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HAZARDs SUMMARY REPORT
BABCOCK & WILCOX TEST REACTOR
July 1963

Submitted to
THE UNITED STATES ATOMIC ENERGY COMMISSION
By
THE BABCOCK & WILCOX COMPANY
Atomic Energy Division
Lynchburg, Virginia

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The Babcock & Wilcox Company
Atomic Energy Division
Report BAW-208

ABSTRACT

The Babcock & Wilcox Company has constructed and proposes to operate a nuclear reactor, the Babcock & Wilcox Test Reactor. The major topics discussed in this hazards report are as follows:

1. Characteristics of the nuclear facilities site.
2. Design and operation of the reactor.
3. Organization and procedures necessary to operate the reactor safely.
4. Accident analysis of the reactor.

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1. INTRODUCTION

The Babcock & Wilcox Company (B&W) proposes to operate a nuclear reactor, the Babcock & Wilcox Test Reactor (BAWTR), at the Company's Mt. Athos nuclear facilities site near Lynchburg, Virginia. Mt. Athos is also the location of the Nuclear Fuel Laboratory (NFL), the Critical Experiment Laboratory (CEL), and the Nuclear Facilities Plant (NFP). (See Figures 1.1 and 1.2.)

The BAWTR is a 6-MW-t pool reactor having a core configuration that includes a 6-inch-square central flux trap extending through the length of the core. The reactor will be operated as an irradiation facility with test space in both the central flux trap and the beryllium reflector for irradiation of fuel and nonfuel materials.

The reactor core contains 12 ETR-type fuel elements; two elements have been modified to provide safety rods, and another has been modified to accommodate a regulating rod. The core is immersed in a 12-foot-diameter, 30-foot-deep concrete pool filled with light water. The core centerline is approximately 21 feet below the pool level. The reactor pool and an area housing experimental loop equipment are enclosed by a 33-foot-diameter, 65-foot-high steel containment vessel. Slightly less than half the containment vessel is below grade level. At an internal pressure of 5 psig, the containment vessel is capable of limiting leakage to 1% of the contained free volume within a 24-hour period. Reactor control room and support facilities are located in a concrete and steel building adjacent to the containment vessel. An artist's conception of this building, which also houses the NFL, is shown in Figure 1.3. The facility floor plan is shown in Figure 1.4.

Table 1.1 gives the principal characteristics of the reactor.

Table 1.1. Reactor Characteristics

Reactor power, MW-t	6
Maximum loop power, MW-t	0.75
Pressure at top of core, psig	9
Maximum coolant inlet temperature, F	108
Primary coolant flow, gpm	3800
Maximum heat flux, Btu/hr-ft ²	326,000
Burnout heat flux, Btu/hr-ft ²	750,000
Fuel loading, kg U ²³⁵	6.4
Fuel alloy	35 wt % U-Al
Clad material	Aluminum
Reflector material	Beryllium
Coolant	Light water
Number of safety rods	2
Number of regulating rods	1
Number of shim rods	32

The reactor, experimental loop, and associated equipment are designed with significant safety margins between the normal operating conditions and the maximum safe-operating conditions. Detailed operating procedures are used in conjunction with safety interlocks to assure that operation of the equipment will not exceed safe conditions. The reactor is self-limiting owing to its negative power coefficient. Fast-acting safety systems will reduce reactor power rapidly in the event of an approach to unsafe conditions. The reactor loop complex is completely enclosed in a low-leakage steel containment building designed to limit the release of radioactive materials to the uncontrolled environment in case the normal barriers should fail.

The maximum credible accident with a loop inserted is defined as a gross failure of the high-pressure experimental loop, resulting in significant release of volatile fission products to the containment vessel. Gross failure of the reactor core is not considered credible.

It is concluded that the inherent safety aspects of the reactor, the operation by experienced personnel using detailed procedures, and

the provision of automatic safety systems reduce the probability and magnitude of an accident to acceptable levels. These features, together with the containment vessel and the favorable site characteristics, ensure that the facility can be operated in the proposed manner without undue hazard to the public.

Figure 1.1. Aerial View of Mt. Athos Site Showing CEL
(Looking North)

Figure 1.2. Aerial View of Mt. Athos Site Showing CEL and NFP
(Looking Southeast)

Figure 1.3. Artist's Conception of New Facility

NUCLEAR DEVELOPMENT CENTER

SECOND FLOOR

315' - 0"

FIRST FLOOR

Second Floor Details:

- CONTAINMENT
- (HIGH BAY)
- STORAGE BALCONY
- SLAB OVER ISOLATION AREA
- CONTROL ROOM
- MECHANICAL EQUIPMENT ROOM
- OFFICE
- OFFICE
- OFFICE
- OFFICE
- OFFICE
- OFFICE
- OFFICE AREA
- LADIES REST RM
- MEN'S REST RM
- TELEPHONE EQUIP ROOM
- STORAGE AREA
- (HIGH BAY)
- STORAGE BALCONY
- MICROFILM READ RM
- LIBRARY
- CONFERENCE ROOM
- LABORATORY
- OFFICE
- OFFICE
- OFFICE

First Floor Details:

- PRIMARY EQUIPMENT CELL (BELOW GRADE)
- EMERGENCY ACCESS AIRLOCK
- EXPERIMENTAL POOL
- AIR LOCK
- CRANE ROOM & CASK HANDLING AREA
- STORAGE POOL
- TRANSFER CANAL
- LOOP SAMPLING RM
- RADIO CHEMISTRY LAB
- HALL
- ISOLATION AREA
- HOT CELL
- HOT INSTRUMENT REPAIR
- CELL OPERATIONS AREA
- JANITOR
- COUNTING ROOM
- OFFICE
- RECEPTION AREA
- MECHANICAL EQUIPMENT ROOM
- HOT CHANGE RM
- COLD LOCKER RM
- DARK RM
- METALLURGY LABORATORY
- HEALTH PHYSICS
- HEALTH PHYSICS
- COLD LOCKER RM
- NON DESTRUCTIVE TEST LABORATORY
- X-RAY RM
- OXIDE AREA
- ASSEMBLY POOL
- MACHINE SHOP & ASSEMBLY AREA
- AUTOCRAVE PIT
- PICKLING RM
- VAULT
- LUNCH RM
- CHEMISTRY LAB
- STORAGE
- OFFICE
- TRACER LAB
- SPECTROGRAPHIC LABORATORY

FIRST FLOOR

2. SITE

2.1. General Characteristics

The reactor site is in Campbell County, Virginia, about 4.25 miles east of downtown Lynchburg, the area's largest population center. The site is a 536-acre plateau bounded by the James River on three sides and Mt. Athos to the south and southwest. The reactor site is at an elevation of about 570 feet above mean sea level and about 100 feet above normal river level. Mt. Athos rises approximately 300 feet above the site. Figure 2.1 is a topographical map of the area.

The tracks of the James River branch of the Chesapeake and Ohio Railroad follow the river around the site as shown in Figure 2.2. The nearest tracks are about 820 feet west of the reactor; 1.6 miles of track are within a distance of 0.5 miles or less. Traffic on this line averages one freight train per hour; there is no passenger service. The railroad traffic may be rerouted to avoid the site in the event that it be necessary. The emergency procedures for the facilities require that the Chesapeake and Ohio Railroad be notified of hazardous conditions at the site.

The NFP, CEL, and NFL are also located on this site. Figure 2.2 indicates the geographical relationship between these facilities and the BAWTR. The NFP is approximately 1700 feet southeast of the BAWTR; the CEL is about 500 feet south of the reactor; and the NFL occupies the same building as the BAWTR support facilities.

The NFP produces fuel elements and related components. Approximately 425 persons work in the buildings associated with the NFP. The CEL normally employs a staff of about 40 to operate the laboratory where various critical experiments are made. A 1-MW-t pool-type research reactor is located in the CEL. The NFL facilities include a megacurie hot cell, laboratories for examination of radioactive materials, chemistry laboratories, a recycle-fuel pilot plant, a feed materials

preparation building, and supporting machine shops. About 50 persons are engaged in the development of improved nuclear fuels at the NFL. The BAWTR staff numbers about 20. All facilities and persons are under the direct control and supervision of B&W.

The NFP, CEL, NFL, and BAWTR are considered to be functionally independent in the sense that a malfunction at one facility will not cause an accident at another facility. However, evacuation of all facilities may be required if a serious accident occurs.

The nearest distance to the river is about 900 feet west of the reactor. However, since the James is not navigable in this area and is not used for recreational purposes, an additional 400 feet of exclusion distance is provided by the width of the river. Access to the site is via the single road indicated in Figure 2.2. A fence and gateway are provided where the road passes the NFP, and armed guards are on duty at this point. In addition to the natural barriers provided by the river and terrain, each of the separate facilities employs security fencing.

2.2. Population

The largest center of population in the area surrounding the NDC site is Lynchburg, Virginia. To the east (the direction of prevailing winds), the population is predominantly rural. The nearest large concentration of population east of the site is Farmville, Virginia, about 40 miles away. Figure 2.3 and Tables 2.1 and 2.2 illustrate the rural layout of the population east of the site.

An actual population count for the area was made in April 1962 by B&W. It includes all the area from the site out to a distance of 3 miles, the city limits of Lynchburg. This population study was made by individuals who actually counted the houses in each section. The number of homes was then multiplied by the average persons per family (reference 1960 census) to obtain the population density. The results of this survey are shown in Table 2.3.

Table 2.1. Inhabited Centers East of the NDC Site

Community	County	1950 Population	Direction from NDC	Distance from NDC, mi
Amherst	Amherst	1,038	N	12
Appomattox	Appomattox	1,094	ESE	13
Buckingham	Buckingham	264	ENE	29
Farmville	Prince Edward	4,293 (1960)	ESE	33
Lovington	Nelson	400 (estimated)	NNE	25
Pamplin City	Appomattox	370	SE	23
Scottsville	Albermarle	396	NE	41

Table 2.2. Population of Counties Around the NDC Site

County	Area, sq mi	Population in 1960	Average population density, persons/sq mi
Amherst	467	22,953	48
Appomattox	343	9,148	26
Buckingham	576	10,877	19
Campbell	530	32,958	62
Charlotte	478	13,368	28
Nelson	468	12,752	27
Prince Edward	357	14,121	39
City of Lynchburg	23	54,790	2,300

Table 2.3. Population Density

Distance, mi	Quadrant				Total
	NE	NW	SW	SE	
0 to 0.5	0	0	0	0	0
0.5 to 1	7	11	47	7	72
1 to 2	107	164	33	55	359
2 to 3	157	2,300	602	226	3,285
0.5 to 3	271	2,475	682	288	3,716

2.3. Meteorology

Weather in the Lynchburg area is influenced by two major air masses: the polar continental air mass and the gulf air maritime mass. The polar continental air mass, usually cold and dry, originates over the polar ice cap and northern Canada and predominates during the winter months. In the summer months, the usually warm, moist gulf maritime air mass, which originates near the equator, either moves into this area after sweeping up along the Atlantic seacoast or comes across the Appalachian Mountains after sweeping northward up the Mississippi Valley. These air masses explain the predominance of northwesterly and southwesterly winds in the Lynchburg area.

Because of its sheltered location east of the eastern ridge of the Blue Ridge Mountains, Lynchburg has a moderate climate. The average temperature is 38.9 F for the winter months and 75.4 F for the summer months. Precipitation throughout the year reaches a maximum in the summer months as a result of the gulf air mass. Most winter precipitation is rain although there are occasional snows.

Although only a limited amount of meteorological data has been collected at the proposed site, weather stations have been maintained in the Lynchburg area for more than 80 years. Originally, the weather station was located in downtown Lynchburg about 4.5 miles from the proposed reactor site. In November 1936 the present U. S. Weather Bureau station was established at the Preston Glenn Airport, 9.5 miles from the site of the proposed reactor. Instrument elevations at the airport vary from 950 to 1000 feet above mean sea level. Between the airport and the reactor site there is a range of mountains (Candler) with an average elevation of from 1100 to 1200 feet above sea level.

2.3.1. Rainfall

Based on data collected at the U. S. Weather Station in Lynchburg from 1906 to 1961, 41 inches of rain, occurring in 124 days, may be expected in the Lynchburg area each year. Distributed by season, the record in inches is as follows:

Winter	10.0
Spring	10.6
Summer	11.7
Fall	8.8

Monthly averages for the period are given in Table 2.4. During the 55-year period, the maximum precipitation for one month was 14.87 inches, and the maximum for one 24-hour period was 7.59 inches (August 1928).

Table 2.4. Average Monthly Rainfalls

Lynchburg, Virginia 1906-1961	
Month	Rainfall, in.
January	3.37
February	2.99
March	3.59
April	3.14
May	3.45
June	3.93
July	4.08
August	4.18
September	3.35
October	3.02
November	2.54
December	3.14
Total	40.78

2.3.2. Snowfall

Based on a record of 68 years, about 14 inches of snow may be expected annually. Snowfalls exceeding 1 inch may occur on four days of each year. The greatest recorded snowfall occurred in March 1960 when 24.9 inches fell in one month.

2.3.3. Winds

Wind data were taken from the records of the U. S. Weather Bureau at Preston Glenn Airport and from observations made at the CEL and the Mead Corporation Plant in Lynchburg. The Mead Corporation

Plant is located on the James River just east of downtown Lynchburg, between Lynchburg and the NDC site. The observations at the Mead Corporation Plant were taken every 15 minutes for about 1 year. The data indicate the direction from which the wind was blowing. Wind roses constructed from these data are shown in Figures 2.4, 2.5, and 2.6.

The U. S. Weather Bureau data are a composite covering the years from 1914 through 1933 in downtown Lynchburg and from 1936 through 1950 at the Preston Glenn Airport. Because the terrain between these sites is hilly, these data may not apply directly to the proposed test reactor site. However, data taken at the CEL from April 27, 1959 to April 26, 1962 indicate, as do the U. S. Weather Bureau and Mead Corporation observations, that winds in the Lynchburg area are predominantly from the west, the northwest, and the southwest. Since the data taken at the CEL is based on observations made during the daytime hours only, it may not be a true representation of the winds at the site. These observations were also made at variable hours throughout the day and only during the five working days each week. The information is used here only as an indication of general wind direction and velocities.

Table 2.5, which shows the frequency of surface wind directions at the Preston Glenn Airport Weather Station for the period from May 1953 to May 1958, indicates that winds may be expected from the east and southeast 8.85% of the time. Calms occurred 1.4% of the time.

Table 2.6, which shows the frequency of surface wind velocities by seasons for the period from January 1953 through December 1953, indicates that calms or light air prevailed 15.8% of the time, light or gentle breezes 68.9% of the time, and winds of greater velocity 15.3% of the time.

Generally, summer winds are from the south, southwest, and west; winter winds are from the west and northwest. For 37 years, the recorded wind velocity averaged 7.6 mph. The maximum recorded velocity of 62 mph occurred during two successive years, 1916 and 1917. Over long periods, winds of less than 4 mph are expected to occur 10 to 12% of the time.

Table 2.5. Hourly Surface Wind Observations

Preston Glenn Airport May 1953 - May 1958	
Direction	Frequency, %
N	14.45
NE	12.35
E	4.15
SE	4.70
S	18.2
SW	20.65
W	12.3
NW	11.8
Calm	1.4

Table 2.6. Frequency of Hourly Surface Wind Observations in Various Speed Groups

Preston Glenn Airport January 1953 - December 1953					
Velocity, mph	Frequency, %				Annual average
	Winter	Spring	Summer	Fall	
0 to 3	10.0	17.4	19.4	16.4	15.8
4 to 12	67.9	64.4	74.5	68.9	68.9
13 to 24	21.9	17.9	6.0	14.5	15.1
25+	0.2	0.3	0.1	0.2	0.2
Average velocity, mph					
	9.4	8.3	7.1	7.9	8.2

Since wind directions during precipitation have not been recorded, it is assumed here that wind directions under this condition are the same as those recorded. Any prediction concerning washouts of air-borne contaminants must be a general statement based on this assumption.

2.3.4. Fog

Over a period of 43 years heavy fogs (visibility less than 0.5 mile) occurred 21 days per year in the Lynchburg area. In 1953 there were 42 days during which fog occurred. It is estimated that only 2% of the fogs extend beyond mid-morning or noon.

2.3.5. Visibility

Table 2.7 gives the frequency of visibilities observed at the Preston Glenn Airport from January 1953 to January 1954.

Table 2.7. Frequency of Hourly Visibility Observations

Preston Glenn Airport January 1953 - January 1954	
Visibility, mi	Frequency, %
0-0.125	1.06
0.1875-0.375	0.87
0.5-0.75	1.13
1-2.5	2.59
3-6	11.71
7-15	61.93
20-30	10.14
35+	10.57

2.3.6. Ceiling

Table 2.8 gives the frequency of ceiling heights based on hourly observations at the Preston Glenn Airport from January 1953 to January 1954.

Table 2.8. Frequency of Hourly Ceiling Observations

Preston Glenn Airport January 1953 - January 1954	
Ceiling, ft	Frequency, %
0	0.42
100-200	2.48
300-400	1.75
500-900	2.49
1000-1900	2.67
2000-2900	2.69
3000-4900	5.67
5000-9500	8.10
9500+	73.73

To establish the meteorological conditions at the site more exactly than is achieved using the Lynchburg, Preston Glenn Airport, Mead Corporation, and limited CEL data, a meteorological station has been installed at the site. Recording wind speed and direction instrumentation are mounted on booms extending from the stack. Two levels of instrumentation are provided: one at the top of the stack, 150 feet above ground, and one about 50 feet above ground. Recording temperature elements are provided at the stack discharge and near ground level to detect inversion conditions. A recording rain gauge is provided for use in correlating wind direction during precipitation. A smoke generator is also installed to study the behavior of a diffusing plume released from the site.

2.4. Geology

The bedrock formations in the area of the proposed test reactor consist chiefly of a thick mass of phyllite and quartzite. The phyllite, which makes up 95 to 97% of all the exposed rock along the bounds of the Mt. Athos site, is a dense, dark-gray slaty rock having a submetallic sheen and composed almost entirely of clay, fine silt, and finely divided mica flakes.

There is a continual slope from the facility location to the river. This slope, steep for the first few hundred feet near the facilities, becomes more gradual nearer the river banks.

The reference elevation for the contour lines in Figure 2.1 is sea level. The contour lines for Figure 2.7, made from a private survey for B&W, uses as a reference elevation a concrete bench mark about midway between the CEL and the NFP. This point gives an elevation of 180.36 feet, about 567.91 feet above sea level as measured on the U. S. Geodetic Survey (USGS) contour map.

The phyllite bedrock exposures are of significance since they dip at steep angles (inclined to the southeast) and trend almost unvaryingly north 25 to 30 degrees east. Thus, both surface and subsurface drainage away from the plant site will not be westward toward the James River. The steep bluff rising above the railroad on the west side of the bend is bedrock, which rises to the surface along the entire length of the meander neck. This rib of bedrock will deflect percolating influent seepage in an eastward direction.

The following quote is from a report by B. N. Cooper of Virginia Polytechnic Institute who studied the geology of the Mt. Athos site.

"In summary, the writer wishes to reiterate his belief that no observed geological conditions are at all adverse to development of the plant site for the industrial purposes of the Company. Indeed, the site because of the geological conditions is almost ideal for control of effluent water, for building foundations, and in all other essential respects."

2.5. Hydrology

2.5.1. Surface Water

The site of the proposed test reactor lies on the James River between two U. S. Geological Survey stream gaging stations. One station is located at Holcomb Rock, upstream from the site, the other is at Bent Creek, 17.5 miles downstream from the site. The water-stage recorder at Bent Creek is 381.39 feet above sea level, at latitude 37 degrees 32 minutes, longitude 78 degrees 50 minutes. Records are available for the period from March 1925 to September 1954.

Figures 2.7 and 2.8 were prepared from the records covering the period from 1926 to 1953. Figure 2.7 shows the monthly mean discharges in cubic feet per second (cfs) as a function of the frequency of observations less than a certain magnitude. In a normal distribution of sufficiently long record, the plot should be a straight line. The absence of the straight plot in the lower portion of the observation (500 cfs versus 4.4%) indicates a disturbing influence caused by irregular river flow at power plants and other dams above the gaging station. Besides regulating the flow of surface water, existing dams serve as a flood deterrent, although none of the dams were designed for flood control. These dams are tabulated in Table 2.9.

Table 2.9. Principal Dams in the James River Basin Above Lynchburg, Virginia

<u>Location</u>	<u>River</u>	<u>Miles above the Lynchburg dam</u>	<u>Principal use</u>
Reusens	James	3.45	Power
Holcomb Rock	James	11.90	Power
Colemans Falls	James	13.0	--
Big Island	James	18.37	Power
Bedford	James	21.58	Power
Cushaw	James	22.75	Power
Balcony Falls	James	27.60	Power
Lynchburg Reservoir	Pedlar	27.90	Water supply
Buena Vista	Maury	38.67	Power
Lyle	James	76.00	--
Covington	Jackson	106.82	Power

Figure 2.9, showing the recurrence intervals of annual flood peaks and the minimum daily discharges at Bent Creek for the years from 1926 to 1953, indicates that a flood peak of 115,000 cfs may occur 1 day in every 30 years. Such floods would not affect the BAWTR since the reactor location is at an elevation approximately 100 feet above the normal level of the river.

Figure 2.9 also indicates that the expected minimum flow is approximately 220 cfs, as recorded on October 31, 1931. A curve extension of this minimum flow indicates that the bottom had been reached, so it is safe to conclude that the minimum river flow will not be less than 200 cfs at the site. This minimum flow is a factor of 100 greater than the BAWTR requirements. River water is used at the BAWTR only for cooling-tower makeup, process-water makeup, and possibly for potable-water makeup.

The nearest significant downstream dam, Boshers Dam, about 130 river miles below Lynchburg, is used for diversion and power.

There are no major recreational areas downstream of the proposed site. Although individual game fishing is prevalent, commercial fishing is impossible in the James River for a long distance downstream from the site.

2.5.2. Ground Water

Precipitation or a spill of water on the ground either percolates down through the soil, evaporates, or flows along the surface to streams. Water that permeates the soil is intercepted by the roots of vegetation or continues downward to the zone of saturation, the upper surface of which is the water table. When the water reaches the zone of saturation, it creates a high point, or point of recharge, on the water table. Low points on the water table, such as springs, seeps, or wells, are points of discharge. Ground water moves from points of recharge to points of discharge through crevices in the unconsolidated material and through fractures in the consolidated rocks.

Prominent along the western edge of the track immediately east of the wooded section shown in Figure 2.10 is a ridge of phyllite that runs from northwest to southeast and divides the surface water drainage. The consolidated material underlying the ridge generally slopes toward the river in a northeast direction. Ground water probably follows along the top of the consolidated material in the northeasterly direction. However, some of the ground water may follow the surface drainage pattern and flows southeast under the slight cut running almost the length of the site. The consolidated rocks restrict the downward flow of ground water.

Small seepages along the Chesapeake and Ohio Railroad track on the site side of the track indicate the top of the ground water table at these points. The vertical distance from the seepages to the reactor is about 65 to 70 feet. The location of the seepages confirms the general direction of ground water flow.

The potential danger of contaminating surface and ground water with radioactive materials has been considered. The course of travel of undefined accidental spills on the ground have been postulated as part of the analysis of this hazard.

During periods of high precipitation or low temperature (frozen ground), the soil would not absorb all of the spill and some of the contaminant could flow over the surface as runoff. Generally, this runoff would be carried by natural drainage northwest or southwest toward the James River. However, two man-made obstacles prevent the immediate discharge of the runoff to the river: the road bed of the single-line railroad track and the abandoned James River and Kanawaha canal.

If the radioactive contaminant percolates into the soil, the probability of contaminating public or private water supplies would be remote since radioactive elements in the fluid probably would be absorbed in clay particles or removed by an ion-exchange process.

If the radioactive fluid reaches ground water, it probably would move down the water table gradient. As previously indicated, the top of the ground water table includes the estimated water elevations in wells and the seepages along the railroad right-of-way. Contaminant discharge at these seepages would follow the course outlined for surface runoff, but the time delay between the accident and the appearance of contaminated seepage would be considerable. Contaminated ground water not discharged as seepage would continue down the gradient and eventually discharge to the river. The probability of contaminating public or private ground water supplies is remote.

2.5.3. Domestic Water Supplies Downstream

The nearest downstream community taking its domestic water supply from the James River is Scottsville, whose population was 396 in 1950. The water supply intake is approximately 60 miles

downstream of Nine Mile Bridge. The U. S. Geological Survey maintains a gaging station nearby.

The Scottsville water treatment system consists of an intake, a small coagulation basin, a rapid sand filter, and a clearwell. Since the clearwell is small, a second reservoir on a nearby hill is needed.

In the event of a radioactive spill at the site and in the unlikely event that the contaminated water reaches the river, domestic water consumers at Scottsville would be protected by several mechanisms:

1. Concentration of contaminants in aquatic animal and plant life.
2. Concentration of contaminants in sludges and slimes.
3. Dilution of contaminated fluid.
4. Treatment at the Scottsville plant.

The large volume of river water available for dilution makes it unlikely that the use of surface water downstream of the site would be hazardous even in the unlikely event of a ground spill.

2.6. Seismology

Table 2.10 summarizes the earthquake history of the Lynchburg area. The following is a definition of the Rossi-Forel Scale used in Table 2.10.

1. Microseismic shock. Recorded by one or two seismographs of a given model.
2. Extremely feeble shock. Felt by a small number of persons at rest.
3. Very feeble shock. Felt by many persons at rest.
4. Feeble shock. Felt by persons in motion. Disturbance of movable objects, such as doors and windows. May cause cracks in ceiling.
5. Moderate shock. Felt by everyone. Disturbance of furniture, beds, etc. Ringing of some bells.

6. Fairly strong shock. Awakening of those asleep. General ringing of bells. Oscillation of chandeliers. Stopping of clocks.

7. Strong shock. Overthrow of movable objects. Falling of plaster. Ringing of church bells. General panic. No damage to buildings.

8. Very strong shock. Falling of chimneys. Cracks in walls of buildings.

9. Extremely strong shock. Partial or total destruction of some buildings.

10. Shock of extreme intensity. Great disaster. Ruins. Disturbance of strata. Fissures in ground. Rocks fall from mountains.

Although the site is in Earthquake Zone 1, the containment building is designed to withstand the more severe earthquakes of Zone 2. There are no known active earthquake faults nearby. The probability of the BAWTR suffering severe damage from an earthquake is therefore remote.

2.7. Conclusions

It is concluded that the reactor site has suitable geological, hydrological, and seismological characteristics for the construction of the reactor and associated structures. It is further concluded that the remoteness of the site, land use and population density of the surrounding area, and area meteorology are favorable characteristics that minimize the probability that the proposed operations will result in an undue hazard to the general public.

Table 2. 10. Earthquakes in Virginia From 1833 to 1947

Date	Locality	Intensity, Rossi-Forel Scale	Area affected, sq mi
Summer, 1833	Central Virginia	Severe	--
December 22, 1875	Arvonnia	7	50,000
January 2, 1885	Maryland and Virginia	4-5	3,500
October 9, 1885	Variety Mills, Nelson County	6	20,000
May 3, 1897	Pulaski	7	150,000
May 31, 1897	Giles County	8-9	280,000
October 21, 1897	Wytheville	5-6	20,000
December 18, 1897	Ashland	5-6	7,500
February 13, 1899	Lynchburg	5	30,000
February 11, 1907	Arvonnia	5-6	2,000
August 23, 1908	Powhatan	5	450
April 2, 1909	Virginia and West Virginia	5-6	2,500
	Maryland and Pennsylvania		
May 8, 1910	Arvonnia	5	350
April 9, 1918	Luray	8	100,000
September 15, 1919	Front Royal	6	--
July 15, 1921	Mevdota	5	--
June 10, 1927	New Canton and Arvonnia	7	2,500
December 26, 1929	Charlottesville	5-6	--

Figure 2.1. Topographic Map of Lynchburg Area

Figure 2.2. Layout of the Mt. Athos Site

Figure 2.3. Map of Area Surrounding BAWTR Site

Figure 2.4. Wind Rose for the City of Lynchburg

NOTE: Length of Bar Represents Probability of Wind Blowing from that Direction. Numbers in Circles Represent Average Wind Velocity in mph.

Figure 2.5. Wind Rose for the CEL

NOTE: Length of Bar Represents Probability of Wind Blowing from that Direction. Numbers in Circles Represent Average Wind Velocity in mph.

Figure 2.6. Wind Rose for the Mead Corporation
(James River Basin, Lynchburg, Virginia)

Figure 2.7. Topographic Map of B&W Area

Figure 2. 8. Runoff as Mean Monthly Discharge
(James River at Bent Creek, Virginia,
1926-1953)

Figure 2.9. Annual Floods and Minimum Flow on the James River at Bent Creek, Virginia

Figure 2.10. Surface Drainage Pattern of BAWTR Site

3. SYSTEMS

3.1. Containment System

The reactor pool and outpile loop space are completely enclosed in a low-leakage steel containment vessel approximately 33 feet in diameter and 65 feet high. The vessel is a cylindrical section with an elliptical head and a flat base. The height above grade, including the dome, is approximately 34 feet. The contained free volume is about 34,000 cubic feet. Figure 3.1 shows a sectional elevation of the arrangement of the BAWTR.

The concrete-walled loop equipment cell contains the high-pressure loop outpile equipment. The cell, vented directly to the above-grade portion of the containment, forms part of the containment system. All of the high-pressure, high-temperature systems inside the containment system, except the underwater inpile section of the loop, are contained in this cell. The 2-foot-thick concrete walls and the 3-foot-thick roof of the cell provide missile shielding for the steel vessel.

Access to the containment is through an 8-foot-diameter personnel air lock, a 5-foot-diameter emergency air lock, a 10-foot-diameter equipment door bolted to the containment vessel, and an underwater gate between the reactor pool and the transfer canal. Personnel access to the containment vessel is permitted during reactor operation, and provisions are made to assure that the containment integrity will not be violated during such access. The personnel air lock is provided with electro-mechanical interlocks that prevent simultaneous opening of both doors. The defeat of this interlock through a key switch automatically scrams the reactor. The emergency air lock is also provided with an interlock to prevent opening of both doors. On this air lock, opening of either door scrams the reactor. The gate between the pool and transfer canal is equipped with a switch that actuates an alarm in the control room if the gate is not properly seated. Administrative control

is used to assure that this gate and the bolted equipment door are properly secured during reactor operation.

An internal system is provided for temperature and humidity control of the containment atmosphere. A recirculating cleanup system, which includes a pre-filter, an absolute filter, and a charcoal filter, is located inside the containment for use if the containment atmosphere becomes contaminated. A purge system brings in fresh outside air and exhausts containment air. Operation of this system is intermittent. The inlet and outlet lines of the purge system are equipped with double automatic valves that close on either a high-radiation or high-pressure level inside the containment.

An intercom system is provided to permit communication between the control room, the containment vessel, and the air locks. This system is supplied by emergency power. Standard telephones are also found inside the containment. Three 12-inch-diameter viewports are installed in the containment vessel to permit viewing of the interior from the control room, the cask handling area, and the loop control room.

3.1.1. Design Bases and Characteristics

The containment vessel is designed to withstand an internal pressure of 5 psig. This is the maximum pressure that would occur following gross failure of a large pressurized water loop. The design leak rate at 5 psig is 1% per day of the contained free volume.

In addition to the normal loads, such as those due to temperature gradients, structural sag, and ground water, which are taken into account in the design of the containment vessel, the structure is designed to withstand loads due to wind, snow, insulation, and earthquake as listed below.

- a) Internal pressure, 5 psig.
- b) Seismic load, Zone 2.
- c) Wind load, 80 mph wind.
- d) Insulation, 5 psf.
- e) Snow load, 30 psf.
- f) Temperature, 150 F maximum.

External pressure is limited to 0.4 psig by two counter-weight vacuum breakers.

The containment walls are steel plate having a nominal thickness of 0.25 inch. The vessel conforms to the following specifications where applicable:

- a) Plate: ASME SA 300 Class 1.
- b) Forgings, pipes, and tubes: ASME SA 350, Grade LF 1 or 2.
- c) Impact strength: ASME Code, Section VIII, Paragraph UG84. Temperature -35 F.
- d) ASA document, "Proposed Standard for Design, Fabrication, and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors", N-6.

A strength test at a pneumatic pressure of 6.25 psig, and an integrated leak test at 5 psig will be conducted before the reactor is operated. At least one additional integrated leak test will be made during the first 3 years of operation. Methods for local testing of penetrations are provided, and the bolted equipment door is provided with a double gasket having a monitored space between gaskets.

3.1.2. Penetrations

The containment vessel penetrations are as follows:

A. Access and Observation

- 1 Equipment access door, 10-foot diameter.
- 1 Double door personnel air lock, 8-foot diameter by 8-foot length.
- 1 Double door emergency air lock, 5-foot diameter by 6-foot length.
- 1 Pool gate, 5 feet 8 inches by 11 feet 8 inches.
- 3 Viewports, 12-inch diameter.

B. Penetrations Below Grade

- 1 1-inch surge tank vent line, Sch 40 carbon steel.
- 1 1-inch service water pipe, Sch 40 carbon steel.
- 1 2-inch demineralized water, Sch 40 stainless steel.
- 2 2-inch emergency spray lines, Sch 40 carbon steel.
- 2 3-inch hot drain lines, Sch 40 carbon steel pipe.
- 2 3-inch cold drain lines, Sch 40 carbon steel pipe.
- 6 4-inch cooling water lines, Sch 40 carbon steel pipe.

- 2 4-inch rabbit tubes, Sch 40 pipes between reactor pool and storage pool.
- 2 6-inch penetrations for passage of smaller high-pressure stainless-steel lines.
- 1 12-inch spare, Sch 40 carbon steel pipe, flanged on both ends.
- 2 20-inch primary piping, one flanged joint in bottom of pool. One penetration in containment shell for passage of 16 inch aluminum pipe.
- 1 24-inch penetration for future pneumatic tubes, Sch 40 carbon steel pipe, flanged on both ends.
- 1 5-inch diameter fuel transfer tube.

C. Penetrations Above Grade

- 1 1-inch compressed air, Sch 80 carbon steel.
- 1 1-inch gas supply, Sch 40 carbon steel.
- 2 3-inch heating lines, Sch 40 carbon steel.
- 2 3-inch cooling lines, Sch 40 carbon steel.
- 4 3-inch instrument air, Sch 40 carbon steel.
- 1 6-inch penetration, carbon steel for sample tubes.
- 2 6-inch purge lines, Sch 40 carbon steel penetrations flanged on both ends.
- 12 12-inch electrical, Sch 40 carbon steel pipe sleeves flanged on both ends.
- 1 12-inch spare, Sch 40 carbon steel pipe flanged on both ends.
- 2 4-inch vacuum-relief tubes.

All penetrations are made through shop installed and inspected sleeves. Expansion joints are supplied where required. All penetrations are x-rayed, magnafluxed, or dye penetrant tested. The containment vessel was designed, fabricated, and erected by the Chicago Bridge and Iron Company.

3.2. Reactor Design and Operation

3.2.1. General Characteristics

The BAWTR is a pool reactor, moderated and cooled by light water. The core consists of a square arrangement (four on each side) of ETR-type fuel elements having the four central elements replaced by an experimental hole. Of the 12 remaining core positions, 9 are filled with 19-plate elements, one with a 10-plate element allowing space for the regulating rod, and 2 with 16-plate elements used as fuel-follower safety rods. As the safety rods are withdrawn from the core, fuel is inserted. The core will operate at a power level of 6 MW-t and contains about 6.4 kg of U^{235} fuel. The core life is 210 days at a 6-MW-t operation.

The core is reflected on four sides by a beryllium reflector. Eight shim rods are located in the reflector pieces on each side of the core (32 total). There are also experimental holes in the beryllium reflector pieces to be used for capsule and sample irradiations. Figure 3.2 shows the reactor arrangements; Figure 3.3 shows a cross section of the core and reflector.

The fuel and reflector elements are surrounded by an aluminum shroud, supported at the bottom by a 7-inch-thick aluminum grid plate, and held in place during operation by the downward flow of the coolant water. The primary inlet plenum provides lateral support for the shim rod guides and also directs the flow downward through the core. Located below the core is a lower shroud and outlet header assembly that directs the flow of coolant water through the core to the circulating pump. The outlet header opens by force of gravity to permit a natural convection of the cooling water through the core in the event of loss of flow or reactor shutdown.

The design of the reactor is based on the use of mechanical equipment and design features similar to those used in existing reactors over a period of several years. Table 3.1 lists the important operating parameters of the BAWTR.

Table 3.1. Operating Parameters

Reactor power, MW-t	6.0
Maximum power of experimental loop, MW-t	0.75
Pressure at top of core, psig	9.0
Coolant temperature, F	
Inlet	100
Outlet	112.5
Average coolant flow, fps	13
Coolant flow, gpm	3,800
Through fuel elements, gpm	3,040
Through reflector elements, gpm	760
Surface heat flux, Btu/hr-ft ²	
Average	96,200
Maximum (hot channel)	326,000
Burnout	750,000
Maximum fuel plate surface temperature, F	248
Fuel loading, kg of U ²³⁵	6.4
Metal-to-water ratio	0.65
Number of fuel elements	
Regular 19-plate	9
Control 16-plate	2
Control 10-plate	1
Reflector pieces, beryllium	20
Control rods	
Number of shim rods	32
Number of safety rods	2
Number of regulating rods	1
Regulating rod	
Stroke length, in.	32
Withdrawal speed, ipm	22.5
Speed of insertion, ipm	22.5
Absorber section	Nickel
Length of poison section, in.	36.25
Safety rods	
Stroke length, in.	33
Withdrawal speed, ipm	4.5
Insertion time, sec	Full flow, 0.48
	No flow, 0.61
Absorber section	Cadmium-Stainless Steel
Length of poison section, in.	31.25

Table 3.1. (Cont'd)

Shim rod	
Stroke length, in.	34.25
Withdrawal speed, ipm	18
Insertion time, min	1.9
Absorber section	Boron-Stainless Steel
Length of absorber section, in.	33
Active core height, in.	30

3.2.2. Core Structure and Support

The reactor and the associated parts of the reactor system are supported from the bottom of the reactor pool by three aluminum braces fastened to the pool floor. (See Figure 3.2.) The three braces extend from the pool floor and are bolted to the sides of the lower shroud. The support braces are calculated to have a critical buckling stress of approximately 25,500 psi. The combined loadings of dead weight, hydrostatic loading, and dynamic loading due to an experiment rupture result in a compressive load of approximately 4,250 psi—well under the allowable stress for the material and 1/6 of the calculated buckling stress.

The lower shroud is a cylindrical aluminum tank including a transition section at the bottom that fits the outlet header (Figure 3.4). The cylindrical grid plate that supports the core and reflector is fastened to the top of the lower shroud. The grid plate is fabricated of aluminum 7 inches thick. The upper shroud, which surrounds the core and reflector, is fastened to the top of the grid plate. This upper shroud, also made of aluminum, extends to the primary inlet plenum and directs the flow of coolant through the core and reflector. It is flattened on one side to permit an experiment to be placed near the reactor core. The primary inlet plenum surrounding the core directs the flow of coolant to the top of the core. Since mixing of the pool and primary system water is limited by directing the coolant to the top of the core, a high level of activity at the pool surface is prevented.

3.2.3. Fuel Elements

The BAWTR uses an ETR-type fuel element. In the operating core there will be three variations of the element. Each type will use the same flat plates, but plate dimensions and the number of

plates in the types of elements will vary. The elements are described as the standard BAWTR element, regulating rod element, and the safety element. The features of each are given in the following sections. Figure 3.3 shows the core location of each type of fuel element.

3.2.3.1. Standard Fuel Element

The standard BAWTR fuel element (Figure 3.5) is approximately 3 inches square and 40 inches long. The active fuel section is 30 inches long. The element is made up of 19 flat fuel plates pinned into aluminum side plates. The fuel plates are fabricated from a 0.020-inch fuel layer of enriched uranium-aluminum and clad on each side with 0.015-inch-thick aluminum. Each element contains approximately 593 grams of U^{235} ; each core loading requires nine of these elements.

The fuel elements in the core are supported by a fitting attached to the lower end of the fuel element section. This fitting is inserted into the holes in the grid plate to position and support the elements. To facilitate handling, a fitting is attached to the upper part of the element. Both fittings are fabricated from aluminum.

3.2.3.2. Regulating Rod Element

The 3-inch-square regulating rod fuel element (Figure 3.6) is located in one of the corner fuel positions of the core. The nine central fuel plates are removed to provide a slot for the regulating rod. Five fuel plates are arranged on each side of the element and are separated from the regulating rod by two 0.125-inch-thick aluminum plates that create the channel for the regulating rod. The 10 fuel bearing plates contain approximately 311 grams of U^{235} . The fitting at the top of the element is designed to offer vertical support for the regulating rod drive mechanism.

3.2.3.3. Safety Elements

The fuel follower for the safety rod contains 16 fuel plates pinned into aluminum side plates. These elements are similar to a standard fuel element except that they are 2.506 inches square. Two of the safety elements are used in each core assembly. Each element contains about 396 grams of U^{235} . Figure 3.7 shows the construction of the safety rod and the details of the fuel element section.

3.2.4. Reflector

The reflector elements (Figure 3.8) are located around the periphery of the fuel elements. The 20 beryllium elements surrounding the core are approximately 3 inches square and 30 inches long. An aluminum nozzle adapter is pinned to the lower end of each element and an aluminum handling piece is pinned to the top. The nozzle adapter supports the element and directs the coolant flow through the element. The four corner elements each have a 1.812-inch-diameter hole to accommodate capsules. Each of the eight elements adjacent to the corner positions have a 1.312-inch-diameter capsule hole and a 0.563-inch-diameter hole for the shim rod. Each of the remaining eight elements contain three 0.563-inch-diameter holes for shim rods and a 0.250-inch-diameter hole for cooling water flow.

3.2.5. Upper Shroud and Filler Blocks

Figure 3.9 shows a cross-sectional area of the upper shroud. The side of the shroud facing the pool gate has been flattened to provide irradiation space. Large items to be irradiated may be brought into the pool through the gate and pushed out onto the transfer car bridge up to the side of the shroud. Aluminum filler blocks are used in the space between the beryllium reflector and the inside of the shroud wall.

Nuclear calculations indicate that aluminum filler blocks will reduce k_{eff} by approximately 1%, and this in turn, will reduce core life. These calculations also indicate that the reactivity change due to the movement of a neutron absorbing material being irradiated from the outside of the shroud will be less than 0.2% Δk .

3.2.6. Control Rods and Drives

3.2.6.1. Safety Rods and Drive Mechanisms

Figure 3.10 shows the arrangement of the various components associated with a BAWTR safety rod. The maximum over-all cross-sectional dimension of this assembly is 3 inches square. Details of the fuel section, poison section, guide tube, and flow tubes are shown in Figure 3.7.

The guide tube extends from below the grid plate up through the reactor core to a point above the grid plate. At this point, the guide tube is attached to a shell assembly tube.

The safety element fuel section is attached to the poison section by a latch device. The top of the poison section is attached to the flow tube that is attached to a long connecting rod with a magnet seat at its upper end.

The electromagnet used for withdrawing the inner assembly of the fuel, poison, and the flow tube is located in the drive mechanism above the water line. An electric motor raises and lowers the electromagnet by an Acme screw. Details of the drive mechanism are shown in Figure 3.11. Limit switches provide an indication of rod position, magnet engaged, and rod seated conditions. A potentiometer with a remote read-out indicator is provided for drive position indication.

The poison section of the safety rod is made up of cadmium between layers of stainless steel. The cadmium is attached to the inner stainless-steel tube by a metallizing process; i. e., it is sprayed on.

3.2.6.2. Regulating Rod and Drive Mechanism

A nickel sleeve of an oval cross section is used as the regulating rod in the BAWTR. (See Figure 3.3.) Figure 3.12 shows the details of the rod and drive mechanism. The regulating rod drive mechanism design is similar to that of the safety rod drive mechanism. The principal difference is the speed of rod withdrawal and the fact that the regulating rod does not scram and therefore does not have an electromagnet.

The regulating rod drive mechanism receives lateral support from the reactor bridge. It is further supported through a long aluminum guide tube bolted to the top of the special regulating rod fuel element. Using this support, it is impossible for a stuck regulating rod to withdraw a fuel element from the core.

3.2.6.3. Shim Rod and Drive Mechanisms

Thirty-two shim rods are used in the BAWTR core. By being located at the core reflector interface, these rods affect reactivity by parasitic absorption and by changing reflector effectiveness.

The absorbing section is a series of short thick-walled tubes mounted on a central locking device. Some of these tubes are boron-stainless steel, and others are materials that produce useful isotopes when absorbing neutrons. The shim rod arrangement is shown in Figure 3.13.

As shown in Figure 3.13, the shim rods are driven from below the core by flexible cables running within conduits. At the bridge above the pool, an electric drive motor engages the flexible cable through a drive gear. The shim rods do not drop on scram but begin driving in after a scram signal. Shim rod guides are attached to the top of the reflector elements and receive lateral support from the primary inlet plenum.

3.2.7. Shielding

Sufficient biological shielding is provided to prevent personnel from being exposed to dose rates greater than those specified in Title 10 Code of Federal Regulations, (10 CFR, Part 20) for continuously occupied areas. In areas where the dose rates are higher than the specified rates, access time will be limited to assure that the maximum exposures never exceed the limits as given in 10 CFR, Part 20, for integrated doses.

Shielding around the sides of the core is provided by the pool water surrounding the reactor and the concrete walls of the pool. Most of the area surrounding the reactor is below ground level and is not accessible to personnel. Areas below ground level that can be entered will be closed and entry will not be permitted during reactor operation.

The principal biological shielding on the BAWTR consists of the pool water directly over the core. (See Figure 3.2.) A water depth of 21 feet over the core keeps the direct radiation well below tolerance levels. The N^{16} level is expected to be quite low since little mixing occurs between the primary coolant flow and the pool water proper. The major source of radioactivity at the surface of the pool arises from Na^{24} activity. At a power level of 12 MW, the dose rate at the pool surface is calculated to be less than 16 mr/hr.

Continuous recirculation of the pool water through a mixed-bed-ion-exchange column removes much of the Na^{24} activity. This, however, causes a buildup in radioactivity at the demineralizer, and so

concrete shielding is used in the primary equipment cell to permit access to the demineralizer valves.

The level of N^{16} activity in the primary coolant loop requires sufficient biological shielding to be used in the primary equipment cell to permit access to certain areas. Normal concrete is used for shielding here.

The area within the containment over the experimental loop area is shielded from the core by the concrete floor and the concrete pool wall. Normal concrete is used in both places and the thickness is sized to reduce the dose rate above the floor to less than 7.5 mr/hr. The pool wall thickness is sufficient to permit access to the experimental loop area within a short period after reactor shutdown.

3.2.8. Transfer Canal and Storage Pool

The transfer canal connects the reactor pool with the hot cell. This canal is connected to the reactor pool through a removable gate (Figure 3.1) and to the hot cell through a 2- by 4-foot hatch in the cell floor. Thus, the canal water provides the shielding required during the transfer of samples to and from the test reactor. Also, the canal provides the shielding required when removing test elements from shipping coffins that may arrive from other sites. The gate permits the independent draining of the reactor pool or storage pool for maintenance.

The canal also contains storage racks for spent reactor elements after they are used in the reactor. These racks support the elements so they are cooled by natural circulation. From the storage racks, the spent elements are transferred into a shipping coffin for shipment to a reprocessing plant. Other items requiring shielded storage will be held in the storage pool until disposal can be arranged.

3.2.9. Reactor Pool

The reactor is located in a pool of demineralized water about 12 feet in diameter and 30 feet deep. The pool, constructed of ordinary concrete, is lined with a liquid tile to waterproof the pool and prevent contact between water and concrete walls. The top of the pool, at approximately ground level, is open to permit access to the reactor and experimental space. The primary coolant outlet penetrates the pool floor in the

center of the pool directly beneath the core. Gutters are provided at the top of the pool for overflow. This water is pumped directly into the regeneration waste storage basins at the pits.

3.2.10. Reactor Physics

3.2.10.1. General Characteristics

The nuclear characteristics of the BAWTR were determined for the core arrangement shown in Figure 3.3 and presented in Table 3.2. Most of the calculations were made by the PDQ code and four-group diffusion theory. A mesh layout was chosen to describe as discrete regions most of the major core internals. The fuel plates and water gaps between the fuel plates were homogenized. However, the aluminum side plates of the fuel elements were described separately. All other water channels were shown separately, as were the capsule holes and shim rods in the beryllium reflector. The typical experiment in the central thimble was homogenized, but the concentric structure around the experiment was shown in considerable detail. In each case, the operating condition for the core is the clean BAWTR operating at 6 MW and 120 F average temperature.

The reference loop experiment, as described in Section 5, uses CETR-type fuel pins, 4%-enriched with U^{235} , as the experimental sample. Comparisons are made with SSCR-type fuel pins that are 10%-enriched in U^{235} . Table 3.2 gives the reactor parameters used for the physics calculations.

3.2.10.2. BAWTR Power Distribution Factors

Calculations show that the maximum radial power factor occurs when the regulating rod and 16 shim rods are inserted in the core. This maximum radial peak-to-average power factor is 2.05. The maximum axial peak-to-average power factor is 1.46.

To facilitate thermal analysis, a plot of power-to-average BAWTR fuel power along the hottest homogenized fuel plate in the BAWTR is given in Figure 3.14. This occurs when 16 shim rods and the regulator are inserted. The hot spot along the plate coincides with the hot spot of the core.

Table 3.2. Core Parameters for Physics Calculations

<u>ETR-type fuel elements</u>	
Full 19-plate	9
Safety rod 16-plate	2
Regulating rod 10-plate	1
Size, in.	3 x 3
Length of fuel, in.	30
Array (central four elements removed)	4 x 4
Reflector, 3- x 3-inch beryllium pieces	20
Central opening in core, in.	6.06
Re-entrant test loop OD, in.	5.75
Number of shim rods (0.5-in. OD)	32
Regulating rod, 7/8- x 2-1/4-in.	1
Number of safety rods (2 to 2.5 in. ²)	2
Number of capsule holes in reflector (1.812 in. diam)	4
Number of capsule holes in reflector (1.312 in. diam)	8
Moderator and coolant	H ₂ O
Reactor power level, MW-t	6
Average operating temperature, F	120
Operating temperature of typical experiment, F	600
<u>CETR-type fuel composition</u>	
U ²³⁵ enrichment at. %	4
Th/U ²³⁵ ratio	24.5
Cladding	304 Stainless Steel
Loading, gm U ²³⁵ /pin	6.3
<u>SSCR-type UO₂ fuel composition</u>	
U ²³⁵ enrichment, wt %	10
U ²³⁸ /U ²³⁵ ratio	8
Cladding	Zircaloy-2
Loading, gm U ²³⁵ /pin	47.2
<u>SSCR-type ThO₂-UO₂ fuel composition</u>	
U ²³⁵ enrichment, at. %	10
Th/U ²³⁵ ratio	9.1
Cladding	Zircaloy-2
Loading, gm U ²³⁵ /pin	43.9

3.2.10.3. Experiment Power Generation Rates

The experiment with CETR-type fuel will generate between 343 and 420 kw of power, depending on the number of shim rods in the core. The BAWTR fuel produces 6 MW of power under all conditions.

The power and the power per pin per foot for the various fuel loadings investigated in the BAWTR experiment are reported in Table 3.3. The maximum power per pin per foot reported is the maximum power in the experiment, which will occur in each case at the outside surface of the outer corner pins. This maximum power includes a 1.46 axial power factor together with the radial power factor of each experiment loading. These values reported in Table 3.3 are for the BAWTR core when no shim rods are inserted; having shim rods inserted, these values will be 15 to 20% higher.

Table 3.3. Experiment Power

Fuel type	Number of pins	Power in experiment, kw	Average power per pin length, kw/pin-ft	Maximum power per pin length, kw/pin-ft
CETR	56	343.2	2.4	4.3
SSCR (10 wt % enriched in U ²³⁵)	16	555.9	13.9	32.2
SSCR (10 wt % enriched in U ²³⁵)	9	370.1	16.4	33.9
SSCR (ThO ₂ -UO ₂ enriched to 10 at. % in U ²³⁵)	16	501.3	12.5	29.1

3.2.10.4. Reactivity Control

A curve of the safety rod worth versus distance of insertion into the core is shown in Figure 3.15. This curve represents the relative effect of one or both of the safety rods.

Calculations indicate that a nickel regulating rod having a wall thickness of about 0.282 inches will provide 0.8% Δk control when fully inserted. The results of reactivity control studies are summarized in Table 3.4.

Table 3.4. Reactivity Control in the BAWTR

Core description	k_{eff}	Δk_{eff} from no rod case
Clean operating, all rods out	1.1329	--
Clean operating, 32 shim rods inserted	0.9072	-0.2257
Clean operating, 16 shim rods inserted	0.9729	-0.1600
Clean operating, 8 shim rods inserted	1.0260	-0.1069
Clean operating, 1 safety rod inserted	1.0586	-0.0743
Clean operating, 2 safety rods inserted	1.0066	-0.1263
Clean operating, regulator inserted	1.1249	-0.0080

3.2.10.5. Reactivity Effects of Experiments

Calculations were made to establish the reactivity effects of various loop experiments in the central irradiation space. The effects are summarized in Table 3.5. The maximum reactivity is expected to occur when the BAWTR fuel is in the specimen section of the test loop. The minimum reactivity expected is when there are thorium pins in the test loop specimen section. Filling the central space with water, aluminum, or a mixture of aluminum and water is expected to give slightly higher reactivity than the test loop with thorium pins.

3.2.10.6. Reactivity Effects of Experimental Accidents

The effect on reactivity of various experimental accidents have been investigated. A loop accident involving fuel specimen meltdown and flooding of the experiment with 120 F water results in a decrease of k_{eff} . Fuel specimen meltdown and loop voiding results in an increase in k_{eff} . These results are summarized in Table 3.6.

Failure of the loop, which results in expulsion of the loop water, gives the reactivity effects shown in Figure 3.16. The maximum addition occurs when the loop is 60%-voided. Complete voiding causes a net reactivity decrease of 0.0163 Δk .

Violent failure of the inpile section of the test loop, which displaced the core and internals in the outward direction, would decrease reactivity as shown in Figure 3.17.

Removal of the aluminum test block or the aluminum basket, described in Section 5, from the central test area decreases reactivity. Insertion of a mixture of aluminum and water into the central test region could add as much as 2% reactivity. The safeguards employed to prevent changes in the central region while the reactor is critical are also described in Section 5.

Table 3.5. Criticality Values

	k_{eff}	Δk_{eff}
Maximum effective multiplication (BAWTR fuel in experiment)	1.1705	+ 0.0376
Typical effective multiplication (CETR-type fuel in experiment)	1.1329	- 0.0497
Minimum effective multiplication (Thorium pins in experiment)	1.0832	

Table 3.6. Experiment Accidents

Core description	k_{eff}	Δk_{eff}
Minimum effective multiplication (Thorium pins in experiment)	1.0832	-0.011
Experiment flooded with water at 120 F (no pins in experiment)	1.0722	+0.006
Experiment voided (no pins in experiment)	1.0892	

3.2.10.7. Temperature and Void Coefficients

The calculated BAWTR core temperature coefficient is $-1.4 \times 10^{-4} \Delta k/F$ at 120 F. Figure 3.18 is a plot of k_{eff} versus core temperature. The core void coefficient has been calculated to be $-0.5 \Delta k/\Delta \rho$ (ρ in gm/cm³) at a density of 0.9880 gm/cm³.

The calculated reactivity coefficients for the core and test loop are given in Table 3.7. The reactivity effect of changing the loop annulus temperature is shown in Figure 3.19.

Table 3.7. Reactivity Coefficients

Source	Value, $\Delta k/^\circ F$
BAWTR fuel-moderator $\times 10^{-4}$	-1.40
BAWTR reflector $\times 10^{-4}$	+0.21
Loop annulus $\times 10^{-4}$	+0.32
Fuel specimen coolant $\times 10^{-6}$	+1.00
Fuel specimen Doppler $\times 10^{-6}$	-0.25

The results of a temperature deficit investigation are summarized in Table 3.8. The k_{eff} increases in bringing the experiment from 68 to 600 F due to the decrease in the poisoning effect of the water annulus being more important than the decrease of k_{∞} in the experiment fuel.

Table 3.8. Temperature Deficit

Core description	k_{eff}	Δk_{eff}
Entire reactor at 68 F	1.1262	-0.0038
Central experiment (CETR-type fuel) at 68 F, remainder of core at operating temperature of 120 F	1.1224	
Entire reactor at operating temperature, experiment at 600 F and remainder of core at 120 F	1.1329	+0.0105

3.2.10.8. Flux Data

Average thermal fluxes from PDQ calculations are given in Table 3.9, and a plot of the thermal flux across the center-line of the experiment when occupied by CETR-type fuel with eight shim rods in the core is shown in Figure 3.20.

The average thermal neutron flux in the 6.06-inch-square center test region completely filled with aluminum is 1.86×10^{13} nv. The fast-to-thermal flux ratio in the center test region when filled with aluminum is 8.3 to 1. The fast-to-thermal flux ratio in the CETR-type fuel in the experiment is 4.8 to 1. The fast flux includes all neutrons above 0.4 ev.

A representative one-dimensional thermal flux profile is shown in Figure 3.21 for the BAWTR when no rods are inserted and CETR-type fuel is used in the experiment.

The average four-group radial fluxes in various BAWTR components are given in Table 3.10 for the BAWTR core operating with no rods inserted and containing thorium-uranium SSCR-type fuel (10 at. %-enriched U²³⁵) in the experiment. The fluxes are average axial values. An axial power factor of 1.46 should be applied to obtain average midplane flux levels in each of the materials.

Table 3.9. Average Thermal Fluxes

Flux location in core	Average thermal fluxes $\times 10^{13}$ nv		
	No rods	8 shim rods	16 shim rods
Experiment with CETR-type fuel	2.85	3.29	3.52
BAWTR fuel	2.08	2.05	2.04
1.312-in. -ID capsule holes (Al-filled)	2.99	2.70	2.73
1.812-in. -ID capsule holes (Al-filled)	3.07	2.90	2.96

Table 3.10. Four-Group Fluxes

Group	Energy range, ev	Flux value, $\text{nv} \times 10^{-13}$				Maximum at outer edge of beryllium reflector
		Experiment	BAWTR fuel	1.312-in. -ID capsules	1.812-in. -ID capsules	
1	0.9119×10^4	9.20	8.41	4.27	2.58	1.08
2	0.9119×10^4 to 0.50×10^2	3.57	2.59	1.73	1.27	0.63
3	0.503×10^2 to 0.4	2.50	2.03	1.29	0.98	0.62
4	<0.4	1.63	2.08	2.96	3.03	4.95

Notes: Experiment is occupied with thorium-uranium SSCR-type fuel (10 at. %-enriched U^{235}).
 Capsules (1.312-in. ID and 1.812-in. ID) are filled with aluminum.
 Core has no rods inserted.

3.2.10.9. Core Life

Lifetime calculations indicate a core life of 210 full power days when CETR-type fuel is used in the experiment. Figure 3.22 shows k_{eff} versus life. A mass balance is given in Table 3.11. It is estimated that the maximum reactivity condition will give a lifetime of 275 days; the minimum reactivity conditions will give a lifetime of 160 days. Reactivity balances for all three cases are given in Table 3.12.

Table 3.11. Mass Balance
(CETR-Type Fuel in Experiment)

Parameters	Initial	Final
U^{235} in fuel, kg	6.442	5.208
Core life at 6 MW		
Days	--	210
MWD	--	1,260

Table 3.12. Reactivity Balances

	Δk_{eff}		
	Maximum	Typical	Minimum
Equilibrium xenon	0.0388	0.0388	0.0388
Equilibrium samarium	.0095	.0095	.0095
Fission products at end of core life	.0251	.0175	.0139
Maximum reactivity available for lifetime and experiment	0.0971	0.0671	0.0210
Total	0.1705	0.1329	0.0832
Control rod worth	0.2257	0.2257	0.2257
Minimum shutdown reactivity (shim rods only)	0.0552	0.0928	0.1425

3.2.10.10. Neutron Kinetics

The calculated neutron kinetics data for the BAWTR is given in Table 3.13. Reactor kinetics studies are described in Section 7.

Table 3.13. Kinetic Data

Neutron lifetime

Beginning of life	52.6×10^{-6} sec
End of life	74.8×10^{-6} sec

Six-group neutron delay fractions with their associated decay constants

Group 1	β_1	λ_{i-1} sec ⁻¹
1	0.000211	0.0124
2	0.001400	0.0305
3	0.001260	0.1114
4	0.002530	0.3013
5	0.000737	1.1360
6	0.000267	3.0130
	0.006405	

Effective total delay fraction

Beginning of life = (1.15) (0.006405) = 0.007366
End of life = (1.05) (0.006405) = 0.006725

3.2.10.11. Summary of BAWTR Nuclear Characteristics

A summary of the nuclear characteristics of the BAWTR core using CETR-type fuel in the experiment is given in Table 3.14.

Table 3.14. Nuclear Characteristics

Maximum radial peak-to-average power	2.05
Maximum axial peak-to-average power	1.46
Reactivity effects (clean core), k_{eff}	
All rods out	1.1329
32 shim rods in	0.9072
16 shim rods in	0.9729
8 shim rods in	1.0260
1 safety rod in	1.0586
2 safety rods in	1.0066
Voided experiment	1.1166
Regulator inserted	1.1249
Average thermal flux (CETR experiment in loop) $\times 10^{-13}$ nv	
Experiment (CETR-type fuel)	3.3
BAWTR fuel	2.0
1.312-in. -diam capsule hole (Al-filled)	2.7
1.812-in. -diam capsule hole (Al-filled)	2.9
Average core life, days	210
U^{235} core loading, kg	6.44
Reactivity balance, Δk_{eff}	
Equilibrium xenon	0.0388
Equilibrium samarium	0.0095
Fission products (end of life)	0.0175
Lifetime and experiment (maximum)	0.0671
Total	0.1329
Control rod worth	0.2257
Minimum shutdown (shims only)	0.0928
Core temperature coefficient, 120 F	$-1.4 \times 10^{-4} \Delta k/F$
Reactor fuel void coefficient, 0.988 gm/cm ³	$-0.5 \Delta k/\Delta \rho$
Maximum positive void effect in CETR experiment (60%-void)	$+0.006 \Delta k_{eff}$
Maximum positive void coefficient of CETR experiment (0%-void)	$+1.5 \times 10^{-4} \Delta k/\% \text{-void}$

3.2.11. Core Heat Transfer Analysis

A summary of the important thermal characteristics of the BAWTR and a comparison of the BAWTR and the Oak Ridge Reactor (ORR) thermal characteristics are given in Table 3.15. The thermal and hydraulic analyses of the reactor core indicate that sufficient heat removal capability is provided to assure that temperatures will be maintained within safe limits during reactor operation.

The operating limitation for the reactor is fuel element failure as a consequence of burnout or breakdown of the heat removal mechanism. In determining this condition, the correlation suggested by Bernath is used. The value predicted by the correlation is reduced to 60% of the calculated value to account for uncertainties and scatter of data.

The equation used is as follows:

$$q''_{Bo} = h_{Bo} (T_w - T_B)_{Bo}$$

where

$$h_{Bo} = \left[10,890 \left(\frac{D_e}{D_e + D_i} \right) + \frac{48V}{D_e^{0.6}} \right]$$

$$T_w = 32 - 97.2 \left(\frac{P}{P + 15} \right) - 0.45 V + 102.6 \ln P$$

D_e = equivalent diameter, ft

D_i = heated perimeter/ π , ft

V = velocity, ft/sec

P = pressure, psia

The value used for design is

$$q''_{Bo}(\text{design}) = 0.6 q''_{Bo} = 750,000.$$

In addition to the nuclear power distribution, the effect of manufacturing tolerances is also accounted for in the design. It is assumed that the allowable deviations have a normal distribution about the nominal value. Engineering Design factors are calculated that apply to the fluid temperature rise ($F_{\Delta T}$), the film temperature difference (F_{θ}), and the heat flux ($F_{q''}$). For $F_{\Delta T}$, consideration is given to variations in flow

area and fuel content, to plenum and channel velocity distributions, and to power and inlet temperature regulation errors. For F_{θ} , consideration is given to variations in channel geometry, local fuel content and heat transfer area; to plenum and channel velocity distributions; and to power regulation errors. For $F_{q''}$, consideration is given to variations in local fuel content, heat transfer area, and power regulation error. The application of these factors gives a greater than 99% confidence level that the heat flux and surface temperature of the hot channel as calculated will not be exceeded.

Figure 3.23 shows the various temperatures of the hot channel. The most probable peak-surface temperature for this channel is 206 F. The maximum surface temperature expected with 105 F inlet fluid temperature is 248 F. The highest curve indicates the surface temperature at which local boiling begins, based on a correlation by Jens and Lottes.

Considering the $F_{q''}$, the maximum heat flux of the hot channel should not exceed 350,000 Btu/hr-ft² during operation. Since the calculated burnout heat flux is 750,000 Btu/hr-ft², the minimum ratio of burnout heat flux to maximum operating heat flux is greater than 2. The transient heat transfer conditions that exist during flow coastdown and flow reversal are described in Section 7.

It is informative to compare the operating characteristics of the BAWTR with the ORR, an operating reactor whose characteristics are close to those of the BAWTR. A comparison of significant parameters is shown in Table 3.15. The thermal design characteristics of the two reactors are seen to be comparable.

Table 3.15. BAWTR and ORR Thermal Comparison

	BAWTR	ORR
Reactor power, MW-t	6	20
Number of fuel elements	12	24 to 28
Average power per element, MW/element	0.5	0.83 to 0.715
Operating pressure at core inlet, psig	9	20
Velocity in fuel element, ft/sec	13	20
Nominal inlet temperature, F	100	120
Temperature rise, F	12	11
Power peaking factor	3.0 (design)	2.45 (measured)
Average heat flux, Btu/hr-ft ²	96,200	163,000@ 24 elements 140,000@ 28 elements
Maximum heat flux, Btu/hr-ft ²	326,000	400,000
Bernath burnout heat flux	750,000	1,160,000
Ratio of burnout flux to maximum flux	2.3	2.9

3.2.12. Stability

Operation of SPERT, BORAX, MTR, ETR, ORR, and many other reactors of this type have demonstrated nuclear stability of reactors of the BAWTR-type up to power densities much greater than planned for the reactor.

The characteristics of the loop-reactor complex were studied using a closed loop analog simulation of the reference test loop and the reactor. A total of 57 sets of calculations were performed. The calculations were divided into six categories as follows:

1. Step changes in loop temperature control valve position.
2. Valve controller failures causing the loop temperature control valve to drive full open or full closed.
3. Step reactivity increases and decreases.
4. Ramp reactivity increases and decreases.
5. Loop temperature control valve oscillations.
6. Reactivity oscillations.

The objectives of the calculations were to determine that the automatic control system was adequate and the reactor-loop combination was inherently stable.

With the valve controller gain set in the correct range (Gain 1 to 2), the reactor power was held to within 5% of 6 MW-t by the regulating rod. The valve was rapidly brought to the correct position, and temperature oscillations were damped out.

Permanent failure of the temperature control valve in the closed position resulted in eventual reactor scram. Failure in the full open position resulted in reactor power reduction to less than 1 MW-t in less than 1 minute as the regulating rod failed to keep up with the reactivity decrease.

Step changes in reactivity up to about ± 10 cents did not result in the reactor power varying more than 5% from the set point.

Ramp reactivity additions up to $\pm 1.5 \times 10^{-4}$ $\delta k/k$ per second were compensated for by the automatic control system, and the reactor power did not drift more than 5% from the set point.

Temperature control valve position oscillations having a peak magnitude of $\pm 100\%$ of the initial position were imposed for a frequency range of about 0.03 to 1.0 cps. No temperature or power resonances were observed.

Reactivity oscillations having a peak magnitude of about ± 7 cents were imposed for a frequency range of 0.02 to 1.5 cps. No instability was observed.

It is therefore concluded that the reactor and reactor-loop combination will be stable.

3.3. Reactor Nuclear Instrumentation and Control

3.3.1. General

The nuclear instrumentation system includes two startup channels, two log N channels, and three linear power channels. The system is an integrated set of instruments provided by the Bailey Meter Company. The exceptions to this composite system are the provision of neutron detectors and of adjustable range picoammeters from other

vendors. These units are fully compatible with the Bailey Meter system. This system functions to perform the following:

1. Monitor the neutron flux level from source level through full power.
2. Provide rate of change (period) information from source level through full power.
3. Provide a signal to initiate regulating rod insertion when selected signals reach their respective set points.
4. Provide a signal to initiate scram when selected signals reach their respective scram set points.
5. Provide a signal for regulating rod control to hold the reactor at a set power level.
6. Provide a convenient means for operation, annunciation, and indication of the reactor condition.

Figure 3.24 shows the approximate range of detection of each channel.

The detectors are contained in aluminum instrumentation thimbles and are rigidly supported and insulated to reduce possible ground currents as well as microphonic and thermal noise in the detector. Major nuclear instrumentation equipment is located in the reactor control room.

3.3.2. Instrumentation Ranges

The instrumentation system is divided into three functional groups: low-level startup instrumentation provides neutron flux and period information from source level to approximately 8×10^{-4} full power; log N channel provides neutron flux and reactor period information, as well as inputs to the safety system, from approximately 10^{-6} full power to full power; and linear power range instruments indicate the neutron level and provide operational information to the automatic control system and reactor safety system from 10^{-5} full power through full power. These three groups operate with overlapping ranges to prevent loss of information as the reactor is brought from startup to full power.

3.3.3. Startup Channel

The startup channels, shown in Figure 3.25, rely on the processing of individual pulses resulting from fissioning of the U^{235} atom coating in the startup fission chambers, and the channels can be arranged to accept pulse input from other types of chambers, such as BF_3 -filled and B^{10} -lined. Each detector pulse is transmitted through the preamplifier, which matches the impedance of the detector to that of the cable and transmits the pulses through the cable to the linear amplifier with maximum efficiency and minimum attenuation. The linear amplifier, composed of a high-gain, low-noise amplifier and discriminator, converts the preamplifier pulses to pulses of given magnitudes and widths. The discriminator limits the linear amplifier's output to pulses from neutrons and eliminates pulses due to noise, gamma rays, and extraneous sources.

The logarithmic count-rate circuit, through a series of integrators, converts the standard pulses from the linear amplifier to a signal proportional to the log of the pulse rate. The period or derivative amplifier differentiates the output of the count-rate circuit and provides a signal proportional to the inverse reactor period.

Available from the count rate and period amplifiers are indicating meters and outputs for a conventional strip-chart recorder. The linear amplifier also has a scaler output for use during critical experiments and reactor startup.

A polonium-beryllium or antimony-beryllium neutron source will be used for startup. After the initial high-power operation, sufficient photoneutrons from the beryllium reflector will serve as the startup source. The startup channel will be interlocked to prevent rod withdrawal unless the count rate exceeds 2 cps.

3.3.4. Log N Channel

The compensated ionization chambers supply a current output proportional to the neutron flux of the reactor. This signal is amplified by a logarithmic amplifier. The log N amplifier utilizes exponential feedback to convert the current signal from the chamber to a voltage proportional to the log of the neutron power. This power signal, indicated on a seven-decade meter, can be recorded on the log N recorder. The log N amplifier output also goes to a period amplifier, which provides period information over the top seven decades of reactor power.

A period trip is set to sound an alarm and initiate safety system action. This trip is designed to guard against rod withdrawal accidents over the source and intermediate ranges. Figure 3.25 shows the arrangement of the log N channels.

3.3.5. Linear Level Channels

Three linear level channels are used to provide power level information to the operator and automatic control system, and to provide input to the scram circuitry. The arrangement of the channels is shown in Figure 3.26. A channel selector switch permits the use of any of the three channels in the automatic control system.

All three channels initiate scram and regulating rod run-down signals if an excessive power level occurs. The scram circuit layout is shown in Figure 3.27. Manual and relay scram remove the line power from the electromagnet power supply. The linear level and period scrams open the line between the magnet power supply and the safety rod magnets. Whenever the current to the safety rod magnets is removed, the safety rods are released and fall into the core. The reactor scram conditions are outlined as follows:

A. Fast Scram (Electronic)

- Short period.
- High linear level.

B. Slow Scram (Relay or Other)

- Building power failure.
- Primary pump failure.
- Loss of recorder power.
- Loss of primary flow.
- Loss of pool level.
- Loss of outlet header.
- Experiment scram signal.
- Containment violation.
- Console key switch.
- Manual push button.

C. Automatic Regulating Rod Rundown

- Short period.
- High linear level.

3.3.6. Reactor Control

The reactor power level is controlled by the 32 shim rods and the regulating rod.

Shim rods are manually withdrawn in preset groups of 4, 2, and 1. The operator does not determine the rod grouping, and only one rod group may be in motion at any one time, except during rundown following a scram. The sequence of withdrawal is specified in operating procedures, and the additional electrical restriction is that the four-rod groups must be withdrawn first. Shim rod positions are individually indicated on the console. Full in and full out limit lights are also provided.

The regulating rod may be positioned manually or by the automatic control system. The automatic control is accomplished by comparing a set neutron level, (a power set point) to an actual neutron power reading from the linear level channel. This is a simple on-off control system in which the regulating rod moves in or out according to power comparator demands. Based on the differences between the power set point and the actual power, the controller utilizes proportional and reset control action to move the regulating rod and maintain the set power level. Regulating rod movement results when the error signal exceeds the controller's dead band. Since this system will drive the rods to reduce this error to zero, the neutron flux level should remain essentially constant. An interlock will trip from automatic to manual control on indication of a scram and whenever the difference between actual and demanded power exceeds a set percentage of the demand.

Figure 3.1. BAWTR Sectional Elevation

SECTIONAL ELEVATION - X-X'

Figure 3.2. BAWTR General Arrangement

[illegible]

Figure 3.6. Regulating Rod Fuel Element

Figure 3.4. BAWTR Layout - Vertical Section

GUIDE TUBE

SECTION-Z-Z

SECTION-Y-Y

Figure 3.7. Safety Element and Guide Tube Details

SECTION-Z-Z

SECTION-Y-Y

Figure 3.9. Upper Shroud and Filler Blocks

Figure 3.10. Safety Element

NOTES: 1. Pressure drop flow tube section
at flow rate of 1.5 gpm.
2. Constant flow 100-500 x 10³

Technical drawing of the generator assembly. The main view is a side elevation showing the internal components: a speed reducer, a self-synchronous generator (SELFSYN GENERATOR #2), and a canyon plug. The overall length is dimensioned as 24 6/16 inches. Three cross-sections are provided: SECTION Z-Z (a cross shape), SECTION Y-Y (a circular view of the generator), and SECTION X-X (a square view of the generator's internal structure).

Figure 3. 12. Regulating Rod and Drive Unit

Figure 3.13. Shim Rod and Drive

Figure 3. 14. P/P_{avg} Along the Hottest Fuel Plate

Figure 3.15. Safety Rod Worth Vs Percentage Insertion

Figure 3.16. k_{eff} Vs Percentage Void in Experiment
(CETR Type Fuel in Experiment)

Figure 3.17. k_{eff} Vs Core Displacement

Figure 3.18. k_{eff} Vs Core Temperature

Figure 3.19. k_{eff} Vs Annulus Temperature

Figure 3.20. Thermal Neutron Flux Vs Distance Across Experiment
(CETR Type Fuel in Experiment and Eight Shim Rods in Core)

Figure 3.21. Radial Thermal Flux Profile

Figure 3.22. k_{eff} Vs Core Lifetime
(CETR Type Fuel in Experiment)

Figure 3.23. Fuel Element Axial Temperature Profile

Figure 3.24. Nuclear Instrumentation — Channel Ranges

Figure 3.25. Nuclear Instrumentation — Log N and Startup Channels

Figure 3.26. Nuclear Instrumentation — Linear Level Channels

Figure 3.27. Nuclear Instrumentation — Scram Circuits

4. AUXILIARY SYSTEMS

4.1. Primary Cooling System

The reactor core, beryllium reflector, core shroud, and water surrounding the core are cooled by circulation of the primary system water through a heat exchanger. The demineralized water is pumped down through the core and reflector, through lower shroud and outlet header, then through the primary pump and shell side of the exchanger back into the pool through a distribution header located above the reactor core. A side stream of 25 gpm is continuously run through the water purification system to maintain cooling water purity. Figure 4.1 is a flow diagram of the reactor cooling systems.

The outlet header, located directly beneath the reactor core, directs flow through the core during the operation of the pumps and permits natural convection cooling when forced cooling is not used. The outlet header, following loss of forced circulation flow, drops down and permits the core to be cooled by natural convection. The header is normally held up by the pressure loss through the core and is interlocked with the primary pump so that the pump cannot be started with the header down. It is also interlocked with the reactor control system so that the reactor cannot be operated above 200 kw with the header in the down position. The header is manually closed before pump startup.

The primary piping is 16-inch OD Sch 40 aluminum. The lines penetrate the containment shell and are sealed and tested to maintain the integrity of the shell. The coolant exit line connects the bottom of the outlet header and the intake side of the primary pump. The return line enters the pool above the reactor core where it is connected to a distribution header. The header and plenum permit an even distribution of water above the reactor core. The valves and flanges used in the primary system are in the 150-pound class. The design pressure for these is about 180 psi at 150 F. The water is circulated through the

primary system at a rate of 3800 gpm by the primary coolant pump. The pump is designed to supply 3925 gpm of coolant water at 45.5 psi and is operated by a 125-hp electric motor, which takes its power from the plant electrical system. There is no auxiliary or emergency power to the pump; on loss of electrical power or coolant flow, the reactor automatically scrams. Following pump coastdown, the reactor is cooled by natural convection circulation of the pool water.

The shell and tube type heat exchanger is fabricated of aluminum. It provides cooling capacity for the reactor and experiments, and is designed to withstand an internal pressure of 85 psig. The flow of coolant through the heat exchanger is measured and indicated on the control room console. Low flow is indicated audibly and visually in the control room. Both the inlet and outlet temperature of the primary water to the heat exchanger are measured and indicated in the control room. Audible and visual alarms alert the operator of a failure of the primary coolant flow or system cooling ability. In case of loss of coolant flow, the reactor is automatically scrammed.

The pool water purification system consists of an activated carbon (charcoal) filter and two mixed-bed demineralizers (Figure 4.2). Purity of the coolant water in the pool is maintained by recirculating 25 gpm continuously through the system and back into the primary coolant. The system is also used for purification of makeup water and fill water for the pool. Normal additive water is obtained from the building demineralized water supply. The charcoal filter removes dissolved gases, colloidal material (algae and rust), and undissolved solids from the pool water. The mixed-bed demineralizers remove the ionic material remaining after a passage through the charcoal filter. There are two mixed-bed demineralizers in the system. Normally, only one of these will be in use, but the other is available as a backup or to provide additional capacity.

The radioactive material in the water will consist largely of A^{41} , N^{16} , and Na^{24} . The Na^{24} will originate from two sources, $Na^{23}(n,\gamma)Na^{24}$ and $Al^{27}(n,\alpha)Na^{24}$. Most of the Na^{24} , which decays with a 15-hour half-life, is removed from the water by the primary system demineralizers. Much of the Na^{24} decays on the bed during the collection period; the remaining Na^{24} is stored in the retention basins for further decay

following the regeneration of the demineralizers. The N^{16} , which has a 7.4-second half-life, is not removed from the pool water. The inlet header is designed to reduce the pool turbulence which, in turn, reduces the concentration of N^{16} near the surface of the pool. The A^{41} is produced by the reaction, $A^{40}(n,\gamma)A^{41}$. It is not removed from the water except when it escapes from the surface of the water.

The water for the pool and primary system is normally taken from the plant water supply obtained from the James River. Table 4.1 is a chemical analysis of the plant water from the river before and after filtration.

Table 4.2 gives the design and operating information for the primary cooling system.

4.2. Secondary Cooling System

The secondary cooling system (Figure 4.1) removes the heat generated in the primary water by the reactor and experiments in the heat exchanger and dissipates it in the cooling tower. Water from the cooling tower basin is pumped through the heat exchanger and returned to the tower. The heat produced from the reactor and experiments is then rejected to the atmosphere as the water passes down through the cooling tower. The water is pumped from the cooling tower base through the tube side of the heat exchanger to the top of the cooling tower. It then flows down the cooling tower into the basin.

The secondary pump has a capacity of 3760 gpm and circulates the secondary water through the system at this rate. It is driven by a 125-hp electric motor, which is supplied electrical power from the plant electrical system. In case of a pump or power failure, the reactor would be shut down and operated only on convection cooling.

Flow of secondary coolant through the heat exchanger is measured and indicated in the control room where there is also an indication of heat exchanger inlet and outlet temperature. Butterfly valves, similar to those of the primary system, are provided in the primary equipment cell to permit isolation of the components of the secondary cooling system for replacement or repair. The cooling tower is of a conventional design. Water is cooled as it moves down through the tower by forced circulation of the air.

Table 4.1. Chemical Analysis of James River Water Before and After Filtration

Date sample collected 8-31-55
Date sample analyzed 9-9-55
Sample taken from Nine-Mile Bridge

Cations	Symbol	ppm as Ca CO ₃	
		Before filtration (a)	After filtration (b)
Calcium	Ca ⁺⁺	54	54
Magnesium	Mg ⁺⁺	20	20
Sodium	Na ⁺⁺	21	38
Hydrogen-acidity	H ⁺	0	0
Total		95	112
Anions			
Bicarbonate	HCO ₃ ⁻	69	66
Carbonate	CO ₃ ⁻	0	0
Hydroxide	OH ⁻	0	0
Phosphate	PO ₄ ⁻	0	0
Chloride	Cl ⁻	7	7
Sulfate	SO ₄ ⁻	19	39
Nitrate	NO ₃ ⁻	0	0
Total		95	112
Total hardness	--	74	74
Alkalinity A (methyl orange)	--	69	66
Alkalinity B (phenolphthalein)	--	0	0
Noncarbonate hardness	--	5	8
Sodium alkalinity	--	0	0
		ppm	ppm
Carbon dioxide	CO ₂	11	22
Silica	SiO ₂	6.8	6.8
Iron	Fe	0.8	0.8
Turbidity	--	35	0.2
Color	--	60	--
pH	--	7.1	6.8

(a) Unpurified James River water.

(b) Coagulated precipitator effluent addition of the following chemicals before filtration:

Alum 0.28 lb/1000 gal H₂O
Soda ash 0.14 lb/1000 gal H₂O
Clay 0.28 lb/1000 gal H₂O

Table 4.2 Primary Cooling System Design and Operating Information

Operating conditions

Reactor power output, MW-t	6
Cooling water flow rate, gpm	3800
Core inlet pressure, psig	9
Core outlet pressure, psig	2.7
Core inlet temperature, F	100
Coolant (core) exit temperature, F	112.5
Flow rate through fuel, ft/sec	13
Maximum fuel plate surface temperature, F	248
Saturation temperature at hot spot, F	230
Saturation temperature at outlet pressure, F	219.8
Heat exchanger pressure drop, psi	8.5
Inlet temperature to heat exchanger, F	112.5
Outlet temperature from heat exchanger, F	100
Pump power, hp	125
Pool water temperature outside core, F	
Normal	100
Maximum	110

Design conditions

Pump design flow rate, gpm	3800
Valve pressure, psig	
1- to 12-in. valve	175
14-in. valve or larger	150
Heat exchanger pressure, shell, psi	85
Heat exchanger pressure, tubes, psi	85
Length of heat exchanger, ft	16.5
Diameter of heat exchanger, ft	4-5/16
Number of tubes	2590
Size of tubes	
OD, in	0.75
BWG	16
Length, in	144
Size of pump motor, hp	125
Material of heat exchanger (shell and tubes)	Aluminum
Pump housing material	Stainless Steel
Piping	Aluminum

Coolant water chemistry

Total solids (maximum), ppm	2.0
Chlorides (maximum), ppm	1.0
Copper and mercury (maximum), ppm	0.01
pH range, pH	4.5-6.5

Initial quantities of water and makeup water are taken directly from the plant water supply. The secondary system water quality is controlled by blowdown and the addition of a mixture of sodium chromate and sodium polyphosphate glass. This material is fed to the system by passing the makeup water over solid balls of the chemicals held in a feeder. Due to the low solubility of the glass, the makeup water dissolves only a few parts per million of the chemical.

The design characteristics and operating information for the secondary cooling system are given in Table 4.3.

Table 4.3. Secondary Cooling System

Heat removal capability, MW	7.8
Cooling water flow, gpm	3760
Pump power, hp	125
Cooling water temperature from tower, F	82
Heat exchanger inlet temperature, F	82
Temperature rise across heat exchanger, F	14.4
Heat exchanger pressure drop, psi	10.0
Water evaporation, gpm	54.0

4.3 Plant Water

The plant water will be broken down into the following components: (1) potable water, (2) process water, and (3) building demineralized water. Figure 4.2 shows the demineralized water purification system.

4.3.1 Potable Water

The water required for personal usage will be taken from wells at various locations on B&W property. This water is not processed or treated in any way. It will supply the fountains, sinks, showers, and laboratories throughout the facility. Figure 2.10 shows the location of the wells.

4.3.2 Process Water

The process water taken from the James River is treated, filtered, and stored in storage tanks. Two 155,000-gallon tanks provide

sufficient storage to assure an adequate supply to the building. The tanks are located south of the NFL. This water will be used for: (1) makeup to the cooling tower, (2) supplying the building demineralized water equipment, (3) supplying the building toilets, and (4) backup emergency cooling.

4.3.3. Demineralized Water

The facility is served by three demineralizer units or packages. Each package will contain at least one mixed-bed demineralizer, one carbon filter, and the necessary equipment to regenerate them. Two of the units are maintained as cold processing water containing very low levels of radioactive material; the third unit is normally used with the primary system. However, if either the storage pool and canal or the assembly canal become contaminated, the primary system demineralizer will be used to clean it up. Figure 4.2 shows the interconnection between the purification equipment used to accomplish the cleanup operation.

4.3.3.1. Primary System Purification

The primary system purification is described under the primary system.

4.3.3.2. Building Demineralized Supply

The demineralizers serving the facility are capable of water purification at the rate of 10 gpm. The system also contains a 350-gallon supply tank for storage purposes. The effluent has a maximum concentration of 2 ppm total dissolved solids. The makeup water for the reactor, test loop, storage pool and canal, assembly canal, and chemistry laboratory is taken from the storage tank. The inlet water to the package comes from the process water system.

When the package is not used to supply building demineralized water, it is used to purify water in the assembly canal. During this type of operation, water is recirculated through the demineralizer and filter at the rate of 10 gpm.

4.3.3.3. Storage Pool and Canal

The water in the storage pool and canal is recirculated through the activated carbon filter and demineralizer at the rate of 10 gpm.

4.4. Radioactive Waste Disposal

The disposal of radioactive waste involves general operating problems of collection, classification, treatment, storage, and disposition. These materials are described in a form of gaseous, liquid, or solid wastes. The handling methods, retention facilities, and disposal methods are discussed for each type.

4.4.1. Gaseous Wastes

Radioactive gaseous wastes will be produced in the primary system, high pressure loops, and other experiments. The design of the radioactive gas disposal system permits control of the gas released from the facility. Control is accomplished by filtration, holdup and decay, and monitoring of any gases discharged from the facility. In no case will the levels of released activity exceed those specified in 10 CFR, Part 20. A diagram of the radioactive gas disposal system is included in the ventilation system diagram, Figure 4.3.

Radioactive gases present in the containment vessel air will be from the pool water disassociation and entrained air. This waste will consist of A^4 , N^6 , H_2 , and O_2 . The escape of these gases from the pool water is greatly reduced by the pool return header, which reduces the turbulence of the water and therefore the rate at which the gases are discharged to the air. The activity of the gas in the containment vessel is monitored and then discharged through a prefilter and an absolute filter to the stack at a controlled rate of release. Personnel will not be permitted to enter the containment if the concentration of gas in the building gives a dose rate higher than those of 10 CFR, Part 20.

All types of activated wastes are expected to be generated in the high-pressure loops. The volume of gases that may be found in the loops is low due to the low inventory. Normally this gas will be monitored, diluted, filtered, and discharged up the stack at or below

permissible concentrations. The stack effluent gases will be monitored by the following systems:

1. A gross beta-gamma radiation detector [Geiger-Muller (G-M) tube].
2. A particulate activities monitoring system.
3. A radio-iodine monitoring system.
4. A gaseous activity monitoring system.

The G-M tube, positioned so that the effluent gas flows past it, is used to detect gross beta-gamma activity. The particulate monitoring system consists of a monitored filter; the radio-iodine monitoring system consists of a monitored charcoal absorption bed; the gaseous activity monitoring system consists of a monitored volume.

A side stream from the stack is first passed through the particulate monitoring system, then through the radio-iodine monitoring system, and finally through the gaseous activity monitoring system. The activity levels detected in each of these systems will be $10^{-6} \mu\text{c}/\text{cm}^3$ — the lowest limit obtainable in the gaseous activity monitoring system. However, in the particulate and radio-iodine systems, it will be possible to lower this limit, if desired, since an integrated sample is being monitored in each of these systems.

4.4.2. Liquid Wastes

Activated liquid wastes will originate in the primary purification system, pool or canal, high-pressure loops, and experiments. The arrangement of the liquid waste disposal system is shown in Figure 4.4, and the fluid flow diagram is given in Figure 4.5. Large quantities of liquid wastes are not expected from the operation of the facility, but provisions are made to permit handling of any amount that could possibly be expected to occur from any source.

The liquid waste system is located underground at the west end of the NFL. It consists of two 3600-gallon regenerant storage basins, two collection tanks of 300 gallons capacity each, and three storage tanks with the capacity of 2000 gallons each. It also provides sufficient pumps, pipe, and sampling arrangements to transfer and evaluate the wastes.

The effluent from the regeneration of the primary purification equipment is routed to storage basins. There are two basins, each capable of handling the fluid resulting from the regeneration of one mixed-bed ion-exchange unit. The regeneration effluent is stored in the basins for the decay of the Na^{24} removed from the primary water. Following the decay period allotted for the decay of Na^{24} , the basin fluid is monitored to determine the disposal method. The waste will be pumped into the James River in quantities and concentrations that do not exceed those permitted by 10 CFR, Part 20.

Before the pool or canal is drained, the water activity level is reduced to a value less than the maximum permissible concentrating as defined in 10 CFR, Part 20, for nonoccupational exposure. The water is drained into the waste system for disposal.

The liquid waste from the high-pressure loops is monitored and transferred to the proper storage tank in the liquid waste system where it is either stored for decay until it may be pumped into the river or is collected for shipment in shielded tank trucks for processing and disposal at an off-site facility.

Blowdown from the cooling tower is piped through the waste disposal line to the river.

4.4.3. Solid Wastes

All solid wastes are collected in shielded containers and shipped to off-site burial grounds for permanent disposal. Rags, wipers, and miscellaneous laboratory equipment will be packed in flexible bags. Contaminated filters, glassware, scrap metal, tools, etc., are collected in metal containers. Spent ion-exchange resins are sluiced into shielded drums. Solidified wastes are collected and prepared for off-site shipment in shielded containers. Fuel element end pieces and experimental equipment are retained in shielded storage until the accumulation is sufficient for economical off-site shipment.

4.5. Sewerage

The facility sanitary sewer will be tied into the existing plant sewerage system servicing the CEL and NFP. The system is large enough to handle all sewage from the site. The waste from the plants is piped to the sewage treatment plant north of the NFP. At this point,

waste is treated and discharged into the James River just above Nine-Mile Bridge. There are no potable water intakes for a distance of 60 miles downstream of the sewer discharge.

4.6. Ventilation System

The office portion of the building, including the control room, will be air conditioned.

The potentially contaminated gases are routed to the stack, where they will be monitored before being exhausted. A blower near the base of the stack will be used to maintain a slight negative pressure throughout the collection system. This fan also maintains a negative pressure to all potentially hot laboratories and hoods of the NFL during those periods of time following power failure or source blower failure. The line diagram, Figure 4.3, shows the exhaust and intakes for the building.

Temperature and humidity in the containment building are controlled by an internal system. The containment atmosphere may be purged using the two 6-inch purge lines and having the exhaust routed to the stack.

4.7. Electrical System

4.7.1. Function

This system supplies and controls electrical power to the reactor during emergency as well as normal operation. The system consists of these power supplies

1. A 440-volt, three-phase normal supply for motors rated from 0.5 to 125 hp.
2. A 115/230-volt, single-phase normal supply for instrumentation, control, lights, and motors rated below 0.5 hp.
3. Uninterruptable supply to maintain power to vital equipment.
4. A 440-volt engine driven generator which provides standby power for essential machinery.

4.7.2. Electrical Distribution System

The present facility is served by 12 kv from an Appalachian Power Company 34.5-kv substation comprised of one 3750-kva, three-phase, 34.5-kv transformer and a related switching structure. A duplicate substation in the same fenced area serves other power customers in the surrounding area.

The 480-volt (nominal) substation is located in the NFL building with the 12kv/480v transformer just outside the building. Dry transformers at convenient locations in the building provide 115/230-v single phase power for lights and small power loads.

4.7.3. Uninterruptable Power Supply

A 10-kw uninterruptable power supply is provided to furnish power for vital instrumentation, control, and alarms. This power supply consists of a 440-volt ac synchronous motor powered by the normal ac source, a 120-volt ac generator that supplies power to the vital loads, and a 115-volt dc motor/generator. Under normal conditions the 115-volt dc machine runs as a generator maintaining the charge on a 60-cell battery. Upon loss of the 440-volt ac source, the dc machine runs as a motor deriving its power from the battery without interruption. The battery is sized to drive the ac generator for a minimum of 30 minutes at a full load of 10 kw with approximately a 33% safety factor.

4.7.4. Standby Power

A 25/30-kw engine-driven generator (25 kw continuous rating - 30 kw standby rating) is provided to furnish power for a hot cell fan, glove box fans, air sample pumps, and other loads requiring standby power. This generator starts automatically on loss of normal power, and loads are transferred to the standby source by a automatic line transfer switch.

4.8. Process Instrumentation

Besides the nuclear instrumentation and other monitoring systems for the facility, process monitoring instrumentation is provided to supply operational information for the various systems—the coolant systems in particular. These monitoring systems are designed to

activate certain alarms and indicators that give an indication of systems not operating within their design limitations. Table 4.4 lists the important monitors for the process systems.

Instrumentation associated with the reactor operation is located in the primary and secondary coolant and the primary purification systems. The primary, secondary, and purification flow rates are measured with differential pressure transmitters. The differential pressure measurements are converted to pneumatic signals and transmitted to the non-nuclear instrumentation process panel. At the panel, the pneumatic signals, except the purification flow indicated on a 4-inch vertical gage, are recorded on 24-hour circular-chart recorders.

The inlet and outlet temperatures are measured on the primary and secondary side of the heat exchanger by resistance elements. The temperature signals are recorded on 24-hour circular-chart recorders. Inline conductivity measurements are made before and after the mixed-bed demineralizers, and the measurements are transmitted to 24-hour circular-chart recorders located on the panel. These measurements are used to determine the necessity for regenerating the mixed-bed demineralizers. Push buttons controlling the demineralizer use are on the console. A differential pressure signal measured across the primary purification system carbon filter is transmitted to the process panel and indicated on a 4-inch vertical gage. An inline pH measurement is made in the primary coolant system and is recorded on a 4-inch strip-chart on the process panel.

The reactor pool level is monitored by float switch devices that annunciate low level on the process panel and scram the reactor. The motor start-stop buttons that control the primary pump motor, the purification pump motor, the secondary pump motor, and cooling tower fan motors are located on the console.

Table 4 4. Monitors for Process Systems

<u>System</u>	<u>Monitor</u>	<u>Location</u>	<u>Indication</u>	<u>Signal</u>
Primary water	Flow and temperature	Primary equipment cell	Yes	Yes
Secondary water	Flow	Primary equipment cell	Yes	Yes
Pool water	Level	Containment	No	Yes
Purification	Conductivity	Primary equipment cell	Yes	Yes
Purification	Filter ΔP	Primary equipment cell	Yes	Yes

4.9. Area Monitoring

A fixed radiation monitoring system, similar to the Tracerlab model RMI-103, is installed within the reactor operations area. The system consists of a minimum of ten monitoring stations with normal operating ranges from 0.05 mr/hr to 10 r/hr gamma. All stations will alarm and read out in the reactor control room. Detector stations outside the containment are equipped with local alarms to warn personnel in the area of unexpected radiation fields. The reactor areas covered by the fixed area monitoring system are as follows:

1. Reactor containment.
2. Storage pool area.
3. Crane bay and adjoining facilities.
4. Primary machinery room.

Besides these active systems, a passive system of radiation monitoring is employed throughout the building. The passive system employs film packets of the same type and nature as used for personnel film badges. These film packets are attached to the building in fixed locations and will be developed and read by an outside laboratory along with the personnel film badges.

4.10. Communications

The building is serviced with a telephone system. In addition, a two-way intercom system is used between the reactor control room and associated areas (containment, loop control room, storage canal, Health Physics Office). Other facilities in the area are connected to the BAWTR by the telephone service. Information concerning accidents or hazardous conditions will be passed to other areas by telephone.

The building is equipped with three alarm systems: a fire alarm, a high-radiation alarm, and a building-evacuation alarm. The high-radiation alarm is used with the area monitoring systems and will furnish a local alarm if radiation exceeds a preset level. The fire alarm and building-evacuation alarm are actuated manually and pull-boxes are located at vital points throughout the plant. Each alarm system is unique to avoid any possible confusion.

4.11. Environmental Monitoring

A program of environmental monitoring has been established. It includes the following:

1. Continuous air sampling in one outside location.
2. Biweekly water sampling of grab samples taken from the James River.
3. Continuous rain water sampling.
4. Grab sampling of river silt, plant life, and fish and animal life every three months.

Preoperational surveys have been conducted to determine the background level of radiation and contamination. These surveys were performed on and off the site particularly in the direction of the prevailing winds. Gross beta activity concentrations and an analysis for a long-lived radioactive isotope (Sr^{90}) were performed. The surveys give a base point for the start of the operational surveys. Iodine monitoring will be included in the operational phase.

Figure 4.1. Coolant and Purification System

Figure 4. 2. Demineralizers Systems

Figure 4.3. Building Ventilation

Figure 4.4. Liquid Waste Disposal System

Figure 4.5. Liquid Waste Disposal System — Flow Diagram

5. EXPERIMENTAL CAPABILITY

5.1. Types of Experiments

Irradiation experiments to be conducted in the BAWTR range from simple capsules to complicated capsules and test loops. The BAWTR design provides for a test loop to be centrally located within the core. Experimental holes are provided at the corners of the core in the reflector pieces for capsule experiments. Also, experiments may be irradiated in the pool outside of and against the reflector pieces.

5.1.1. Capsules

Irradiation capsules vary from simple static environmental types to complicated lead types. Within a single static capsule, if the material to be irradiated is noncorrosive, the capsule may be of an open design where the specimen is directly cooled by the primary coolant water. Most capsules will consist of a cannister containing the specimen. Liquid metal may be used to provide better heat transfer from the specimen to the cannister wall. For lead capsule types, externally supplied heating or cooling along with sensing and control instrumentation may be added to perform experiments at elevated temperatures. Other environmental conditions will also be monitored and controlled.

In a representative capsule experiment, the samples are supported in a sealed steel cannister filled with NaK and instrumented with thermocouples. A small heating element is wrapped around the cannister and the assembly is inserted into a large cannister. A gas-filled annulus may be retained between the two cannister walls for thermal insulation. The leads extending from the outer cannister are sheathed by a flexible tube until they reach the top of the pool. The sheath tube may be wet or dry and slightly pressurized. The leads are then taken to the control and monitoring instruments.

5.1.2. Test Loops

The reactor core is designed around a central test facility approximately 6 inches square and will accommodate an experiment somewhat smaller than 6 inches. As shown in Figure 3.3, the typical specimen occupies about 3-1/2 inches of this space. The remainder is occupied by an insulating gas jacket and annulus, a pressure tube, and flow distribution baffles. The corners are filled with aluminum filler blocks. The principal material of construction for the high-pressure loop is Type 316 stainless steel. Small quantities of Zircaloy are also used at certain points in the inpile assembly.

The experimental loop design conditions are as follows:

Temperature, F	650
Pressure, psig	2500
Pressurizer temperature, F	670
Flow, gpm	250
Heat removal capability, kw	750

The high-pressure loop is located in the containment vessel; the inpile portion is located in the reactor pool itself where pipes penetrate the pool wall below the concrete shield level and lead to the outpile equipment area under the concrete. For ease of handling and resin replacement, the high-pressure loop demineralizer is located above the concrete shielding in a small shielded container.

The central experiment is cooled by circulating water in a closed loop. The coolant flow path, as shown on Figure 5.1, runs from the experiment outlet piping upward through the pool wall to a point where the flow splits into two parallel paths. At this point, approximately 90% of the total flow passes through a diaphragm operated bypass flow control valve, the remaining flow passes through one of two helical tube and shell counter-flow heat exchangers through a flow control valve, and then to a mixing nozzle where the two streams are recombined. The mixing nozzle assures complete thermal mixing of the two flows that are at different temperatures. From the nozzle the flow is directed to the suction of the high-pressure loop canned circulating pumps through the outpile control specimen container or its bypass, and then

through the gas-insulated pool piping downward to the inpile assembly inlet nozzle. From the inlet nozzle, the coolant flows downward through an annular space to the grid plate area where it is turned 180 degrees and directed up through the specimen and back to the outlet nozzle.

Heat removal from components in the high-pressure loop (heat exchangers, pump motors, gas cooler, and purification-sample coolers) is accomplished by circulating pool water through the components; hence, a cooling coil rupture would not permit any direct escape to an external cooling system. Normally, the pressure head for circulating this water is achieved by the main primary pump, but during an emergency, circulation is provided by a small emergency circulating pump having a battery backed power source.

The outpile heat exchangers and piping are arranged physically with respect to the inpile assembly so that natural circulation will occur following a pump failure or loss of power to the high-pressure loop pumps. To achieve greater circulation with the available head under these conditions, a flow short circuit is provided, so the coolant may pass directly from the outlet of the heat exchangers through a check valve and back into the inlet line to the inpile assembly. This arrangement permits heat removal in the heat exchangers and eliminates passing the coolant through the entire outpile piping.

Pressure control is achieved by an electrically heated pressurizer. Two relief valves are connected in parallel downstream of a rupture disc directly on top of the pressurizer. The pressurizer is large with respect to the loop volume and consequently loop temperature changes will produce only small pressure fluctuations. By interconnecting the stainless-steel gaseous waste collection tank with the pressurizer and makeup system, it is possible for the tank to receive all of the high-pressure loop excess water that is generated. The sizing of the pressurizer relief valves was based on an abnormal operation that assumed inadvertent closing of the heat exchanger flow control valve. This assumption results in all specimen heat being absorbed by the water, thus causing an insurge to the pressurizer.

Makeup to the high-pressure loop may come from one of two sources: the water stored in the gaseous waste collection tank

or the primary system demineralizers. Makeup is added by one of two small reciprocating pumps, each capable of approximately 15 gph makeup.

High-pressure loop water chemistry is maintained and checked by a demineralizer, gas addition system, and sampling system. It is also possible to degas the water in the high-pressure loop by spraying it into the pressurizer and then bleeding the steam and gas mixture into the waste collection tank. The steam is condensed from the stream by a small heat exchanger. The gases may then be bled at a controlled rate from the waste collection tank to the stack. Any necessary chemical addition to the loop can be achieved by placing the chemicals in a sampling bomb and forcing them into the loop after the bomb is inserted in the sample stream.

A separate system is also provided to assure that a dry gas is always present in the insulating gas annulus of the inpile assembly.

The instrumentation and controls for the high-pressure loop have been designed with simplicity and safety in mind. The only automatic controls on the high-pressure loop are those of average temperature control of the loop coolant and pressure control on the pressurizer heaters. The remainder of the parameters, such as total loop flow and pressurizer level, are controlled manually. There are three separate scram inputs originating in the high-pressure loop: low-loop flow, low-loop pressure, and high-loop temperature. All of these signals are obtained from two independent instrument chains. Alarms and parameter indications are also provided on the test loop panel to give the operator full cognizance of loop conditions.

The outpile auxiliary equipment is located in the shielded loop equipment pit having controls and instrumentation positioned on the outside of the containment vessel. The concrete shield also serves as a missile barrier for the high-pressure auxiliary equipment. The loop system is designed so that a specimen can be subjected to dynamic environmental conditions differing greatly from those of the BAWTR core. This design allows not only different conditions, but also the flexibility of varying conditions during the course of the irradiation. The experiments may consist of single specimens or simulated fuel element assemblies. These experiments may be instrumented with

thermocouples, flow orifices, sampling lines, and wires for determining the flux profile to provide more exact data than is provided by loop instrumentation.

The inpile portion of the loop will be a top-entering, re-entrant tube (Figure 5.2). The inlet and outlet coolant lines will leave the pool through pool wall penetrations into the loop equipment pit below the pool water surface. The inpile tube will normally remain in place while the irradiated samples are removed through a closure located below the pool water level. The water height above the closure is to provide adequate shielding during irradiated sample removal. The irradiated samples may be transferred to the canal by the underwater pool gate for subsequent insertion into the hot cell complex for examination.

5.1.2.1. Typical Loop Experiment

In a typical high-pressure loop experiment, the sample may be a single pin or as many as 16 pins arranged in a bundle using square or triangular pin pitch. The fuel pins would be positioned in grid assemblies and spacers at a number of points over the 2.5-foot core length. Thermocouples can be inserted in the bundle. The complete bundle assembly would be contained within the flow tube that directs water flow as required by thermal and hydraulic considerations.

All instrumentation leads would penetrate the loop pressure vessel at the loop closure port through a high temperature-pressure fitting (Figure 5.2). The instrument leads are then taken to the monitoring and control instruments.

Instrumentation of the center experiment which relates to reactor safety is listed in Table 5.1. The scram action is initiated through the safety amplifier, which shuts off the magnet holding current.

Table 5.1. Experiment Variable Monitoring Actions

Monitored variable	Action
Loop coolant temperature	Scram on high temperature
Loop pressure	Scram on low pressure
Loop coolant flow	Scram on low flow
Loop coolant pumps	Scram on loss of operating pump
Loop coolant conductivity	Alarm on high conductivity
Pressurizer level	Alarm on high or low level
Pressurizer temperature	Alarm on high temperature
Waste collection tank level	Alarm on high level
Gas supply to insulation annulus	Alarm on low pressure
Sample line temperature	Shutoff sample line and demineralizer on high temperature

5.1.2.2. Operation Without a Loop

For reactor operation without a test loop in the central region, the region is filled with either an aluminum block or the aluminum can arrangement shown in Figure 5.3. Use of the can arrangement will not be permitted above a power level of 200 kw. A sample placed in the can and positioned by spacers contacting the can wall permits flux and reactivity measurements.

5.1.3. Isotope Production

A fraction of the short hollow tubes of boron-stainless steel, which are part of the shim rods may be replaced with tubes of an alloy that will yield quantities of useful radioactive isotopes after exposure in the reactor.

5.2. Reactivity Effects of Experiments

One of the primary safety features of the BAWTR is the limitation on the amount of available excess reactivity that can be inserted into the reactor by any operational error, equipment failure, or experiment failure. This is accomplished by limiting the available excess reactivity in the control rods and the total reactivity in the experiments.

The experiments for the reactor will be designed so that reactivity greater than $0.6\% \Delta k/k$ cannot be added to the reactor as a result of a credible failure or malfunction in the experiment. Prior to operating an experiment, the reactivity effect of the experiment on the reactor will be calculated by conventional computational methods. This value will be checked by measurement.

Calculations made for some types of test loop experiments that could be operated in the central facility of BAWTR indicate that there are areas over which a positive temperature or void reactivity contribution may occur in the experiment. Over the range of temperatures and voids possible in the experiment, the contribution of positive reactivity is less than $0.6\% \Delta k/k$. For all experiments placed in the BAWTR, the magnitude of the core negative power coefficient is greater than the positive power coefficient from all experiments combined.

5.3. Experimental Handling

Experiments initiated by B&W will generally be fabricated and assembled within the NDC complex. Experiments by others will be fabricated and assembled to BAWTR requirements. Prior to the initiation of insertion procedures, all experiments must meet certain acceptance criteria and receive the BAWTR Operations Staff approval. The BAWTR Safeguards Committee will periodically review the design and operation of all experiments and will review all novel experiments prior to installation.

Prior to a scheduled BAWTR shutdown, the experiment is moved into the containment staging area. Monitoring and control instrumentation are set up in the assigned space, and the necessary connecting lines and wires are readied at the assigned location on the edge of the pool.

After shutdown and the removal of terminating experiments, the new experiments are inserted. A loop experiment would be inserted into the inpile thimble through the top closure assembly and positioned by extensions of the flow director and bundle assembly. The positioning extensions would enclose and protect the instrumentation leads extending from the test bundle against the turbulent water flow at the inlet and outlet loop coolant line connections. The instrumentation leads would penetrate the closure through high-temperature, high-pressure fittings.

The leads then extend upward through the pool water to the side of the pool and are connected to the leads to the monitoring and control instruments.

Nonlead capsule-type experiments would be inserted into the irradiation position in support baskets. A capsule located within one of the reflector positions can be no larger than the hole size less spacing for the basket and the required water coolant channel. A lead capsule may be inserted allowing clearance for only the water coolant channel provided additional clearance for the leads is not required. Such an experiment will employ the leads as a handling device.

On the termination of an experiment, loop or capsule, the instrumentation leads, if any, will be severed at the side of the pool and sealed if required. A capsule would then be removed from the irradiation position and passed under water through the open pool and canal gates or through the fuel transfer tube into the storage canal. Discharge of a loop experiment starts with the shutdown of the loop after an adequate cooling period. During and after loop shutdown, the test would be monitored to avoid accidental overheating. After severing and protecting the instrumentation leads the top closure is removed first, followed by the experiment removal. The experiment is then transferred to the canal.

Figure 5.1. Test Loop Flow Diagram

Figure 5.2. Pressurized Water Test Loop

Figure 5.3. BAWTR Control Region Can

6. ORGANIZATION AND PROCEDURES

6.1. Organization

Figure 6.1 shows the B&W organization with particular reference to the BAWTR. The Director of the NDC reports directly to the Vice President of the Research and Development Division. The Chief of the BAWTR is responsible for all operations of the BAWTR and reports to the Manager of Reactor Operations, who has cognizance over all BAWTR and CEL operations.

A Safeguard Committee will advise the Director of the NDC regarding safety and technical soundness of changes or proposals under consideration for the NDC. The Safeguard Committee will act as an independent group to review operation and changes to the facility. This committee will be made up of senior scientists from the Research and Development Division and the AED. On matters of safety, this group will advise the cognizant NDC managers, and their reports will also go to the Manager of the AED in Lynchburg. They will periodically review the design and operation of all experiments and will review all novel experiments prior to installation.

The key staff members of the BAWTR will be recruited from people who have had considerable practical experience with the everyday problems of reactor operation. Much of the experience gained in operating the CEL facilities will be used in operating the BAWTR. The key staff members will train the additional operators and shift leaders during the initial performance testing of the reactor. A training officer on the staff will study the educational and training needs of the organization and make arrangements for the required instructions.

Reporting to the Chief of the BAWTR are two groups: one involved principally with reactor operations, and the other with project engineering for experiments being performed within the BAWTR. The first group is devoted exclusively to reactor operations and experimental observations. The Operations Supervisor will have four shift

leaders reporting to him. Each shift leader will be licensed for BAWTR operation. The responsibilities of the Operations Group will be as follows: (1) perform all maintenance required to keep the BAWTR at peak performance, (2) handle all fuel cycle work associated with the BAWTR core, (includes ordering of new fuel, and unloading and disposition of spent fuel elements), (3) take all data during any irradiation period (includes both reactor operational and experimental data), and (4) perform any special critical experiments or experiments designed to measure reactivity effects of various devices to be ultimately tested at high power.

A project engineer from the Project Engineering Group will be assigned to each irradiation experiment being conducted within the BAWTR. In practice, one project engineer may have several projects under his jurisdiction. Responsibilities of a project engineer will depend on the type of project involved. The principal variation is the manner in which he participates in the design of the experiment, as well as in the actual performance evaluation of the experiment. In every experiment, a project engineer must satisfy the Operations Staff that the experiment will be carried out safely. The project engineer will perform the following duties: (1) coordinate all design and fabrication work associated with each experiment, (2) coordinate nuclear and thermal analysis needed in optimizing irradiation experiments, (3) work with the BAWTR Operations Supervisor to establish the necessary operating procedures and data-taking procedures to be employed during irradiation cycles for his assigned project, (4) participate in coordinating all operational efforts during the period of transition to a shutdown wherein an experiment is removed and a new one installed, and (5) after irradiation, participate in the transfer operations to remove the unit to a hot cell or for transport to an outside area by a cask. For the project engineer to more fully understand and apply the safety aspects of the reactor to the design of experiments, he will be required to spend time within the BAWTR Operations Group.

6.2. Operating Procedures

This section of the report sets forth general operating rules and regulations covering all phases of operation of the BAWTR to ensure that these operations are well within the safe limits established by the

technical specifications. The procedures contained in this section are outlined below.

1. Initial approach to criticality.
2. Post-critical testing.
3. Normal operation at power.
4. Recovery from scram.
5. Shutdown procedures.
6. Refueling of core.

6.2.1. Initial Approach to Criticality

6.2.1.1. Prerequisites for Initial Loading

Prior to initial loading of the fuel elements all systems associated with the reactor will be tested for correct operation and mechanical stability as follows:

1. The primary system and pool will be operated under forced circulation with a strainer installed to clean up the system. Demineralized water conditions will be established within acceptable limits.
2. Proper operation of outlet header will be checked.
3. The control rod drive mechanisms will be tested for operation, rate of withdrawal and insertion, scram, and automatic insertion capability.
4. Current response from neutron detectors will be checked with a source.
5. All trip circuits and safety devices will be checked for proper settings, response, and appropriate alarm and shutdown functions.
6. Area radiation monitoring systems will be checked for operation and response.
7. All process equipment and systems including pumps, motors, valves, and demineralizers will be hydrostatically and functionally tested.
8. All test data will be recorded and filed.

9. The beryllium reflector pieces will be installed with the shim rods inserted.

10. The fuel elements containing the safety rods and the regulating rod will be inserted and their rod drives mounted in place.

11. A startup source will be installed.

6.2.1.2. Instrumentation Employed

All nuclear detectors will be in a normal position with respect to the source, and the instrumentation system will be checked for adequate count rate or output currents. In addition, if auxiliary instrumentation is provided for the initial loading, it will be functionally tested. All neutron detecting channels will be checked for indication by withdrawing and inserting the neutron source. There will be a minimum of two linear-level channels and one period channel in operation in the safety system. Two count-rate channels will be in operation.

6.2.1.3. Method Used to Predict Critical Loading

Fuel elements will be carefully loaded into the core, one element at a time, and count rates recorded. A reciprocal multiplication curve versus fuel loading will be constructed and an estimate of critical mass determined. After the addition of each fuel element a new estimate of critical mass will be made. Count rates will be taken with and without various control rods inserted in the core. The final core loading will be such that the reactor will be critical on the regulating rod with the two safety rods fully withdrawn, and shim rods withdrawn as required.

6.2.1.4. Procedures Used During Loading

The following procedures will be adhered to during loading:

1. A minimum of two people will be located at the pool level.

2. A prearranged loading diagram will be available, and the position of the fuel element as it is lowered into the core will be checked by at least two people.

3. A licensed reactor operator and required assistants will be stationed at the control room console.

4. Communication will be maintained between the containment vessel and control room through an intercom system.

5. Count rates will be taken after the addition of each fuel element with various control rod positions, and the data plotted and analyzed before the addition of another element.

6. A senior licensed operator will be present during loading operations.

6.2.1.5. Safety Precautions

The following safety precautions will be observed during loading:

1. The nuclear safety instrumentation will be in operation during the loading of the core.

2. The area monitoring system will have been checked and in operation.

3. Portable radiation survey instruments will be in operation at the pool level.

4. All personnel other than those actually involved in the loading of the reactor will be cleared from the containment area.

5. Observers will be restricted at the discretion of the operator.

6. The positioning of fuel elements will be checked by at least two people to ensure proper placement.

7. Personnel involved in the loading operation will be thoroughly familiar with equipment and loading procedures.

8. No fuel additions will be made until the senior reactor operator has studied the data and authorized further loading.

6.2.2. Post-Critical Testing

6.2.2.1. Low-Power Tests

Low-power tests will include the following:

1. The worth of control rods will be measured to confirm calculational models and to ensure adequate control and shutdown capability.
2. Sensitivity of nuclear instrumentation will be tested to ensure adequate response at low power levels and overlap of instrumentation at increasing power levels.
3. Temperature coefficient measurements will be made at low power levels to confirm calculations.
4. Gross flux mapping will be made to determine flux distributions throughout the core.
5. Initial calibration of nuclear instrumentation will be performed to establish the approximate power level of the reactor.

6.2.2.2. Power Tests

Power tests will include the following:

1. Calibration of intermediate and power channels will be performed to establish instrumentation response and indication of reactor power.
2. Health Physics personnel will perform shielding surveys to ensure that adequate shielding exists at all power levels.
3. Temperature and power coefficients will be measured to confirm calculations.

6.2.3. Normal Operating Procedures

The following operating procedures will be maintained when the reactor is at power:

1. There will be a minimum of two Operations personnel in the central control room at all times, one of which shall be a licensed reactor operator.

2. The reliability of instruments shall be frequently checked by intercomparison, heat balance, and by such internal testing circuits as are provided.

3. Regular periodic recording of significant process data and nuclear instrumentation readings on log sheets will be carried out by the Operations Group.

4. There will be frequent observations of nuclear controls and critical process variables that are indicated, recorded, and controlled from the reactor control room.

5. There will be regular periodic checks of equipment operation to minimize failures or malfunctions. Standby equipment throughout the reactor systems will be energized periodically.

6. Upon indication of equipment malfunction or deviation of operating parameters beyond permissible limits, immediate action must be taken by the operator to correct the condition or shutdown the reactor.

6.2.4. Recovery From Scram

Reactor startups following an unscheduled shutdown will not be made unless the cause of the shutdown can be identified and safely corrected. The startup shall not be performed under time duress. Startups will include resetting trip circuits and ranges of flux monitors before withdrawing rods. A senior reactor operator will be present at recovery from scrams resulting from high flux levels, high radiation levels, or experiment failure.

6.2.5. Normal Shutdown Procedures

Shutdown procedures will be designed for the servicing of experiments, maintenance of auxiliary equipment, or preparing the reactor for fuel changes. These procedures will vary according to the type of shutdown planned. Precautions will be taken to ensure that unplanned or accidental criticality is not established, exposure of personnel to radioactivity is kept to a minimum, and equipment is placed in a suitable condition. The control rods will be fully inserted into the core, and the drive power will be disconnected during periods of extended shutdown.

6.2.6. Refueling of Core

Refueling operations will be accomplished under conditions designed to assure safety of personnel and the reactor. Water level in the reactor pool will be maintained within the normal limits. Constant radiation monitoring will be performed during refueling operations. The elements and test specimens will be removed, one at a time, using a prearranged plan. The unloading plan will be prepared by the Reactor Operations Group and checked by the Chief of the BAWTR. The elements will be transported under water to the storage canal where they will be placed in designated positions in storage racks. Complete records of fuel element inventory in the storage racks will be maintained. The elements will remain in the storage racks for a decay period. New fuel elements may be handled without shielding.

6.2.7. Operating Limitations

The following operating limitations shall be strictly adhered to:

1. The BAWTR, excluding experimental fuel, shall not be operated at a power in excess of 6 MW-t.
2. The BAWTR shall not be operated at a power in excess of (200 kw) without forced cooling.
3. The total excess reactivity above clean, cold, critical shall not exceed 19%.
4. The maximum reactivity addition rate shall be restricted to $5.5 \times 10^{-4} \Delta k/\text{sec}$ above $k = 0.99$.
5. The maximum worth of a fuelcd test insert shall be $9\% \Delta k/k$.
6. The maximum worth of voiding the central test insert shall be $0.8\% \Delta k/k$.
7. The maximum worth of any individual capsule insertion or removal shall be $1.5\% \Delta k/k$.
8. At least two safety rods shall be cocked and available for shutdown during high-power operation.

9. One startup channel, two intermediate channels, and two power channels must be operative during startup.

10. One intermediate and two power channels, not used for automatic control, must be operative during power operation.

6.3. Safety Procedures

6.3.1. Radiation Protection

Maintenance will be performed under strict regulations and with approval of the Health Physics Group. During the performance of work in any contaminated area, the Health Physics Group will be responsible for monitoring and assuring safe working conditions for the maintenance workers and assuring that proper protective clothing is worn when needed.

All individuals who work with radioactive materials and sources of ionizing radiation or who may be exposed to significant levels of external radiation will be trained in the proper use of portable survey instruments. Portable direct-reading radiation survey instruments of the Geiger-Muller (G-M) and ionization type, portable and fixed air sampling equipment, and counting-room equipment (including gas flow, proportional, scintillation, and end-window GM counting systems) are available for Health and Safety assays. Routine monitoring for exposure or contamination control will be accomplished by the individual or group directly concerned whenever practicable. When the safety of personnel involved critically depends on proper techniques or operating procedures, a work permit system will be utilized and Health and Safety personnel will be actively involved in the work situation.

Personnel monitoring equipment consists of portable monitors, hand-and-shoe counters, and count-rate meters. The latter two instruments, equipped with survey probes, are located at strategic points throughout the facility for personnel contamination surveys.

6.3.2. Personnel Monitoring Devices

All individuals admitted to the BAWTR will be issued and required to wear personnel monitoring devices. Personnel assigned to work in areas where significant exposures are possible, but not probable, will wear a beta-gamma film badge. Personnel assigned to work with

sources of ionizing radiation or assigned to areas where potentially high levels of radiation are probable will wear film badges and pocket ionization chambers.

6.3.3. Bioassay Program

All individuals assigned to work with radioactive materials or to work in hazardous areas where potentially high concentrations of air-borne radioactivity may be encountered or where significant contamination of the person or work area is possible will be required to submit biological specimens. Specimens may be collected on a nonroutine as well as on a routine basis. Where iodine exposures are suspected, thyroid uptake measurements will be made with a scintillation counting system.

6.3.4. Personnel Decontamination

The decontamination of personnel will be accomplished as quickly as possible. Emphasis will be placed on self-help. Mild hand soaps and water will be used as the initial step in the decontamination process. All injuries involving contamination, potential contamination, or injuries occurring in controlled zones will be reported to the Health and Safety Officer with referral to a physician as indicated.

6.3.5. Medical Examinations

All B&W personnel on an assigned status at BAWTR will be given routine physical examinations by physicians at least once a year. These examinations will be directed toward determining the normal or assumed pre-irradiation condition of each individual; special emphasis will be placed on the detection of any abnormalities that might later be associated with radiation exposures.

6.3.6. Personnel Exposure Records

Exposure records will be maintained on all individuals for whom personnel monitoring is required. All console keys and bypass controls shall be under cognizance of the Operations Supervisor and shall not be used without his specific approval, or the approval of an alternate delegated by the BAWTR Section Chief. No entry to the containment and primary equipment cell shall be made without the

approval of the control room operator. A log shall be maintained of entry and exit from these areas. The readings of the remote area monitoring system in these areas shall be recorded in the log at time of each entry.

6.3.7. Personnel Protection

Protective clothing will be worn by individuals working with radioactive materials, contaminated materials or items of equipment, or while working in zones where significant radioactive contamination is likely.

Also, respiratory protection and other standard safety items will be available for use in hazardous areas. A safe work-permit system will be used to control specific jobs where unusual or abnormal hazards are likely to be encountered or where close coordination of work affecting other operations is necessary.

Protective clothing (coveralls, lab coats, shoe covers, surgeons caps, and gloves) will be available for routine use in controlled zones. Specific items of protective clothing required in each controlled area will be posted at all entrances. Additional items of protective clothing or safety gear will be specified by the Health and Safety Group as required.

Protective clothing or other apparel and safety equipment will be monitored at established transition points when going from zones of higher contamination to zones of lower potential contamination. Garments contaminated to levels exceeding that permissible for the lower level zones will be exchanged for clean items at the transition point. Instructions will be given on the proper use of protective clothing.

6.4. Emergency Procedures

Emergency and evacuation procedures will be rehearsed periodically to ensure that all personnel will know the rules to follow in case of an emergency. In the event of an incident involving high radiation, personnel will be alerted by the use of alarm systems located throughout the BAWTR operations area. A fixed radiation monitoring system will include the following areas:

1. Cell operations area.
2. Hot-instrument repair shop.
3. Crane room and cask handling area.
4. Primary equipment cell.
5. Liquid waste area.
6. Containment area.
7. Radiochemistry lab.
8. Isolation area.
9. Control room.
10. Base of stack.

These monitors have local indication and alarms and also a read-out and alarm in the control room and Health Physics Office.

6.4.1. High-Radiation Alarm Signal

The procedures to be observed after the high-radiation alarm signal are as follows:

1. At the sounding of a high-radiation alarm (ringing bell) all personnel will leave the immediate area and Health Physics personnel will be notified.
2. Personnel will not re-enter the area until given Health Physics clearance.
3. The Health Physicist will decide if the hazard is localized or if it affects the entire building.
4. If it is decided to evacuate the building, the evacuation alarm (horn) will be manually actuated. Manual pushbuttons to actuate the building evacuation alarms are located in the reactor control room and the Health Physics Office.

6.4.2. Evacuation

Procedures to be observed in evacuating the danger area are as follows:

1. Upon signal to evacuate, all operations will be stopped. The reactor operator will immediately scram the reactor, leaving all instrumentation on, and will check to see that the containment integrity valves have tripped closed. All personnel will gather at the northwest corner of the parking lot in front of the building with all available portable survey instruments.

2. A roll call will be made and reported to the site Emergency Officer.

3. Visitors will be the responsibility of the persons whom they are visiting. The receptionist will have a list of such visitors.

4. An emergency team will make a preliminary evaluation to determine the extent and magnitude of radioactivity at the site and prevailing weather conditions.

5. The Health Physics Group will have emergency equipment available (protective clothing, respiratory protection, and portable survey instruments).

6. The incident will be reported immediately to the Guard Shift Supervisor at the NFP.

7. An assistant to the Health Physicist will record all dosimeters in an emergency notebook.

8. If the Emergency Officer deems it necessary, civil authorities will be notified immediately.

9. In the event evacuation of the site is necessary, the Health Physicist will determine the evacuation route to be taken. Transportation will be by private automobile.

10. If the incident is a fire involving radiation problems and off-site help is needed, Health Physics personnel will meet the off-site firemen at the main gate, provide them with the necessary protective clothing, and advise them on the radiation hazards involved.

6.4.3. Emergency Personnel Assignments

1. Emergency Officer - Manager of Reactor Operations; Chief of the BAWTR.

2. Health Physics Officer - Site Health Physicist; Assistant Health Physicist.

6.5. Security Measures

Physical security of the facility is important as a safety measure and for the protection of classified information. NDC physical security consist of the following:

6.5.1. Physical Barriers

The NDC will be surrounded by permanent security fencing; the front wall of the main building will act as part of the front border of fencing. The fencing will include two locked vehicular access gates. The front door to the main building will be locked and of heavy duty construction. The windows along the front of the main building will be of the "permanently closed" design with heavy glass construction.

6.5.2. Security of Matter in Storage

Security matter other than Special Nuclear Material (SNM) will be stored in a heavy duty steel safe-type filing cabinet. SNM will be stored in a vault-type room with an automatic alarm system.

6.5.3. Protective Alarms

A central station alarm system will be used as an aid to the full-time armed guard force. In addition to the vault alarm all doors to buildings will have complete circuit type contact alarms and all alarms will be connected to an alternate source of power.

6.5.4. Protective Lighting

The protective lighting system will illuminate all perimeter fencing with overlapping light distribution. The front face of the main building will be floodlighted. No lights that directly or indirectly use gas will be used, and all protective lighting will be connected to an alternate source of power.

6.5.5. Personnel Identification

A pass-badge exchange system will be employed at the front door of the main building during normal working hours and by the full-time guard force at other times.

6.5.6. Guard Force

The NFP Guard Force will perform all of the functions appropriate to the proper maintenance of NDC security, such as routine patrols as well as a physical inspection of the facility once every hour during other-than-normal working hours. The guards will monitor the

central station alarm system on a full-time basis and maintain a constant check on employees and visitors entering and leaving the area.

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7. ACCIDENT ANALYSIS

Figure 6.1. NDC Organization Chart

Initial operation of the BAWTR will be at a power level of 6 MW-t with the central hole occupied by an aluminum block or aluminum can described in Section 5. Later operation at 6 MW-t will include a loop of the type and design described in Section 5. The maximum power of the loop will be 750 kw.

It is anticipated that the core will be later expanded to accommodate two test loops and a core power of 12 MW-t. It is also anticipated that the two loops will be sufficiently independent so that a failure in one loop would not cause failure in the other. Acceptance of the independence of two loops, approval of operation with two loops, or approval of operation at 12 MW-t are not requested at the present time. However, analyses involving the release of fission products from the reactor are being performed for the 12 MW-t core.

The analysis of the hazards associated with reactor operation, as described in the following sections, indicate that the reactor can be operated at the site in the proposed manner without undue risk to the public.

7.1. Safety Features

Engineered, administrative and procedural, and inherent safeguards are incorporated into the BAWTR to assure that the probability of a serious accident is remote, and to assure that any credible accident will not result in undue hazard to the public. Some of these features are described below.

7.1.1. Negative Reactivity Coefficients

The reactor is designed and constructed in such a manner that it exhibits inherent self-control and regulation. Increases in power level are therefore countered by decreases in the multiplication constant of the reactor, which renders the reactor power self-limiting.

The anticipated reactivity coefficients of the reactor and associated loop are tabulated in Table 7.1. These are identified as the reference coefficients.

Table 7.1. Reactor and Loop Reactivity Coefficients

Source	Numerical value, $\Delta k / ^\circ F$
Reactor fuel moderator $\times 10^{-4}$	-1.40
Reactor reflector $\times 10^{-4}$	+0.21
Loop annulus $\times 10^{-4}$	+0.32
Fuel specimen coolant $\times 10^{-6}$	+1.00
Fuel specimen Doppler $\times 10^{-6}$	-0.25

It is anticipated that the fuel specimens being irradiated in the loop will be 4- to 10%-enriched oxide pins. Increases in the power level produce increases in the specimen temperature that in turn, produce Doppler broadening and increased neutron absorption in the nonfissioning material. The net effect is a small decrease in reactivity.

The coolant flowing past the fuel specimens acts as a neutron poison. An increase in power produces an increase in the specimen coolant temperature. The resultant decrease in coolant density produces a small increase in reactivity.

The water annulus between the loop pressure tube and the guide tube acts as a neutron poison. A temperature increase in this water results in an increase in reactivity. However, since the loop time is approximately 15 seconds, the short time changes in annulus temperature are governed by internal heating of the water and heat transfer through the guide tube. The relatively small heat additions by these mechanisms make this reactivity contribution small.

Heating of the water used to cool the beryllium reflector produces a positive reactivity effect. The heat transfer from the beryllium and internal generation in the water result in only a small temperature rise. Although the reactivity coefficient is fairly large, the net reactivity addition during a rapid transient is small.

The moderator-coolant for the BAWTR fuel elements exhibits a strong negative temperature coefficient. Rapid heat transfer from the fuel plates to the coolant and the significant temperature increase of the moderator with power increases make this reactivity contribution the controlling influence on the behavior of the reactor.

Figure 7.1 indicates the response of the reactor and loop reactivity effects when a 50% step change in power level is imposed. The net response is seen to be a rapid and negative reactivity insertion with the short-time behavior controlled by the BAWTR fuel moderator-coolant.

The consequences of the reactor having coefficients differing from the reference coefficients given in Table 7.1 were investigated. The results are indicated in Figure 7.2. The three curves are the total response for the following.

1. Reference coefficients.
2. A 50% increase in the positive coefficients and a 50% decrease in the negative coefficients.
3. A 50% increase in positive coefficients, where the fuel specimen has a positive Doppler coefficient, and a 50% decrease in the negative coefficients.

The reactivity response for even these extremes is seen to be rapid and negative.

7.1.2. Low Available Excess Reactivity

The amount of positive reactivity that can be added while the reactor is critical is limited by the worth of single shim rods and rod groups, and the rate is limited by the rod speeds. The maximum worth of a shim rod group is calculated to be less than 2% $\Delta k/k$, and the maximum ramp addition rate is calculated as less than $5 \times 10^{-4} \Delta k/k$ per second. The maximum step addition of reactivity is associated with the partial voiding of the loop annulus, and is calculated to be 0.6% $\Delta k/k$. Complete voiding of the loop results in a decrease in reactivity, as described in Section 3.2. The self-controlling characteristics of the reactor, and the reactor control system, can compensate for such reactivity additions.

7.1.3. Control and Safety System

The reactor control and safety system are designed so that no single rod or rod group withdrawal will make the reactor critical. In addition, the reactor will not be critical when both safety rods are withdrawn and the shim rods are inserted.

Provision of interlocks preventing shim rod withdrawal unless both safety rods are fully cocked assures that the reactor will not become critical without adequate shutdown protection. The worth of the shim rods that are operative is limited to the extent that the reactor will be subcritical when the safety rods are inserted, regardless of the position of the operative shim rods. Automatic insertion of all withdrawn shim rods occurs immediately after a reactor scram.

7.1.4. Design Characteristics

The thermal and hydraulic design of the reactor is conservative, as shown in Section 3.2.11. The characteristics of the BAWTR are comparable to reactors that have operated successfully and safely for long periods of time. A significant example is the use of fuel elements similar to those used in the MTR, ETR, and ORR. Most of the mechanical components of the BAWTR are similar to those used and proven by service in operating reactors.

7.1.5. Containment

The use of the low-leakage containment vessel to totally enclose the reactor core and experimental loop provides additional protection from the consequences of accidents.

7.1.6. Site

The remote site is a significant safety feature of the BAWTR. The natural features of the site, such as the river and mountains, provide assurance that significant population buildup will not occur. The geological and hydrological characteristics also indicates the suitability of the site.

7.1.7. Personnel

The contribution to reactor safety resulting from the use of experienced personnel is significant. Personnel from the CEL, which

has operated without incident for more than 6 years, will be engaged in the operation of the BAWTR.

7.2. Consequences of Equipment Failure

The consequences of equipment failure have been investigated. Excluding the failure of the experimental loop, which is defined as the maximum credible accident, and the flow blockage accident described in Section 7.2.10, equipment failures do not result in the release of fission products from the core or loop.

7.2.1. Primary System Coastdown

Failure of the power supply to the BAWTR would result in release of the safety rods from their electromagnets and simultaneous flow coastdown. A more serious condition is that of loss of power only to the primary pump. This latter situation has been analyzed.

Loss of pump power would be detected by the pump current monitor, which generates a scram signal. The backup system for the detection of loss-of-flow utilizes a flow measuring device, which has a somewhat slower response than the pump current monitor.

Figure 7.3 indicates the short-term response of the system when loss of flow is detected by the flow meter at 95% full flow; the scram delay time is 250 milliseconds. The actual flow is seen to exceed the required flow during the first part of the coastdown. The curve of required flow is that flow required to prevent hot channel hydrodynamic instability that could eventually lead to burnout.

At about 20% of full flow, a pressure difference of 0.5 psi exists across the outlet header. As flow falls below this value, the header drops to permit natural circulation through the core. After the header drops, flow in the hot channel decreases and reverses as indicated by Figure 7.4. The hot spot temperature during coastdown and reversal is indicated in Figure 7.5. During the early stages of coastdown the surface temperature decreases rapidly from the full power value of about 250 F until it is only a few degrees above the coolant temperature. As flow further decreases, and reversal takes place, the fuel plate temperature rises to about 270 F. Calculations indicate that neither bulk boiling nor burnout will occur during flow coastdown and reversal. In the event that a temporary burnout did occur during

the flow reversal, more than 8 seconds would be required to raise the hot spot temperature to 500 F.

It is therefore concluded that accidents or equipment failures that cause loss of pump power will not result in the release of fission products from the core, nor will such incidents produce damage in the fuel elements.

7.2.2. Cold-Water Accident

The possibility of a serious accident involving the introduction of cold water to the BAWTR core is nonexistent. The core inlet water is only about 13 F cooler than the outlet water, and there is no other significant source of cooler water that could be injected into the core. Any failure resulting in gross admission of the pool water to the core would add approximately 0.2% reactivity, which can readily be compensated for by the reactor and the safety system. The cold water accident is therefore not a significant hazard in operation of the BAWTR.

7.2.3. Loss of Pool Water

The walls of the reactor pool are reinforced concrete, and they are coated with plastic to provide a waterproof lining. Sudden, gross failures of the pool walls are not considered credible. Small leaks would be readily detected before they became serious, and any drainage would be confined within the containment vessel and would be treated as contaminated waste.

Significant loss of pool water may occur through failure of the primary system or through inadvertent opening of the pool gate when the storage pool is drained. Excessive radiation level in the storage pool area restricts reactor operation when the storage pool is drained; procedural safeguards are also employed to prevent opening of the pool gate. In addition, an automatic scram would occur due to pool level decrease, and an alarm would sound in the control room when the gate was moved from its fully closed position. In the event that the pool gate was opened when the storage pool was drained, the equilibrium level would be approximately 2.5 feet above the active core.

7.2.4. Primary System Failure

The consequences of serious failure of the primary system have been considered. The worst credible failure is defined as complete severance of the primary system pipe at the primary pump.

Following the failure of the primary pipe, the pool water would drain into the pump sump until the level was the same in the pool and the pump sump. This equilibrium level would be established in a minimum time of about 90 seconds; the level would be 2 to 3 inches above the top of the core.

Reactor scram would be initiated by the pool level detector and would be backed up by loss of flow scram, pump current scram, and high containment radiation level as the core shield water was lost. The power would be rapidly reduced, and no core melting or fuel damage would occur.

The water that drains into the pump sump would be confined in the primary equipment cell and would be treated as liquid waste. Activity that might become air-borne would be retarded in its escape to the uncontrolled atmosphere by the primary equipment cell, which has a single door and a single ventilation discharge. Significant concentrations of fission products do not normally exist in the pool and primary system water, and therefore would not represent a serious hazard.

Reactor and loop accidents that could cause failure of the primary system piping are not considered to be credible. The weakest point in the primary system is the heat exchanger, and this equipment normally operates at a pressure that is at least a factor of two below its design pressure. Although it is credible that failure of the high pressure loop could hurl missiles in the direction of the primary pipe, it is not considered credible that such missiles would penetrate the pipe after having traveled through more than 10 feet of water and at least two metal baffles.

Manual valves are provided on the inlet and outlet pipes of the primary system in order that the pump, heat exchanger, and other equipment may be isolated from the reactor pool if required. The handles for operation of these valves are located in the room above the primary equipment cell.

It is concluded that failure of the primary system will not result in the release of significant quantities of radioactive material from the reactor building, and therefore will not present a hazard to the public.

7.2.5. Secondary System Failure

A loss of the secondary cooling system actuates alarms in the control room. As a result, the reactor could be scrammed and operated on convection cooling with no difficulty.

In the unlikely event that the reactor continued to operate after failure of the secondary system, the temperature of the core inlet water would increase. The reactor would experience a power decrease because of the negative temperature coefficient. Compensation of this reactivity loss by the regulating rod would result in a maximum temperature rise of less than 60 F, but the inlet and outlet temperature sensors would cause scram well before this occurred. Therefore, loss of the secondary cooling system does not result in reactor damage.

7.2.6. Loss of Instrument Air

The primary and secondary pressure transducers and most of the nonnuclear instrumentation for the plant will be operated by compressed air. On low air pressure or with a complete loss of compressed air, an alarm signal will be actuated. The reactor coolant flow is not affected by the loss of instrument air. The nuclear instrumentation does not utilize compressed air and will monitor the nuclear behavior of the core at all times.

7.2.7. BAWTR Fuel Element Failure

Fuel element cladding failure may result in the release of fission products to the primary coolant system. These would be removed by the primary coolant purification system to maintain an acceptable level of activity in the water. Previous and latter sections of this report indicate that gross failure of the core is not credible.

7.2.8. Excess Fuel Addition

Handling of fuel and experiments in the core will be carried out when the reactor is shut down, and in accordance with the operating procedures. The design of the components, fuel, experiments, and

reflector assures that there can be no inadvertent interchange of these parts and an accident cannot result from faulty location. Once the reactor is loaded, the closely-packed lattice arrangement of the core prevents the addition of fuel.

The only foreseeable accident caused by an addition of excess fuel could result from dropping or deliberately placing an additional fuel element on top of the core. This accident was investigated by calculating the amount of reactivity added by placing one element across the top of the core. If the element could be positioned across the top of the core, it would add less than 0.5% reactivity to the core at normal operating conditions. This would immediately scram the reactor due to a power increase, and no core damage would occur.

7.2.9. Reactivity Additions

The effects of accidental reactivity additions have been determined. The self-regulating characteristics of the reactor and adequacy of the safety system are demonstrated. The only scram utilized in the investigation was the 130% overpower scram.

Ramp additions of reactivity were made beginning with the reactor 1% subcritical. This represents the worst case for ramp additions since the reactor is already on a period when it becomes critical, and the low level at the start of the ramp permits the greatest accumulation of reactivity before the scram level is reached.

Figure 7.6 indicates the peak power and the integrated power above 9 MW-t as a function of reactivity addition rate. The reactivity coefficients are the reference values given in Table 7.1, and the scram delay time of 0.5 milliseconds is the maximum design value. The significance of the 9 MW-t power level is that higher power levels at the maximum coolant inlet temperature approach the point of hydrodynamic instability and eventual burnout. The conservative assumption is made that the fuel plate becomes insulated at the 9 MW-t level.

The hot spot would reach the melting temperature when 6 MW-seconds of energy are released with the power above 9 MW-t. To achieve melting at the hot spot would require 11 MW-seconds. A section of the core having the average power density would reach the

melting temperature for 18 MW-seconds of operation above 9 MW-t, and would achieve melting when 33 MW-seconds of energy are released above the 9 MW-t level.

The calculated maximum reactivity addition rate for the rod in its most reactive position is $3.6 \times 10^{-4} \Delta k/k$ per second. Withdrawal of a four-rod group as the last rods to be withdrawn would result in a maximum reactivity addition rate of $5.4 \times 10^{-4} \Delta k/k$ per second. This latter rate would only occur through gross error in the prepatched rod program and failure of the operator to observe that four rods were moving.

Figure 7.6 indicates that, even for the rod misprogramming and continuous withdrawal, the 9 MW-t level would not be exceeded. Melting at the hot spot would require a reactivity addition rate exceeding $25 \times 10^{-4} \Delta k/k$ per second, five times the maximum credible rate due to rod withdrawal. Melting of a significant fraction of the core would require a reactivity rate many times that deemed to be credible.

The effects of other-than-reference coefficients were investigated. The results are indicated in Figure 7.7. The scram delay time was varied from 50 to 150 milliseconds, and the coefficients were varied from the reference values to those values representing a 50% increase in positive coefficients (Doppler included) and a 50% decrease in negative coefficients.

Peak power is influenced mainly by the values of the coefficients, particularly for the higher reactivity addition rates. Integrated power above the 9 MW-t level is determined, for the most part, by the scram delay time. Even under these extreme conditions, release of fission products from the reactor core will not occur for credible ramp additions of reactivity.

The consequences of step additions of reactivity to the core while operating at full power are shown in Figure 7.8. It is seen that melting at the hot spot requires a step addition of reactivity exceeding $0.8\% \Delta k/k$. Accidental voiding of the loop annulus introduces a maximum transient step addition of $0.6\% \Delta k/k$. The hot spot would not reach the melting temperature for reactivity steps of this magnitude.

Figure 7.9 indicates the effects of varying the reactivity coefficients and scram delay time as previously described. Peak power

is not significantly influenced by these changes. The integrated power above the 9 MW-t level is governed by the scram delay time. The figure indicates that melting at the hot spot will not occur for the reactivity step addition associated with loop voiding.

The aluminum block or can that occupies the central hole during operation without a test loop prevents the addition or movement of material in this space. The design of capsules or specimens placed anywhere in the reactor will limit the reactivity addition due to any credible failure to less than $0.60\% \Delta k/k$. It is therefore concluded that credible reactivity additions, either ramp or step, will not cause the release of fission products from the BAWTR fuel elements.

7.2.10. Fuel Element Flow Blockage

An accident is postulated in which foreign material is introduced into the reactor system in such a way as to block the coolant channels between fuel plates of the core elements. Complete flow blockage of the channels would result in fuel-plate melting. It is not expected that more than half the fuel plates in a single 19-plate element would melt as a result of this accident. Therefore, the amount of fission products released would not be greater than that in a region generating 500 kw of power.

The meltdown and release of fission products would not be accompanied by a pressure rise in the containment, and the escape of fission products to the atmosphere would be about 0.10% per day because of atmospheric pressure fluctuations. The required exclusion distance is 50 meters, and the low population distance is 100 meters.

When a loop is not in the core, the flow blockage accident may be considered the maximum credible accident. Nevertheless, the more serious loop accident discussed in the following section (Section 7.2.11) is given primary consideration when evaluating the reactor site.

7.2.11. Maximum Credible Accident

The maximum credible accident (mca) is defined as failure of the experimental loop followed by release of fission products from the loop fuel sample to the containment atmosphere. The assumptions employed in the analysis of the mca are as follows:

1. The loop volume is the maximum permissible value of 18 cubic feet.

2. The average loop temperature is 625 F.

3. The loop power is the maximum permissible power of 750 kw.

4. At the time of the loop failure, 100% of the noble gases, 50% of the halogens, and 1% of the solid fission products in the loop are released to the containment vessel. Of the halogens released to the containment vessel, 50% are immobilized so as to prevent escape from the containment vessel.

5. The containment leak rate is the maximum value of 1% of the contained free volume per day for the accident duration.

6. Atmospheric dispersion of the material that escapes from the containment is according to the Sutton equation for a ground release. The meteorological parameters are $C_y = 0.40$, $C_z = 0.07$, a stability parameter of 0.50, and a wind speed of 1 meter/second. Wind speed and direction are assumed constant for the accident duration, and no depletion of the diffusing cloud through fallout or rainout is assumed.

7. Release of the loop contents to the containment is instantaneous.

These assumptions are all considered to be conservative.

Failure of the loop in the loop equipment cell would result in flashing of the loop water to steam. The pressure of the loop cell and the above grade containment vessel would increase to 5 psig as air and steam are expelled from the equipment cell. The expected pressure history following loop failure is given in Figure 7.10. For the curve labeled "without air conditioning", the only heat sink is the containment vessel steel above grade, and the effects of the concrete and pool water are neglected. The second curve describes the pressure history with the containment vessel air conditioning system functioning. Even without credit for the containment cooling system, the pressure would be less than 1.5 psig in 2 hours after loop failure, and the leak rate would not exceed 0.4% per day. The leak rate would decrease to 0.25% per day in less than 8 hours. The containment cleanup system

would remove more than 90% of the air-borne halogens within a few hours. However, no credit is taken for any of these effects.

Failure of the loop either in the core or in other parts of the pool would result in the escaping steam being condensed by the pool water. Significant washout of the solids and halogens would also be expected. The insulation jacket is designed such that it would be expected to maintain its integrity despite the occurrence of a loop failure. Failure of the insulating jacket as a result of pressure tube rupture may cause outward movement of the core, but this would tend to reduce reactivity. Since the safety rods are totally enclosed in a guide tube, scram would not be prevented. Damage to the core, even to the extent of crushing a fuel element, would not result in the element reaching the melting temperature as long as it remains immersed in the pool water. It is expected that an undamaged BAWTR fuel element could be safely cooled in air within 15 minutes after shutdown.

A transient reactivity addition of up to 0.6% $\Delta k/k$ may occur because of partial voiding of the loop annulus. As the void fraction increases beyond 0.60, the net reactivity added becomes negative. The analyses described above indicate that the reactivity addition associated with loop rupture in the core will not cause the release of fission products from the BAWTR fuel. Also this accident will not cause pressure buildup in the containment vessel because of steam condensation by the pool water. Therefore, a loop rupture in the core is less severe than a rupture in the loop equipment cell.

The analysis of the mca according to the previously stated assumptions yields the iodine inhalation dose curves given in Figure 7.11. Using the site criteria of 10 CFR, Part 100, and the supporting document "Calculation of Distance Factors for Power and Test Reactors", TID-14844, the required exclusion zone for thyroid dose is found to have a radius of less than 50 meters. The required low population zone is seen to be 550 meters, less than 0.4 mile.

Direct gamma dose from the fission products in the containment as a function of distance is given in Figure 7.12. Using the 25-rem criterion of 10 CFR, Part 100, it is seen that the required exclusion distance for whole body dose is 70 meters. The low-population distance required is about 190 meters. The whole body dose due to submersion in the diffusing cloud is insignificant.

The required exclusion is 70 meters, and the required low population distance is 550 meters. The exclusion area does not include the railroad, CEL, or NFP. The distance to the natural boundary of the left bank of the river is about eight times the required exclusion distance. The available low-population distance of two miles exceeds the required low-population distance by a factor of five. A summary of doses due to the maximum credible accident is given in Table 7.2.

It is concluded that the mca will not cause fission product release from the core fuel elements. Also, the occurrence of the mca does not result in undue hazard to site personnel or to the public.

Table 7.2. Doses (rem) Due to Maximum Credible Accident

	CEL	NFP	River exclusion		0.5 mi
			900 feet	1300 feet	
Thyroid dose, iodine					
Exclusion, 2 hr	4.5	7.2	19	10.5	3.83
Low population, 30 days	2200	335	880	510	182
Direct radiation					
Exclusion, 2 hr	4.5	0.06	0.73	0.20	0.005
Low population, 30 days	50	0.23	5.3	0.95	0.018

7.2.12. Hypothetical Accident

The mca previously described is the worst accident with a credible probability of occurrence. Although core meltdown is considered to be incredible, the consequences of a hypothetical accident that assumes this to occur have been determined. The analyses of the previous sections indicate that the hypothetical accident does not occur as a result of other accidents.

It is assumed that the core has operated at a power level of 12 MW-t for an extended period. At the on-set of the hypothetical accident, 100% of the noble gases, 50% of the halogens, and 1% of the solids are assumed to be released from the core. Of the halogens

released from the core, half are assumed to be immobilized inside the containment. There is no pressure buildup due to the hypothetical accident, but atmospheric pressure variations are assumed to produce leakage of 0.1% per day. The meteorological parameters assumed for the mca are also assumed for the hypothetical accident.

The results of the calculation of the iodine doses are indicated in Figure 7.13. The exclusion distance is about 60 meters; the low-population distance is 800 meters. These distance requirements are readily met by the site.

The dose due to direct radiation from the containment vessel is shown in Figure 7.14. The required exclusion distance is about 220 meters. Since this exceeds the thyroid dose criterion, direct dose is controlling. The low-population distance is 550 meters, but this is less than that required by the thyroid dose.

Table 7.3 gives a summary of the doses due to the hypothetical accident. It is concluded that the Mt. Athos site meets the criteria of 10 CFR, Part 100, even for the hypothetical accident.

Table 7.3. Doses (rem) Due to Hypothetical Accident

	CEL	NFP	River exclusion		0.5 mi
			900 feet	1300 feet	
Thyroid dose, iodine					
Exclusion, 2 hr	72	11	28	16.5	5.9
Low population, 30 days	3800	570	1530	870	300
Direct radiation					
Exclusion, 2 hr	76	--	11.5	2.5	0.08
Low population, 30 days	900	--	82	14.2	0.26

Figure 7.1. Response of Reactivity Feedback Contributions to 50% Power Charge (Reference Coefficients)

Figure 7.2. Response of Reactivity Feedback Contributions to 50% Power Change (50% Change in Coefficients)

Figure 7.3. Primary System Flow Coastdown

Figure 7.4. Hot Channel Flow Coastdown and Flow Reversal

Figure 7.5. Hot Spot Temperature During Flow Coastdown and Flow Reversal

Figure 7.6. Ramp Reactivity Additions From 1% Subcritical
(Reference Coefficients, 85 ms Scram Delay)

Figure 7.7. Ramp Reactivity Additions From 1% Subcritical
(Reference to 50% Coefficients, 50 to 150 ms Scram Delay)

Figure 7.8. Response to Step Reactivity Additions From Full Power
(Reference Coefficients, 85 ms Scram Delay)

Figure 7.9. Response to Step Reactivity Additions From Full Power
(Reference to 50% Coefficients, 50 to 150 ms Scram Delay)

Figure 7.10. Containment Pressure History for Maximum Credible Accident

Figure 7.11. Iodine Inhalation Doses From Maximum Credible Accident

Figure 7.12. Direct Dose From Containment — Maximum Credible Accident

Figure 7.13. Iodine Inhalation Doses From Hypothetical Accident

Figure 7. 14. Direct Dose From Containment — Hypothetical Accident

END