation is approximately 107 Mw-seconds. This would cause a building over-pressure of 0.38 psi. Obviously, there are uncertainties in predicting the energy release associated with the DBA, but the conservative approach taken above, and the fact that the design pressure for the building is 0.5 psi, should provide sufficient leeway to cover those uncertainties.

With the postulated metal-water reaction, there will be an evolution of free hydrogen from the pool as the aluminum and water react to form aluminum oxide. This evolution of free hydrogen into the Reactor Building will not pose a flammability or detonation hazard. Approximately 200 cubic feet of free hydrogen gas will be formed in the 5% metal-water reaction which was postulated. The reactor bay area has a free volume of approximately 64,000 cubic feet; the hydrogen percentage in this region will be less than 0.3%, which is well below the flammability limit for a hydrogen-air mixture and even further below the detonation limit.

However, one might postulate that the hydrogen evolved during the metal-water reaction might detonate while rising up to the open bay area before it can be diluted by the building air. If this were to occur, the additional energy input from the hydrogen detonation would be 20 Mw-seconds. This is small compared to the energy input already postulated, and, were it to occur, should not markedly change the conclusions which will be drawn regarding the consequences of this accident.

The net affect of the accident as regards the release of fission products from the reactor core is as follows:

1. 100% of fission product noble gases released.

2. 50% of fission product iodine released.

3. 1% of all other radioactivity released.

In the analysis of the radiological consequences of the DBA, it will be assumed that 50% of the elemental iodine released during the course of the accident will either be adsorbed on the building walls or in the reactor pool water. The effects of formation of methyl iodide as a consequence of the release of elemental iodine into the Reactor Building will also be discussed in that section of this Report.

15.4 Radiological Consequences of Accidents (All Types)

15.4.1 Core Conditions Prior to Accident

The following general comment applies to the condition of the reactor core prior to each postulated accident whose radiological consequences will be summarized:

1. Reactor power level - 10 Mw.

2. Reactor core burnup level - 20% average for the core.

3. Reactor core mass - 5 kg. of U^{235}.

4. Dilution factor for facility stacks - 100.

5. Time delay between initiation of incident and release of radioactive material - 1 second (sufficient time to achieve containment closure).

6. Operational cycle - continuous operation sufficient to generate average burnup of 20%.

This last core condition, i.e., the continuous operation of the reactor for the entire operating period prior to the accident, simplifies the calculation of the magnitude of the fission product inventory for each of the radioisotopes considered in the calculations summarized in Appendices A through D. While this approach of considering continuous operation is conservative, it is not overly so. Figure 83 compares the amount of radioactivity of fission products of various half-lives which would be present at the end of the operating portion of a reactor operating cycle compared to the amount of activity which would be present if the same total period of operation was accumulated on a continuous operating basis. As can be seen from the figure, the use of an operating cycle which involves approximately 100 hr./wk. operation and an operating duration of 50 cycles does not reduce the activity of any fission product by more than 35% compared to the continuous operating cycle. The FNR currently operates on the 28-day cycle (25 days of operation -- 3 days of shutdown) and that operating cycle has an even lesser effect. As was discussed in Section 13 of this Report (Administration), the 28-day cycle with intermediate short-term shutdowns for core re-fueling would probably be continued for the FNR at a power level of 10 Mw. Thus, the effect of cycle operation as contrasted with continuous facility operation, would not markedly affect the radiological consequences of an accident in the FNR.

15.4.2 Facility Conditions Immediately Following Accidents

In calculating the radiological consequences of the various types of accidents which have been postulated in this section, certain basic assumptions were made as regards the condition of critical equipment within the reactor facility. These assumptions are:

1. The reactor facility containment, i.e., dampers and doors, will be sealed and will prevent any release of fission products from an accident through them.

2. The Emergency Ventilation System will not be damaged by any accident and will be capable of functioning properly.

3. The Emergency Power System and the Emergency Air Systems will both be capable of functioning if there is a failure of normal electrical power during the course of the accident.

4. The appropriate emergency procedures will be put into effect by available personnel.

With regard to this last point, it should be pointed out that emergency procedures will be designed to mitigate the consequences of the Design Basis Accident so that doses received by personnel and the general public as a consequence of such an accident will be limited to the values which are reported here. The data to be presented in this section of this Report regarding estimates of radiological doses are based on a minimum effort by operational personnel. The actual emergency procedure should be more effective in reducing the radiological exposure of North Campus occupants.

15.4.3 Review of Emergency Ventilation System Functions

Before summarizing the calculations which were performed to estimate the radiological doses which North Campus personnel might receive as a consequence of the DBA and other classes of accidents which have been described, the function of the Emergency Ventilation System will be summarized.

The Emergency Ventilation System is an air filtration system which goes into operation whenever the emergency sequence is initiated as a result of any of the types of accidents which have been described. The emergency ventilation sequence initiates shutdown of the Reactor Building fans, closure of building dampers, and the opening of the control damper to the Emergency Ventilation System filters. The building exhaust fans in the Phoenix Memorial Laboratory then draw a fraction of the Reactor Building air volume through the Emergency Ventilation System of particulate and iodine removal filters. After filtration, this fraction is then released through the reactor and laboratory facility stacks where it is further diluted by the outside atmosphere.

The particulate filters are high efficiency "absolute" type filters of which a number of manufacturers are available. Because of the potential thermal heating considerations which were discussed in Section 4 (Containment) of this Report, the particulate filters will be constructed of high temperature resistant material. These filters will have a DOP efficiency of better than 99.9%.

The iodine removal filters will probably be organic impregnated activated charcoal with a removal efficiency for elemental iodine of better than 99.9%. Filters of this efficiency are also available from a number of manufacturers and are in routine use at several reactor facilities which use the "controlled-release" type of containment philosophy.

The exhaust fans located in the Phoenix Memorial Laboratory which are used to draw the air through this filter system are powered from both the normal and emergency power bus so that a failure to the facility of the supply electrical power should introduce only a momentary interruption to the operation of the Emergency Ventilation System.

15.4.4 Methods Used to Calculate Doses

As a consequence of the accidents which have been described, there will be a release of various types of fission product activity, dependent upon the severity and the type of accident, into the Reactor Building area.

This presence of radioactivity within the Reactor Building will convert the Reactor Building itself in a large-volume source of radiation.

Filtration of this air by the Emergency Ventilation System will release some fraction of the radioactivity into the exhaust plume from the facility stacks. This plume will serve as a second source of external whole body radiation.

Inhalation by persons outside the reactor facility of portions of the activity contained in the building exhaust plume will result in ingestion of some fraction of the radioactivity released by the Emergency Ventilation System. This ingestion of radioactivity will result in an internal whole-body radiation exposure caused by the general distribution of radioactive material throughout the person's body as well as specific organ doses which are the result of the tendency of certain fission products to concentrate in specific body organs.

Personnel re-entering the Reactor Building to perform clean-up operations face a potential radiation exposure from the Emergency Ventilation System filters since radioactivity removed from the building atmosphere will be deposited upon those filters.

The method by which the external whole-body dose from the building, the external whole-body dose from the exhaust plume, and the ingested doses including the whole-body internal dose were all calculated as summarized in detail in Appendix A to this Report. The majority of the conversion constants to which this appendix refers was taken from Reference 13.

The radiation level from the fission products which are deposted on the Emergency Ventilation System filters was calculated using the technique shown in Appendix C.

The actual computer program which was written to perform these several calculations, including the heat generation rate on the filters of the Emergency Ventilation System, is presented in Appendix D along with a tabulation of the fission products which were included in the radiological dose calculations.

This same computer program also performed the calculation of the dose rate from the emergency ventilation system filters not only as a total dose, but by separating each gamma emitter into one of ten gamma energy group classifications. The dose rate contributions from each of these energy groups was also computed. This additional computation serves as the basis for the shielding requirement specification outlined in Section 4 (Containment) for the Emergency Ventilation System.

15.4.5 Guidelines for Evaluation

In 10CFR100 "Reactor Site Criteria" the AEC sets forth the general criteria which guide its evaluation of the suitability of proposed sites for power and testing reactors. This part of the Federal Regulations makes no mention of the applicability of these criteria to the evaluation of research reactor sites. It can only be assumed that the same general criteria guide Commission evaluations of research reactor sites. Moreover, the Regulations make it clear that unusual geographic or geological factors as well as the proximity of a reactor site to heavily populated areas could markedly affect the manner in which these criteria are applied.

However, since more definitive criteria for evaluating the adequacy of a research reactor site following an accident are lacking, these general criteria are considered useful as an initial basis for safety evaluation.

Those portions of the Federal Regulations which are most pertinent can be summarized as follows:

1. An exclusion area must be provided around the reactor facility in which the licensee has the authority to determine all activities including the removal of personnel and property from that area and within which residence will normally be prohibited. In the event of a "Maximum Credible Accident" an individual located at any point on the boundary of this exclusion area for two hours following the accident should not receive a whole body radiation dose in excess of 25 rem or a thyroid dose in excess of 300 rem from iodine exposure.

2. A low population zone immediately surrounding the exclusion area must exist which contains residents, the total number and density of which are such that there is reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident. An individual located at any point on the outer boundary of this low population zone who is exposed to the radioactive release resulting from the "Maximum Credible Accident" would not receive a whole body radiation dose in excess of 25 rem when exposed for the entire duration of the accident.

Based on the calculated results to be presented in the next section of this Report, the North Campus Area of The University of Michigan satisfies the criterion for both areas (exclusion and low population) and thus the occurrence of the DBA will not adversely affect the activities of anyone not under the direct administrative control of the University.

15.4.6 Results of Calculations

Figures 84 through 89 present the results of the calculations performed to estimate the radiological doses which could result from the types of accidents which have been described.

Figures 84, 85, 86, and 87 show the external whole-body radiation dose which would be received by individuals standing at various distances from the Reactor Building wall for the entire duration of the Class I, II, III, and Design Basis Accidents. As can be seen from the notes on the figure, a difference in removal effectiveness for iodine of a factor of 2 does not affect the results of these calculations. There is also a slight dependency of dose with exhaust rate, since at higher exhaust rates the radioactive material in the building volume will be displaced and deposited on the Emergency Vent System filters more quickly.

At this point it should be noted that the lobby of the Phoenix Memorial Laboratory is located approximately 200 feet from the south wall of the Reactor Building. The lobby at the current time is used as the control center for emergency operations and the point to which all emergency personnel report in the event of a building evacuation. For the FNR at a power level of 10 Mw and any accident of greater severity than the

Class I accident, the emergency plans will have to consider the use of an alternate area to the Phoenix Memorial Laboratory Lobby as a collection point for emergency personnel. This has been discussed in Section 13 (Administration) of this Report. The lobby will still serve as the primary collection point, but if conditions indicate a severity of accident which is resulting in a significant dose rate at the lobby area, the alternate emergency control point will have to be used.

Furthermore, it is the data presented on these four figures which justify that the 10 Mw emergency plans will have to consider evacuation of a number of North Campus buildings. Referring back to Figures 3 and 4, which show the location of buildings on the North Campus area as well as the projected construction for the North Campus area in the vicinity of the Reactor site, it can be seen that all buildings within a 1000 foot radius from the Reactor facility are completely under the control of the University administration. While evacuation of these facilities might be disruptive as regards North Campus normal activities, it would not prove to be as difficult as if there were residences in that area. Evacuation to the 1000 foot radius would result in external whole-body dose exposures from the DBA of less than five rem for persons staying at that location for the entire duration of the accident. Thus, the 1000 foot radius easily satisfied the 10CFR100 criteria.

Figure 88 presents essentially the same information but as a function of distance from the Reactor Building with exhaust rate as a parameter. The information presented in Figure 88 is the worst case since it represents the DBA circumstances for the range of exhaust rates which are the probable bounds within which the Emergency Vent System design exhaust rate will be selected.

Figure 89 presents the dose which people would receive when exposed to the exhaust plume from the reactor facility stacks. This is the external whole-body dose from the fission product carried in the exhaust air plume. The doses contributed by the exhaust air plume both for the "2-hour" case and the "duration of accident" exposure are quite small (few millirem) compared to the exposures from the Reactor Building.

The total radiation level from the Emergency Ventilation System filters upon which the fission product activity will deposit as a function of time after the accident for various emergency exhaust system flow rates is shown on Figures 90 through 92. The Class I accident does not involve the release of iodine or particulate activity, and these filters do not remove gaseous fission product activity. Therefore, there is no figure for the Class I accident. The most severe case is the DBA in which an Emergency Vent System exhaust rate of 300 CFM would produce a radiation level from the filter system of approximately 50,000 R/hr. three hours after the accident occurs. The shielding requirements for the Emergency Ventilation System because of this radiation level has been discussed in Section 4 (Containment) of this Report. For the purposes of the radiological evaluation of the accidents as regards personnel exposures, it is presumed that adequate shielding will be provided for these filter systems. Therefore, an additional dose contribution to personnel exposure will not be made by these units.

Figures 93 through 99 present the internal doses which are to be expected for the various classes of accidents. The exposures to the following organs are summarized:

Figure	93	Whole body (internal dose caused by general distribution in body of ingested activity)
	94	Bone
	95	Kidney
	96	Lung
	97	Liver
	98	Testes
	99	Thyroid

The dose to muscles was also calculated but the calculated result was a muscle dose of less than one millirem for the "duration of accident" exposure following the DBA.

The bone, kidney, liver, testes, and whole body doses are less than 1 rem for the "two-hour" exposure and less than 10 rem for the "duration of accident" exposure to each of the classes of accidents including the Design Basis Accident. The lung dose for the "two-hour" exposure is slightly higher because of the major contribution of the gaseous fission products to this exposure; however, even in this case the lung dose for the "two-hour" exposure is less than 2 rem with the "duration of accident" exposure less than 10 rem.

Figure 99 presents the dose to the thyroid for various exhaust rates and two different removal effectiveness values for the iodine filters. The results shown are based on the assumption that no methyl iodide is formed during the release of iodine from the classes of accidents which have been discussed. Under this assumption, operation of the FNR at 10 Mw on the North Campus area meets the 10CFR100 criterion for thyroid exposure. It should be pointed out that these calculations were done on the basis of a stack dilution factor of 100. Tests which were done at this facility (Reference 5) demonstrated a dilution factor that was low only at a point very close to the Reactor Building (within 50 feet) and under "down-wash" atmospheric conditions. For distances further from the Reactor Building and for other types of atmospheric conditions, the dilution factor was much higher. Thus, while the thyroid dose which has been computed is certainly not negligible, the method of computation is extremely conservative.

Table 15-2 on the following page has been assembled to permit a readier comparison of doses from the different types of accidents than can be seen from the several graphs. The Table is based on the following set of circumstances:

> A person remains at a down-wind point, 400 feet from the Reactor Building, for the duration of the accident. The Emergency Vent System is at 100 cfm with a:

Iodine filter efficiency	99.8%
Particulate filter efficiency	99.9%
Stack dilution factor	100

Based on the data which has been presented in this section of the Report, it is clear that the North Campus area of The University of Michigan does satisfy the 10CFR100 reactor site criteria and does so within a radius of 1000 feet from the Reactor Building.

15.4.7 Effect of Methyl Iodide Formation

References 17 and 18 both suggest that under accident conditions as much as ten percent of the fission iodine released during an accident could be in the form of methyl iodide. This possibility of the formation of methyl iodide introduces a complication into the analysis of the accidents treated thus far because of an apparent discrepancy in the effectiveness of iodine filters to remove methyl iodide from an air stream compared to their ability to remove elemental iodine.

TABLE 15-2

SAMPLE TABULATION OF "DURATION OF ACCIDENT" EXPOSURE

Conditions: Location: Downwind, 400 ft. From Building
EMERGENCY VENT SYSTEM: 100 CFMS DFI = 500

DOSE RECEIVED (REM)		Accidents: Class I	Class II	Class III	DBA
Whole Body	a. From Building	negl	2.0	14.0	22.0
	b. From Plume	0	negl	negl	negl
	c. Inhalation	0	.05	0.4	0.7
		<1	<3	<15	<23
Internal:	Bone	0.0	negl	2.5	3.0
	Kidney	0.0	0.1	0.9	1.2
	Lung	0.0	0.4	4.0	6.0
	Liver	0.0	negl	0.7	1.0
	Testes	0.0	negl	0.12	0.2
	Thyroid	0.0	14.0	60.0	130.0

Doses less than 50 millirem considered negligible for this Table

Reference 17 presents a table of recommended design values for methyl iodide efficiency of impregnated activated carbon adsorbers as a function of the relative humidity in the air passing through the filter units. They recommend an efficiency value of not more than 98% when the relative humidity is less than 85% and the air temperature is 270° F. If the air temperature should be lower or the relative humidity higher, they recommend significant reductions in the design efficiency value used. If their observation of significant reductions in the methyl iodide removal efficiency of activated charcoal filters compared to its effectiveness for elemental iodine removal is current, then the fraction of iodine which would pass through the filter system as methyl iodide could significantly increase the thyroid doses to which persons outside the building are exposed.

In a discussion with Prof. G. H. Whipple of this University, it has been suggested that the methyl iodide should be considered as effective as elemental iodine in causing thyroid radiation exposures. Figure 100 presents a conversion chart by which the reader can convert the elemental iodine thyroid exposure which was reported on Figure 99 to the thyroid dose which would be the result of a combination of:

1. The fraction of iodine converted to methyl iodide.
2. The filter system efficiency for methyl iodide removal.

To demonstrate the contribution of methyl iodide to increased thyroid doses, Figure 101 presents the results of the calculation which assumes that 10% of the iodine released from the core is (1) in the form of methyl iodide, (2) does not experience any "plate-out" in the Reactor Building, and (3) that the filter efficiency from methyl iodide is 90%. Under these conditions it can be seen that the "two-hour" exposure at a value of 300 cfm is approximately 700 rem and that the "duration of accident" exposure under those same conditions would be approximately 2000 rem. While one can continue to argue that the dilution factor which was used in these calculations is overly conservative, the point still remains that conformance to the 10CFR100 criteria is not established.

The point of confusion which must be resolved during the design phase of the Emergency Ventilation System arises from reports issued by Oak Ridge National Laboratory. Reference 33, which reported preliminary work on the in-place testing of the iodine removal filters at the Bulk Shielding Facility noted that additional equipment had to be installed to reduce the relative humidity of the air which would be passing through the filters. This is recognized as being necessary and such a design will be incorporated into the Emergency Ventilation System for the FNR. However, Reference 32, which discussed the initial performance of the iodine filters used in the HFIR Emergency Ventilation System noted a removal efficiency of greater than 99.99% for elemental iodine and greater than 99.9% for methyl iodide. A number of subsequent reports as abstracted in the NSIC information cards published periodically indicates that this removal efficiency for methyl iodide continues to be demonstrated at each testing of the HFIR iodine filters. It should be noted that the HFIR filters are commercial units, and were not specially designed just for that application. If commercially available equipment can, with proper relative humidity control upstream of the system, provide removal efficiencies of 99.9% for methyl iodide on a regular and reliable basis, then the data presented on Figure 99 for the elemental iodine case will still be valid and there is no further question as to the adequacy of the North Campus area to satisfy 10CFR100 criteria. If such is not the case, then it will be necessary to further evaluate the methyl iodide problem and to ascertain answers to the following questions:

1. Will methyl iodide be formed as a consequence of an accident in a research reactor like the FNR?

2. What is a reasonable upper limit to the amount which might be formed?

3. Presuming suitable engineering of equipment to provide relative humidity control to the airstream entering the Emergency Ventilation System filters, what is a reasonable value of filter efficiency for methyl iodide removal for commercially available filters?

Until these questions can be resolved, it will be necessary to rely on the conservatism of the dilution factor used in the stack analysis to support the position that the North Campus area suitably meets the criterion of 10CFR100.

REFERENCES

1. "Site and Building Plans for an Engineering and Technology Center"; Report of the 1966-67 North Campus Planning Committee, The University of Michigan, College of engineering, June 1967.

2. Climatology of The United States No. 20; U. S. Department of Commerce Weather Bureau. Available from State Climatologist, Weather Bureau Office, Lansing, Michigan

3. Michigan Tornado Fact Sheet 1916-1965; U. S. Dept. of Commerce, Environmental Science Services Administration; Weather Bureau - Office of State Climatologist; East Lansing, Michigan.

4. Earthquake History of The United States; "Part I - Stronger Earthquakes of The United States (Exclusive of California and Nevada)", U. S. Coast and Geodetic Survey. 1963.

5. MMPP - 272; "The Correlation of Wind Tunnel and Field Measurements of Gas Diffusion Using Krypton-85 as a Tracer", Doctoral Thesis, James E. Martin, The University of Michigan.

6. Phoenix Memorial Laboratory Memorandum Report No. 7; "Re-evaluation of the FNR Stack Dilution Factor"; R. D. Martin; Michigan Memorial Phoenix Project, April, 1967.

7. Docket No. 50-313, Supplement No. 6 to Application for Licenses, Arkansas Power and Light Company, July 11, 1968.

8. ORNL-NSIC-5, Vol. 1 and 2, U. S. Reactor Containment Technology, A Compilation of Current Practice in Analysis Design, Construction, Test, and Operation, Williams B. Cottrell and A. W. Sabolainen, Editors, Oakridge National Laboratory, August, 1965.

9. Paper No. 3269, American Society of Civil Engineers, "Wind Forces on Structures", Final Report, Task Committee on Wind Forces, Committee on Loads and Stresses, Structural Division

10. Tornadoes of the United States, Snowden E. Flora, University of Oklahoma Press, Copyright 1953, Revised Edition, March 1954.

11. Technical Paper No. 20, "Tornado Occurrences in the United States", U. S. Department of Commerce Weather Bureau, Washington, D.C., Revised 1960.

12. ORNL-2127, Part I, Vol. I, "Uranium 235 Fission Product Production as a Function of Thermal Neutron Flux, Irradiation Time, and Decay Time", J. O. Blomeke and Mary F. Godd, August 1957.

13. IDO 17151, "RSAC-A Radiological Safety Analysis Computer Program", R. L. Coates and N. R. Horton, May 1966.

14. AECU-3066, "Meteorology and Atomic Energy", United States Department of Commerce Weather Bureau, July 1955.

15. Radiation Hygiene Handbook, Hanson Blatz (Ed.), McGraw Hill Book Company, 1959.

16. Meteorology and Atomic Energy 1968, D. H. Slade (Ed.), USAEC, Division of Technical Information, July 1968.

17. ORNL-NSIC-65, "Design, Construction, and Testing of High-Efficiency Air Filtration Systems For Nuclear Application," C. A. Burchsted and A. B. Fuller, January 1970.

18. Nuclear Applications and Technology, Vol. 8, "Pilot Plant Studies of Methyl Iodide Cleanup By Sprays", L. F. Parsly, January 1970.

19. NSIC Special Survey for Power Uprating, Nuclear Safety Information Center.

20. ORNL-NSIC-23, "Potential Metal-Water Reactions in Light-Water-Cooled Power Reactors", Howard A. McLain, Nuclear Safety Information Center, Oak Ridge National Laboratory.

21. "Argon-41 Measurements at a Pool Reactor", J. D. Jones, Phoenix Memorial Laboratory, The University of Michigan; Prepared for presentation at the Health Physics Society Midyear Topical Symposium, Los Angeles, January 24-31, 1969.

22. "Safety Analysis to Predict The Kinetic Behavior of the Proposed FNR Core After a Large Reactivity Input", P. Boyse, NE599 Master's Project, The University of Michigan, Nuclear Engineering Department, June 1969.

23. Vol. 13, Proc. 3rd International Conference on the Peaceful Uses of Atomic Energy, "Reactor Excursion Behavior", W. Nyer, p. 283, 1964.

REFERENCES (cont.)

24. ANS Trans, 10(1), "Loss of Flow Accident Studies on the 10 Mw GTR", C. J. Hale, June 1967.

25. AFIT TR 67-9, "Fuel Plate Temperature Studies At The Air Force Nuclear Engineering Test Facility", F. B. Buoni, July 1967.

26. ORNL-4169 (Vol. II), "The Oak Ridge Research Reactor - Safety Analysis", F. T. Binford, March 1968.

27. DP-1209, "Iodine and Noble Gas Retention Studies", R. Milham and L. Jones, August 1969.

28. ORNL-TM-2875, "ORR Quarterly Report, July, August, and September 1969", W. Tabor and S. Hart III, February 1970.

29. ORNL-NSIC-24, "Emergency Core Cooling Systems for Light-Water-Cooled Power Reactors", C. G. Lawson, October 1968.

30. Private Communication by Letter from D. Bruce of Smith, Hinchman, and Grylls Associates, Inc. to R. D. Martin dated September 30, 1968.

31. "Calculation of Maximum Fuel Cladding Temperatures for 2 Megawatt Operation of the FNR", PML Memo Report No. 1, J. B. Bullock, June 1962.

32. ORNL-TM-1677, "In-Place Iodine Filter Tests at the High Flux Isotope Reactor", J. H. Swanks, December 1966.

33. ORNL-TM-3043, "Bulk Shielding Facility Quarterly Report", W. H. Tabor, S. S. Hart III, T. B. Nixon.

34. Inflatable Seals, B. F. Goodrich Aerospace and Defense Products Design Guide, 500 S. Main Street, Akron, Ohio.

35. AFFDL-TR-65-131, "Pre-Startup Hydraulic Tests at the Air Force Nuclear Engineering Test Facility", F. B. Buoni, L. B. Hary, V. G. Lewis and E. J. Vallish, Air Force Flight Dynamics Laboratory, Wright-Patterson Air Force Base.

36. BNL-7661, Vol. I and Vol. II, "Final Safety Analysis Report on the Brookhaven High Flux Beam Research Reactor", J. M. Hendrie, April 1964.

REFERENCES (cont.)

37. Memorandum Report No. 9, "A Low Cost Shim-Safety Rod for Pool Type Reactors", R. D. Martin, Phoenix Memorial Laboratory, Ann Arbor, Michigan, October 1968.

38. E. R. 7527, "Hydraulic Flow Test of an 18 Plate Dummy Fuel Element", Engineering Report, American Machine and Foundry Co.

39. "Nuclear Constants of Clean Cold and Depleted MTR Fuel Assemblies", C. Papastergtoa, Master's Thesis, The University of Michigan.

40. ORNL-4169 (Vol. 1), "The Oak Ridge Research Reactor - A Functional Description", T. P. Hamrich and J. H. Swanks, September 1968.

41. Final Hazards Summary Report For The UCNC Research Reactor, Union Carbide Nuclear Co., Tuxedo, N.Y., November 1960.

42. INL Research Reactor Start-up Test Program Report, AMF Atomics, May 18, 1959.

43. "The Study of Practical Experimental Procedures Necessary to Measure Emergent Gamma Radiation of the FNR Biological Shield in the Vicinity of D-E Port", D. Jackson, Master's Project, The University of Michigan Nuclear Engineering Department, 1970.