Rensselaer Polytechnic Institute
Critical Facility
Safety Analysis Report

Docket No. 50-225

License No. CX-22

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Contents

1 Introduction

RPI CRITICAL EXPERIMENTS FACILITY

Construction of the Facility was completed in July of 1956 by ALCO Products, Inc. Originally the Facility was constructed as a laboratory in which reactor experiments, necessary for the design and development of military and commercial power plants, could be performed in a safe and convenient manner. The experiments performed here were either critical experiments or zero-power experiments, all of which took place at power levels below 100 watts. In years of operation the volume of technical data generated is most impressive. In 1964 Rensselaer Polytechnic Institute assumed operation of the Facility for the instruction of graduate students in the Institute's Department of Nuclear Engineering and Science, and for research and testing for the Institute, AEC, and others.

2. SITE CHARACTERISTICS

2.1 Location

The facility is at Schenectady, N. Y. bordering on the Mohawk River.

The relationship of this location to the city of Schenectady and surrounding area is shown in Fig. 2.1.

The city of Schenectady is geographically situated in the eastern section of Schenectady County which has an area of 209 square miles. The Schenectady area is more generally considered to be the western boundry of a larger metropolitan area - the so-called Capitol District - composed chiefly of the cities of Albany, Troy, Watervliet, Rensselaer, Cohoes, and Schenectady. The center of this area is in the vicinity of the Albany Airport which is about 7 miles to the southeast of the proposed facility.

Other points of interest indicated in Fig2.1 include the facts that the site is one mile north-northeast of the commercial center of the city and about 3 miles downstream from the public Schenectady water supply. This supply is taken from drilled wells 56 - 70 feet deep near the Mohawk River in the vicinity of Lock 8.

The Schenectady urban area as defined by the U. S. Census Bureau includes the city, the village of Scotia, and parts of the towns of Colonie, Glenville, Niskayuna and Rotterdam. The estimated total resident population within various distances of the site is indicated in the following table and Figue 2.2.

Population Density in the Vicinity of Site

Miles from Site	Estimated Total Population	Direction from Site
0.5	3,500	E - SE - S
1.0	18,476	E - S - SW
2.0	61,807	E - S - WNW
3.0	104,026	E - S - WNW
4.0	121,480	E - S - WNW

The nearest commercial establishment is 700 feet distant.

The nearest residence is 1150 feet to the southeast.

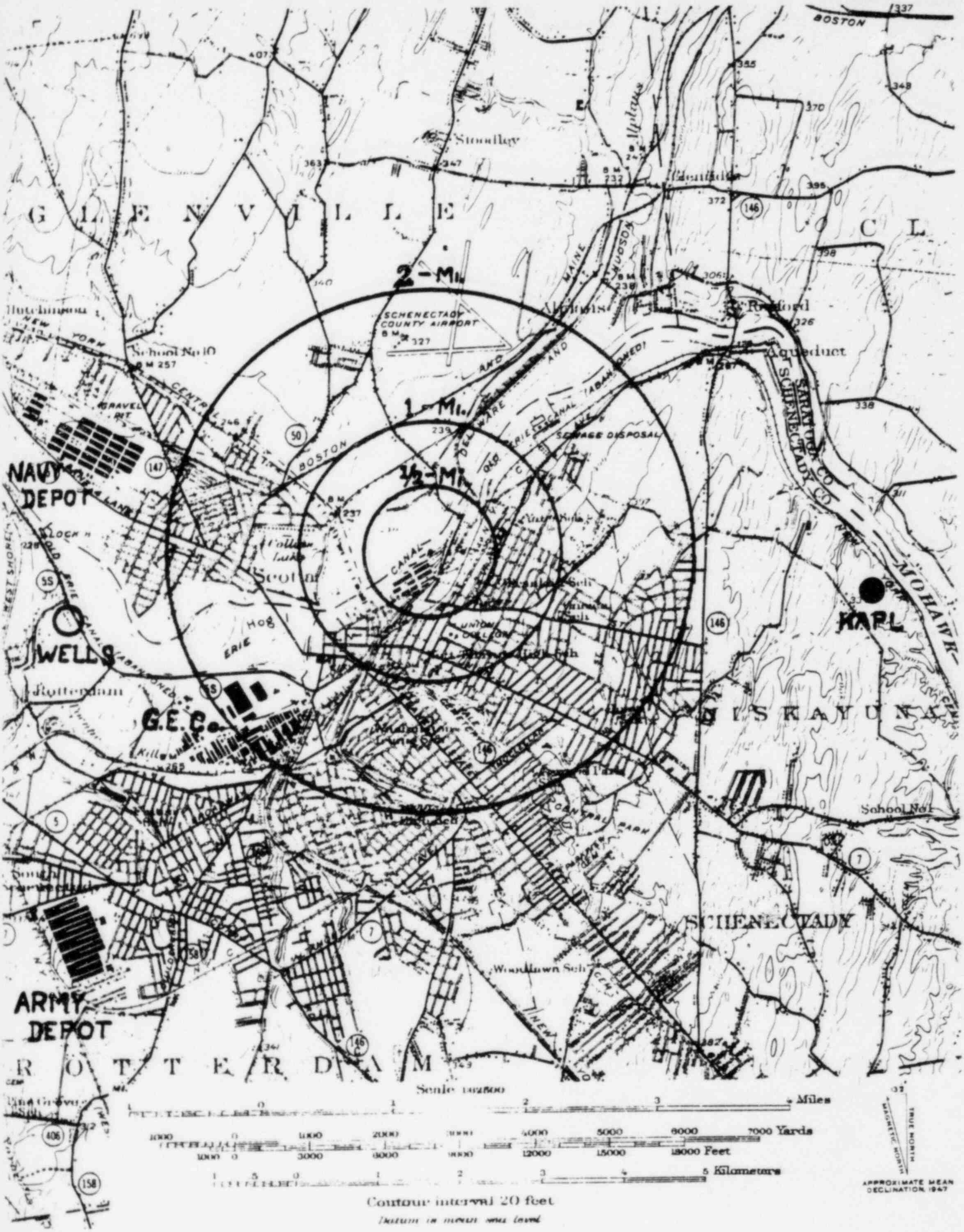


Figure 2.1 - Map of Schenectady Area

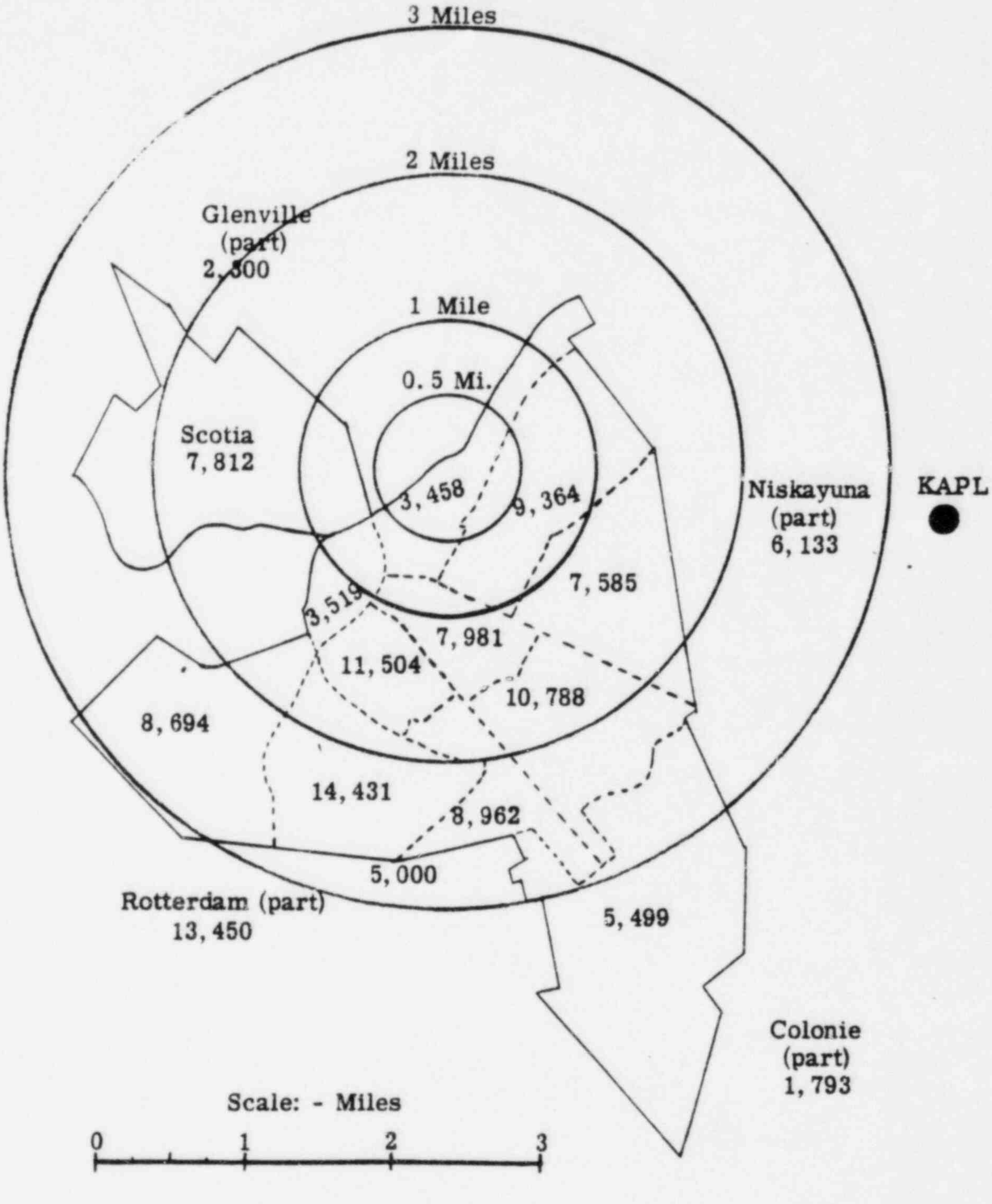


Figure 2.2 - Population Distribution of Schenectady Urban Area

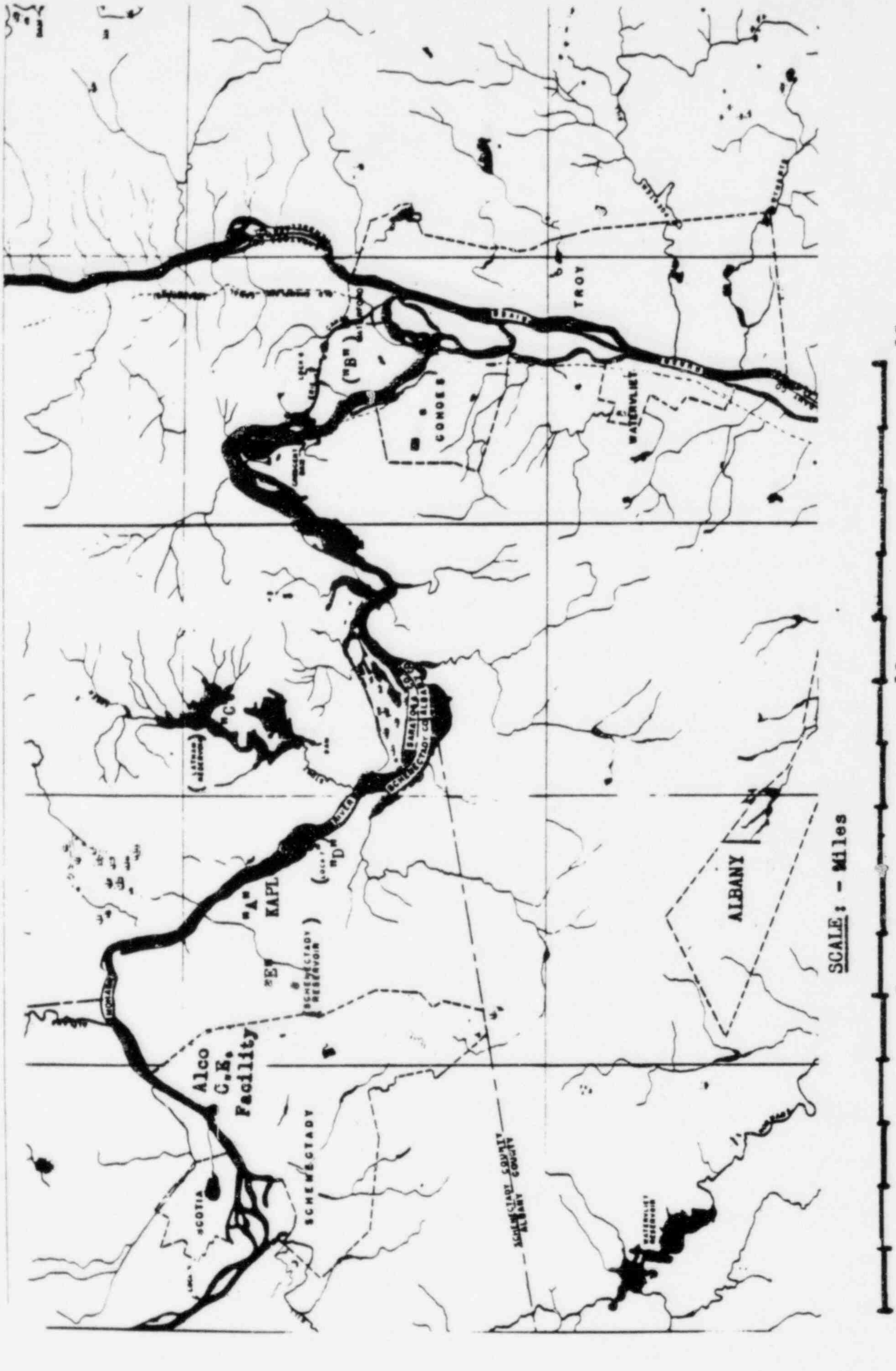


Figure 2.3 Waterways in the Schenectady Area

2.2 Geology and Hydrology

Topography and Drainage

Schenectady County lies almost entirely within the lowland area bounded by the Adirondack Mountains on the north and by the Helderberg escarpment of the Allegheny Plateau province on the south. The lowland has been deeply eroded and has considerable relief. The altitude of the County ranges from about 200 feet above sea level in the flood plain of the Mohawk River to about 1100 feet at Glenville Hill on the north side of the Mohawk, and to more than 1400 feet in the hills near the center of the County on the south side of the Mohawk.

The Mohawk River enters the County at the village of Hoffmans and flows south-easterly for about 9 miles on a flood plain about a mile wide, until it reaches the city of Schenectady. There the flood plain flares out to a width of more than 2 miles and the river changes its direction of flow to the northeast. About 4 miles farther downstream the river bends again to the southeast and continues in that direction through a narrow rock channel, about 100 feet deep, almost until it leaves the county near the village of Niskayuna. All drainage in the County is to the Hudson River, mostly via the Mohawk River.

At the southern edge of the flood plain of the Mohawk River, in the area of the facility site, the land surface rises rather abruptly within 1/2 mile from an altitude of about 230 feet to 350 feet above sea level. The higher level is a sand plain, in a youthful stage of dissection, which extends from Schenectady south-eastward toward Albany. Most of the residences in the County are built on this sand plain.

Figure 2.1 indicates the waterways and potable water sources in the Schenectady area.

Stratiography Summary

Rocks underlying Schenectady County were deposited in two widely separated eras; in early Paleozoic time and in late Cenozoic time. The Paleozoic rocks consist mostly of alternate layers of shale and sandstone deposited in shallow Ordovician seas as clay, silt, and sand. These sediments were buried by younger sediments, consolidated, raised above sea level, and subjected to erosion and weathering (after removal of younger sediments) during succeeding geologic time. The rocks in the eastern part of the County are folded and faulted, having been affected by crustal deformation originating near what is now New England.

The Paleozoic rocks are mantled almost everywhere by unconsolidated glacial drift deposited during Pleistocene time. During this period a continental ice sheet that originated in Labrador repeatedly advanced and retreated across the entire State. In some areas the glacier eroded the rocks deeply and in other areas it laid down thick deposits of unconsolidated material. It is believed that during the final stage of ice advance, called the Wisconsin stage, the glacier was thick enough to submerge completely the highest peaks in the Adirondack and Catskill areas. The Wisconsin ice advance, within Schenectady County, seems to have removed or reworked all or almost all the material that had been deposited during previous advances of the ice sheet. Wisconsin deposits in Schenectady consist mainly of glacial till containing a high percentage of clay, and of fluvioglacial deposits of gravel, sand, and clay. In addition, smaller deposits of clay, silt, and sand have been deposited on the flood plains of the larger streams of the County during Recent time.

Structural Geology

The structure of most of the consolidated rocks in Schenectady County is relatively simple. Almost the entire County is underlain by the Schenectady formation, a series of alternating beds of shale, sandstone, and grit about 2,000 feet thick which dip gently west and southwest. In most places the dip ranges from 1° to 2°, but in places it is as much as 5°. Although the Schenectady formation has never been subjected to stresses sufficient to produce folding, its continuity near the surface is broken by sets of intersecting nearly vertical joints.

Summary of Ground-Water Conditions

An average of more than 25 million gallons of ground water is pumped daily in Schenectady County. Ground water is the source of every municipal supply and water district in the County, with the small exception of the village of Delanson. In addition, several thousand wells have been drilled, driven, or dug to supply ground water to surburban and rural homes and to farms. Municipal supplies serve approximately 100,000 people, or about 80 percent of the area population, and several large industries including the General Electric Company, and the Knolls Atomic Power Lab. The principal pumpage is from an unconsolidated gravel deposit underlying the Mohawk River between the city of Schenectady and the village of Scotia. This deposit is relatively small in size but has produced large volumes of water continuously for more than half a century with no sign of depletion, undoubtedly because of recharge to the gravel from the Mohawk River.

Except for ground water derived from river recharge, essentially all potable ground water in the County originates from precipitation that falls on the surface of the County and its immediate vicinity. At any given spot the direction of ground water movement ordinarily is toward the nearest stream channel. The movement is usually under water-table conditions, and although artesian horizons are found locally, flowing wells are scarce.

Underlying more than 90% of the County, the Schenectady formation is its most widespread consolidated-rock aquifer, consisting of an alternating series of shale and sandstone beds as much as 2,000 feet thick. This formation and the other bedrock formations of the County are essentially impervious to the flow of ground water except insofar as they contain joint openings and bedding planes. Such openings are difficult to anticipate and generally tend to pinch out with depth. Yields from the rock wells show a considerable range and depend in large part on the thickness and nature of the overburden. In general, the yield is greatest (up to 150 gallons per minute) where the overburden consists of gravel or sand, and least (as low as 1 gallon per minute or less) where the overburden consists of clay or till. In most places, however, the consolidated rock will yield to drilled wells, ranging from about 50 feet to about 250 feet deep, enough water of satisfactory quality for domestic or farm needs. The mineral content of water from rock wells ranges over wide limits, both in hardness and in dissolved solids. The hardness may range from very low to very high but the dissolved solids are rarely low. The water from some wells is so highly mineralized as to be undesirable for most uses. Hydrogen sulfide gas in small amounts is not uncommon; traces of natural gas are occasionally found; carbonated mineral water of the Saratoga Springs type was found in one well.

Unconsolidated deposits of glacial origin, consisting of till, clay, sand, and gravel, mantle the consolidated racks almost everywhere. Glacial till is the most widespread of the unconsolidated deposits and, in Schenectady County, is dense and almost impervious, yielding only a few hundred gallons of water per day to large diameter dug wells. Deposits of till up to about 300 feet thick are found, but ordinarily the deposits are less than 50 feet thick. Clay of alluvial or lacustrine origin, which is much less common than till, will yield about the same quantity of water to large diameter dug wells.

By far the largest quantity of water is pumped from deposits of sand and gravel of relatively limited size. Most of these deposits occur along the principal stream channels. A deposit of sand occurs over a wide area in the section south of the city of Schenectady and in scattered places elsewhere in the County. Hundreds of shallow wells have been driven into the sand, usually yielding ample water for all domestic needs. The most productive aquifers in the County are part of a series of more or less interconnected deposits of sand and gravel that underlie the Mohawk River flood plain from the city of Schenectady upstream approximately 8 miles to Hoffmans. This series is the source of all the ground water pumped for municipal use in the County. The individual wells yield as much as 3,000 gallons per minute with relatively small drawdowns.

The water from the unconsolidated deposits is generally acceptable for industrial or municipal use, usually without treatment. So far, no public or industrial ground-water supply is treated in any way, except for stand-by chlorination. Small portions of water, however, are treated for particular industrial uses. Dissolved solids rarely exceed 500 ppm and hardness usually is less than 300 ppm. Iron or manganese occasionally is found in high-enough concentration to be troublesome.

Test borings were taken at the site about 100 feet from the southeast bank of the Mohawk River. Three holes were drilled; two to a depth of 25 feet, and one to a depth of 70 feet. The character of natural soil from 15 to 70 feet below the surface is classified as a fine, relatively uniform silt or silty sand with considerable evidence that much of the material is organic. The particle sizes range from 0.4 to less than 0.001 mm in diameter with the "50% finer than" point at 0.05 mm.

Artificial fill consisting of cinders, sand and brick in varying degrees of compactness was experienced to a depth of 15 feet below the surface. The apparent ground water level was reached at a depth of 12 feet which compares closely to the elevation of the Mohawk River.

Because of the character of this unconsolidated material, the Critical Facility building was supported by a reinforced concrete foundation resting on 104 treated wooden piles driven to a depth of 50 feet. Each pile is rated for a 20 ton bearing pressure.

Seismology

N.H. Heck's "Earthquake History of the United States", which reports on all recorded disturbances to 1927, indicates there have been two tremors in the immediate Schenectady area. These occured on January 24, 1907 and February 2, 1916. The former had an intensity of 5; and the latter, 4 to 5 on the Rossi-Forel scale of intensity. A quake with this intensity is described as a moderate shock, generally felt by everyone, and with some distrubance of furniture and ringing of bells. No damage results to a structurally sound building at this intensity level.

2.3 Meteorology and Climatology

In addition to the meteorological data taken during 1956-57 at the facility very complete records covering many years were available from the U.S. Weather Bureau in Albany. The Meteorology station at the Albany Airport is approximately 7 miles to the southeast and on a relatively level plain with an elevation approximately 120 feet above the proposed site. General land contours toward the southeast rather abruptly rise from an elevation of 230 feet at the site on the bank of the Mohawk river to the elevation of the Albany Airport within 1/2 mile from the site. The differences in the data taken at the Facility and the Albany Airport are no doubt influenced by the difference in location and the relatively poor statistics of facility data collected during a period of just 18 months.

Climatological Summary

The climate at Schenectady is primarily continental in character but is subjected to some modification from the maritime climate which prevails in the extreme southeastern portion of New York State. The moderating effect on temperatures is more pronounced during the warmer months than in the cold winter season when outbursts of cold air sweep down from Canada with greater vigor than at other times of the year. In the warmer portion of the year temperatures rise rapidly during the daytime to moderate levels. On the average, there are only 9 days per year with maximum temperatures of 90 degrees or above at Schenectady. The highest temperature of record is 104 degrees. As a rule, temperatures fall rapidly after sunset so that the nights are relatively cool and comfortable.

Winters are usually cold but no commonly severe. Daytime maximum temperatures in the months of December, January and February average around 37 or 38 degrees; the minimum during the night is about 20 degrees. On the average, there is an expectancy of 9 days during the year with sub-zero temperatures and the minimum temperature of record is 26 degrees below zero. Snowfall averages about 50 inches annually and the number of days in which one inch or more of snow covers the ground is approximately 50.

The precipitation at Schenectady is derived from moisture-laden air that is transported from the Gulf of Mexico and the Atlantic Ocean. Instrumental in the importation of this air are cyclonic systems which progress from the interior of the country northeastward over the St. Lawrence Valley, and also similar systems which move northward along the Atlantic Coast. It is only occasionally that the centers of these storms pass directly over Schenectady. Nevertheless, the area enjoys sufficient precipitation in most years to adequately serve the requirements of water supplies, agriculture and power production. Only occasionally do periods of drought conditions become a threat. The months of heaviest rainfall are from May through October when the average monthly totals range between three and four inches per month. The greatest fall to occur in any individual month is 13.48 inches while the least amount is 0.08 of an inch. Thundershowers are infrequent during the winter although they have been recorded for each month in the year. The mean number for the period of record is 22 annually. A considerable portion of the rainfall in the warmer months is supplied by storms of this type, but they are not usually attended by hail of any consequence.

On the whole, wind velocities are moderate. The prevailing wind direction from May through November is from the south, from the north in January, and from the west in the remaining months of the year.

Generally speaking, November, December and January are cloudy months but the remainder of the year is comparatively sunny with abundant sunshine to be expected in June, July and August. In fact, the average of cloudy days for the three summer months is only 7 or 8. Usually there are only a few days in the year when the relative humidity of the air causes personal discomfort to a great degree.

The extremes of atmospheric pressure over the 75 year period of record range from 28.46 to 31.10 inches of mercury.

With only those differences which are the result of differing latitudes, and topographical effects, the climate of Schenectady is representative of the humid area of the Northeastern United States.

Surface Wind Directions and Velocities

Hourly wind observations for an 8 year period (1938 - 1946) taken at the Albany Airport, on a plain about 7 miles southeast of the proposed site, are presented in Fig. 2.5 in terms of the average annual percentage frequency of surface wind direction and associated velocity. Figure 14 presents similar data taken at the Facility for the period September 1956 thru December 1957. It will be noted in Fig. 14 that 8.8% of the time prevailing winds occur from the south with an average velocity of 7.5 miles per hour (3.4 meters per sec.). On the other hand, winds from the northwest quadrant occur a total of 28.9% of the time with an average velocity of 8.9 miles per hour (4.0 meters per second). For purposes of this report, therefore, prevailing winds can be considered as originating in the northwestern quadrant and affecting the populated Schenectady area about 29% of the time. The relationship between wind direction and velocity and potential hazards to the surrounding population will be discussed in Supplements to this Application.

Flood History

Records kept by Alco Products since 1914 are summarized in Table . . This indicates a general flooding of the plant on several occasions with some flooding in buildings. No structural damage of significance has been experienced. From the last recorded high water in February 1939 to January 1956 there have been no floods exceeding an elevation of 227 feet. The floor level of the facility is at 230 feet, so no serious threat is anticipated in this respect. Precautions, however, will be taken to minimize or prevent damage which could result in the uncontrolled release of activity in a severe flood. For example, valves will be provided on sewer drains and the waste storage tank will be filled or securely anchored to prevent it from breaking loose during a flood although it is expected no significant amounts of activity will be stored in it.

Table
Maxima Recorded High Water at Alco Products, Plant #1

Date	Elevation (ft.)
March 28, 1914	232.0
April 2, 1916	229
February 20, 1918	227.3
February 12, 1925	227.0
March 15, 1929	227.1
March 19, 1936	228.0
February 21, 1939	227.5

From 1939 there have been no water levels exceeding an elevation of 227 feet.

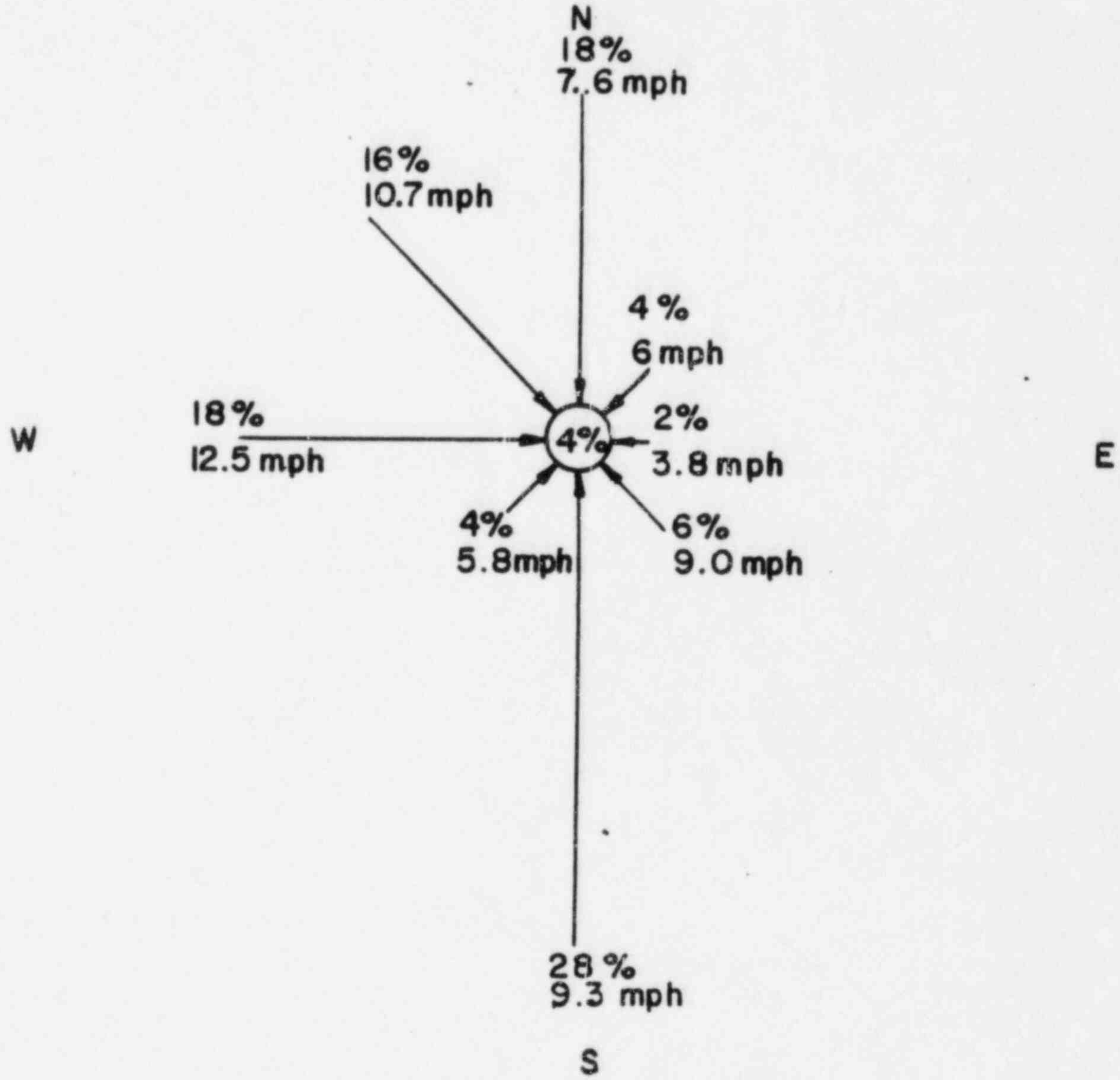


Figure 2.4- Average Annual Frequency of Surface Wind Direction (Albany Airport)

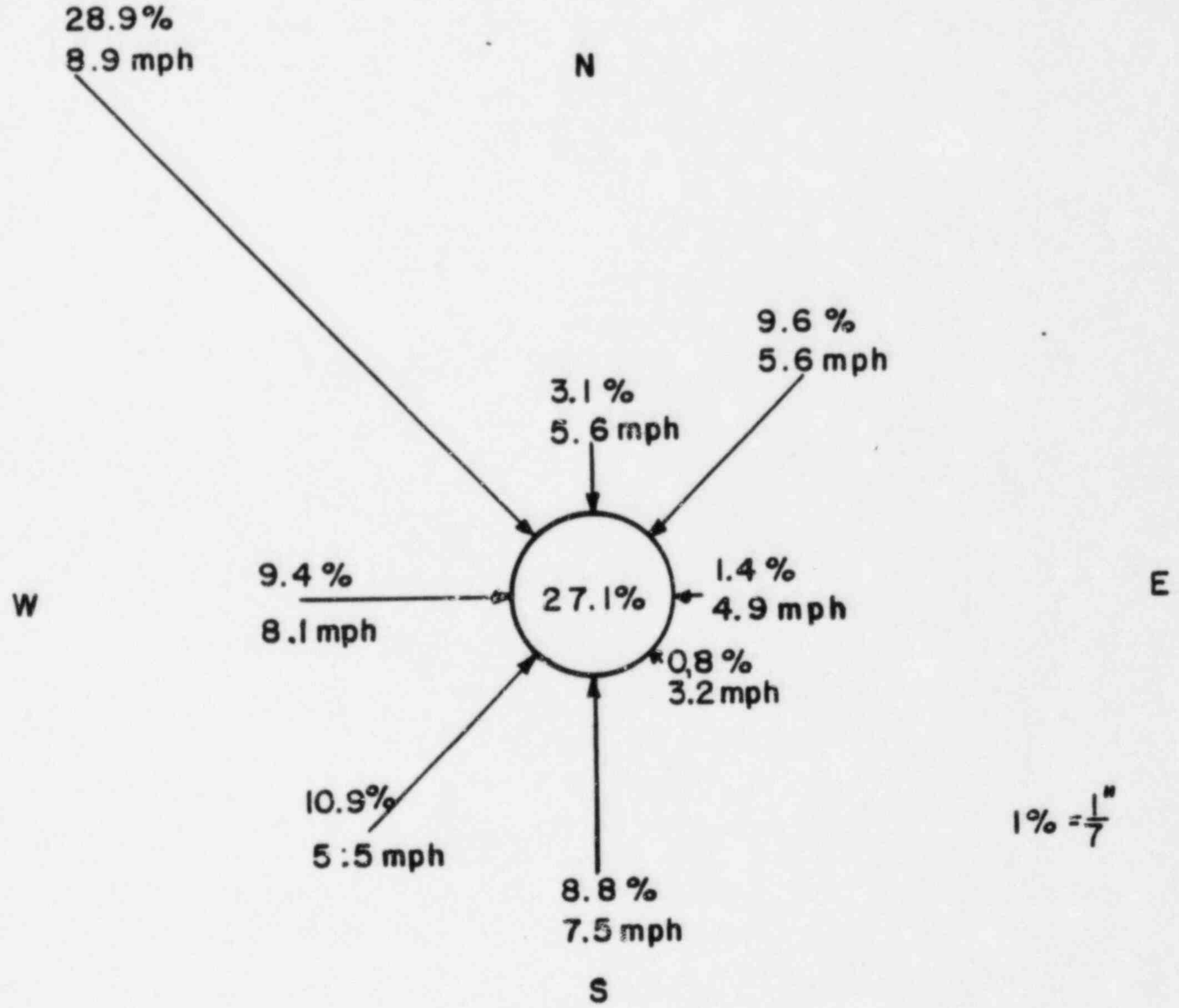


Figure 2.5 - Average Annual Frequency of Surface Wind Direction (Criticality Facility)

2.4 Evaluation of the Facility Site

It is believed the facility site as considered in this chapter is a satisfactory one for the construction and operation of a zero power critical assembly or critical experiment.

The static and southwestern quadrant wind conditions 38% of the time and the sparsely populated area beyond 1/2 mile to the north of the proposed site are highly favorable. On the other hand, winds from the northwest quadrant 29% of the time could potentially affect the heavily populated Schenectady area which makes the site less favorable.

The location of the facility on the bank of the Mohawk River three river miles downstream of the Schenectady public water supply is of no apparent hazard to this water source. The relation of the site to other potable water supplies downstream and downwind introduces no real hazard.

From all other considerations of the meteorology, geology, hydrology and seismology, the facility site appears to be as satisfactory a site as could be selected.

General Facility Description

The RPI Critical Facility is situated on the south bank of the Mohawk River, adjacent to the property of General Electric in the city of Schenectady, New York. The orientation of the Critical Facility and its relationship to the immediate vicinity is indicated in Figure 3.1.

The exclusion areas are also shown in Figure 3.1. These areas may be considered as three zones, each enclosed by a chain linked fence with controlled access gates. The inner zone is a fenced area 100 feet by 100 feet enclosing the Critical Facility with a minimum distance of about 40 feet to the reactor. The outer zone consists of the perimeter of the General Electric property. The middle zone is enclosed by a fence and encompasses the access road to the Facility from Maxon Road and the Facility parking lot. The minimum distance from the reactor to this fence is 80 feet. The middle and outer zones are open to the river on the northwest side.

Building Structure

Figure 3.2 outlines the Facility floor plan as an aid to understanding the following paragraphs.

Reactor Room

A reactor room 40 feet long, 30 feet wide, and 30 feet high is provided with walls of reinforced concrete. Concrete block outer walls on two sides contain the supporting facilities illustrated in Figure 3.2. The reactor room, three sides of which consist of one foot reinforced concrete, is separated from the office, control room, and men's room by three feet of reinforced concrete. The counting room is shielded by a total of five feet of concrete from the reactor room, two feet on the adjacent sides, and one foot on the roof.

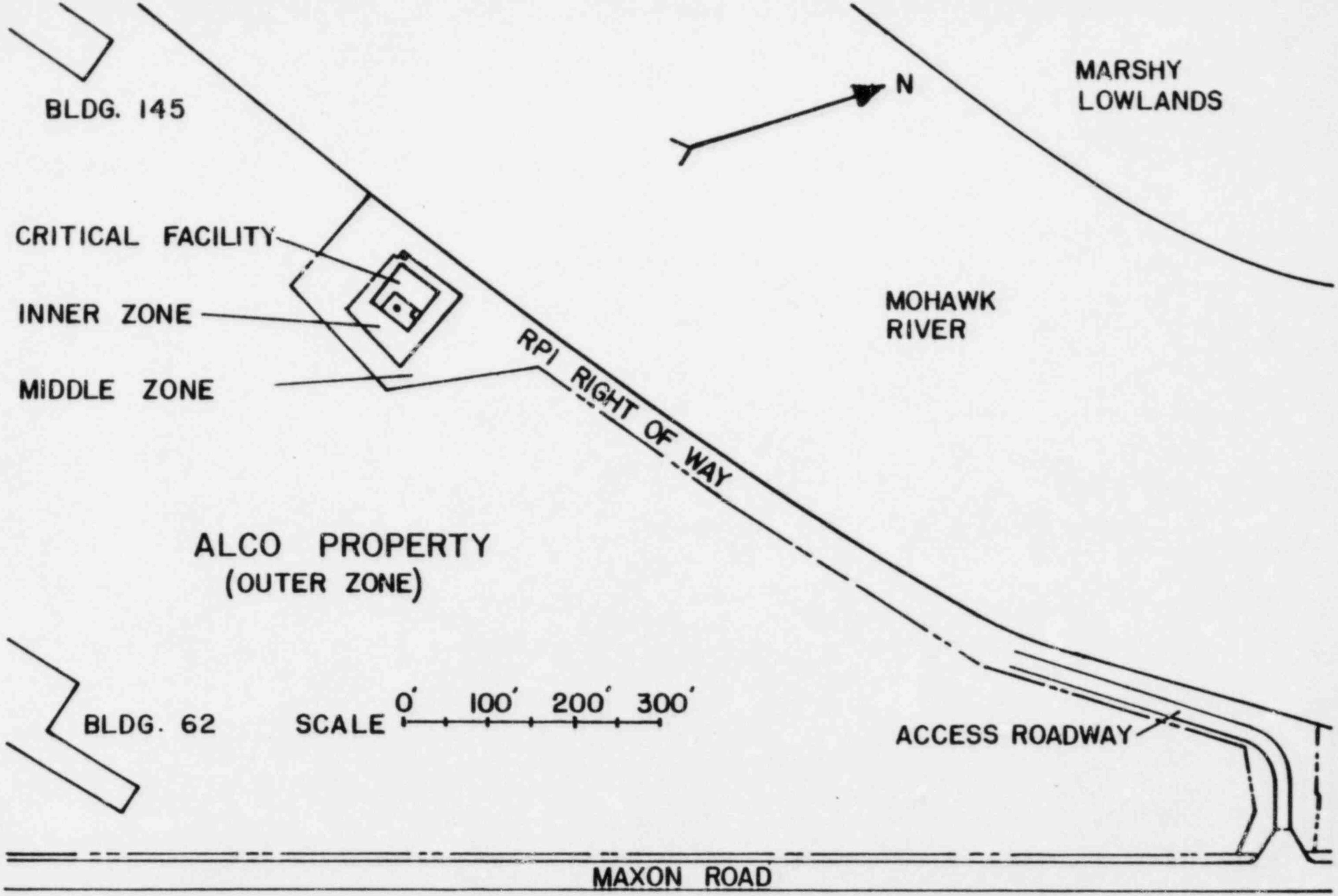


Figure 3.1

SITE AND VICINITY

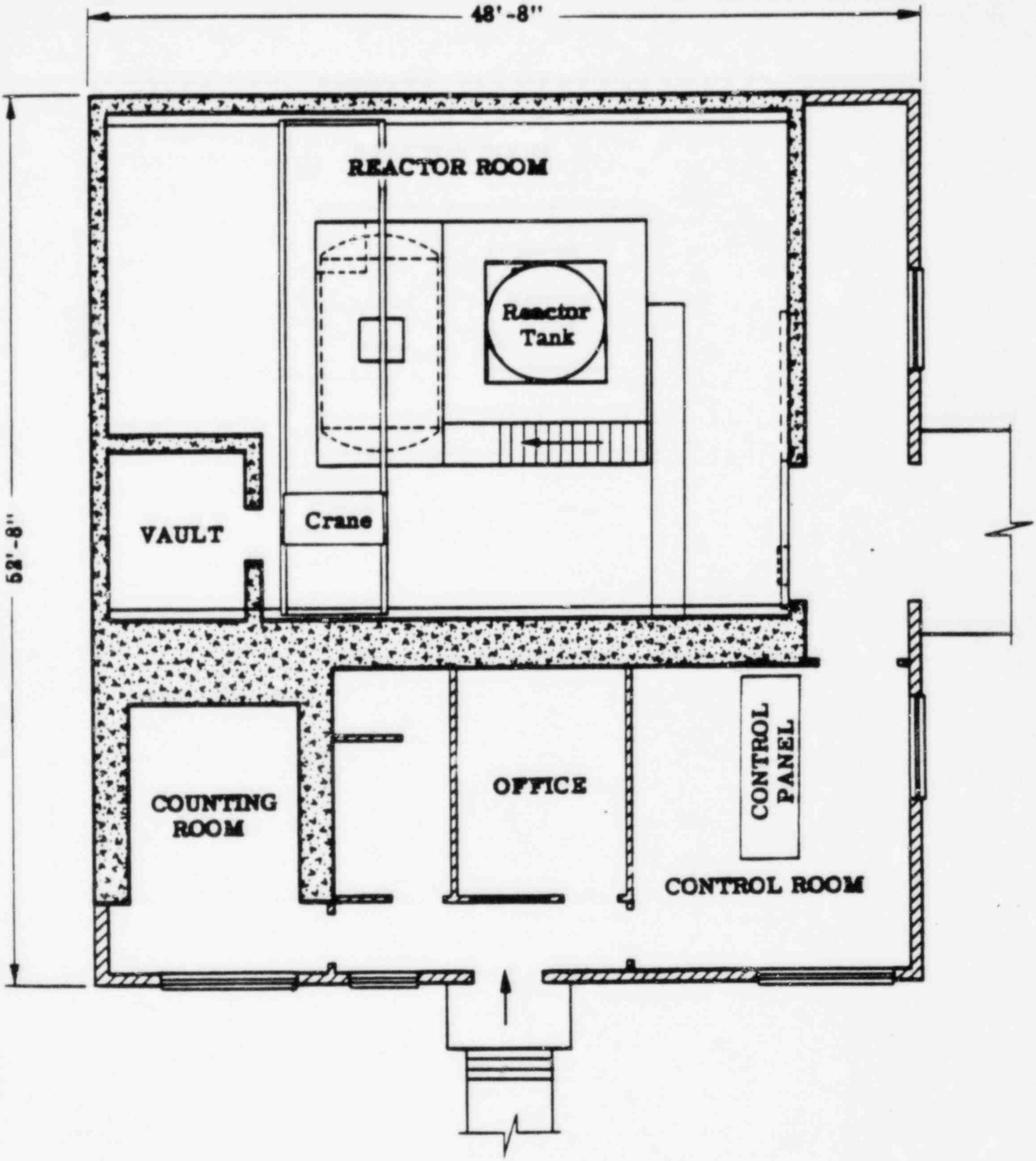


Figure 3.2 Plan View of Building

A stack extending to 50 feet above ground level is provided and contains a CWS filter for removing the small amount of fission products which might evolve from a maximum credible accident. Access is through an eight feet wide sealed sliding door for material and a three feet wide sealed personnel door in this material access door. The reactor may be viewed through a 2 feet by 3 feet 3/8 inch safety glass window located adjacent to these doors.

Storage Vault

In one corner of the reactor room, an 8 feet by 10 feet storage vault is provided with walls, ceiling, and floor of 1 foot reinforced concrete. This special nuclear material vault is equipped with ultrasonic alarms as part of the material access area. A storage rack constructed of unistrut is mounted against one wall of the vault opposite the vault access door. Figure I-3 displays the storage arrangement. Stainless steel tubes five inches in diameter surrounded by 0.015 inches of cadmium metal are bolted to the unistrut frame in parallel rows. With these 81 cells filled to U^{235} capacity, one kilogram, conservation calculations indicate an infinite multiplication factor of about 0.90 under flooded conditions.

Control Room

Figure 3.4 is a photograph of the control and counting rooms. Major features of the control room are the sealed instrument cable trench, an enclosed sight glass indicating reactor tank water level, the control console, and the auxiliary electric panel.

Radiation Monitoring

An area monitoring system provides continuous indication of gamma radiation levels at various locations in the Facility. Flashing light alarms indicate high radiation levels. A variety of portable radiation detection equipment is also available.

Counting Room

The additional shielding constructed for the counting room has already been described. This room contains the scintillation counting equipment, a Mettler balance, an oscilliscope, a multi-channel analyzer (MCA), and a terminal for the RPI main computer.

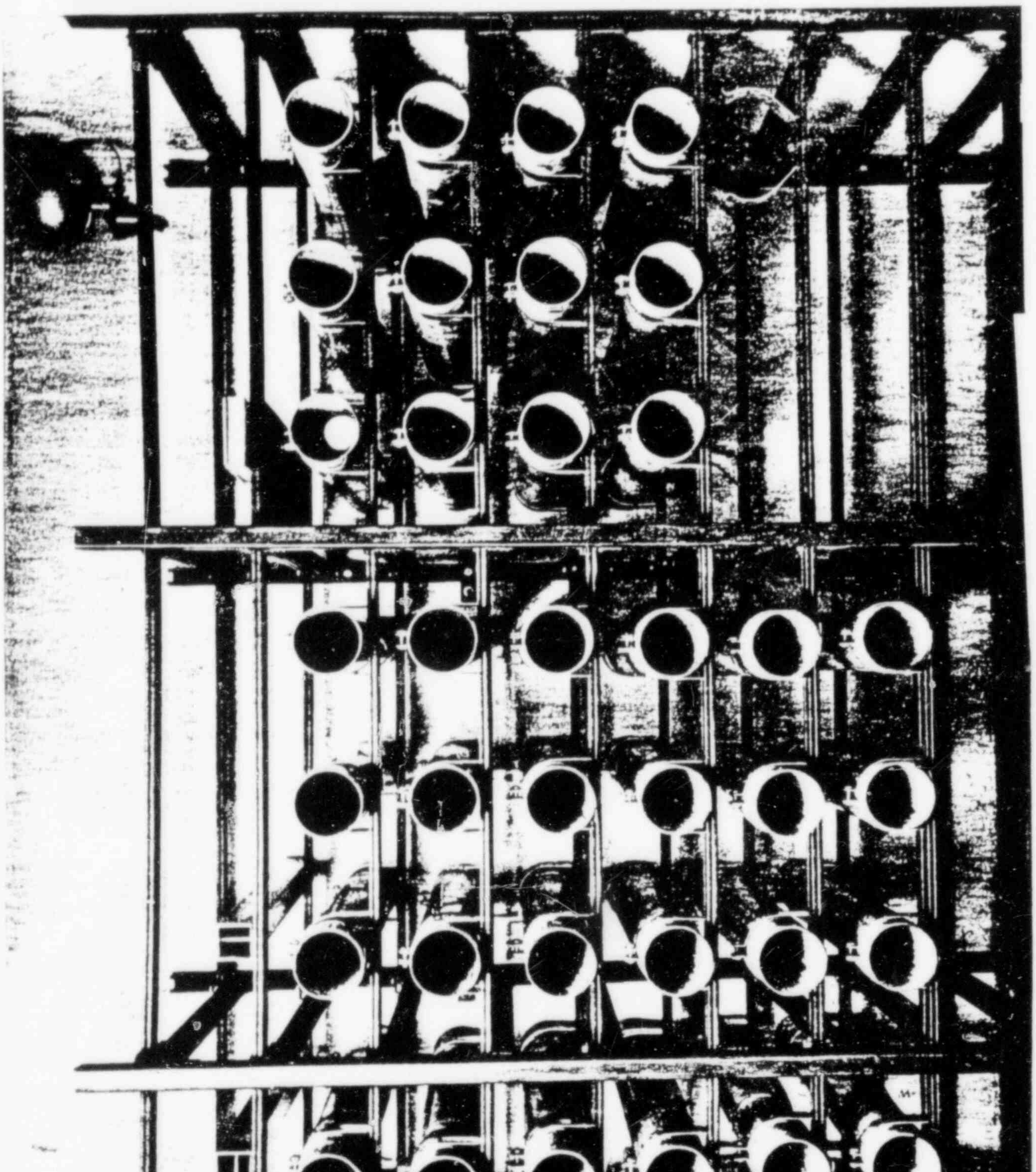
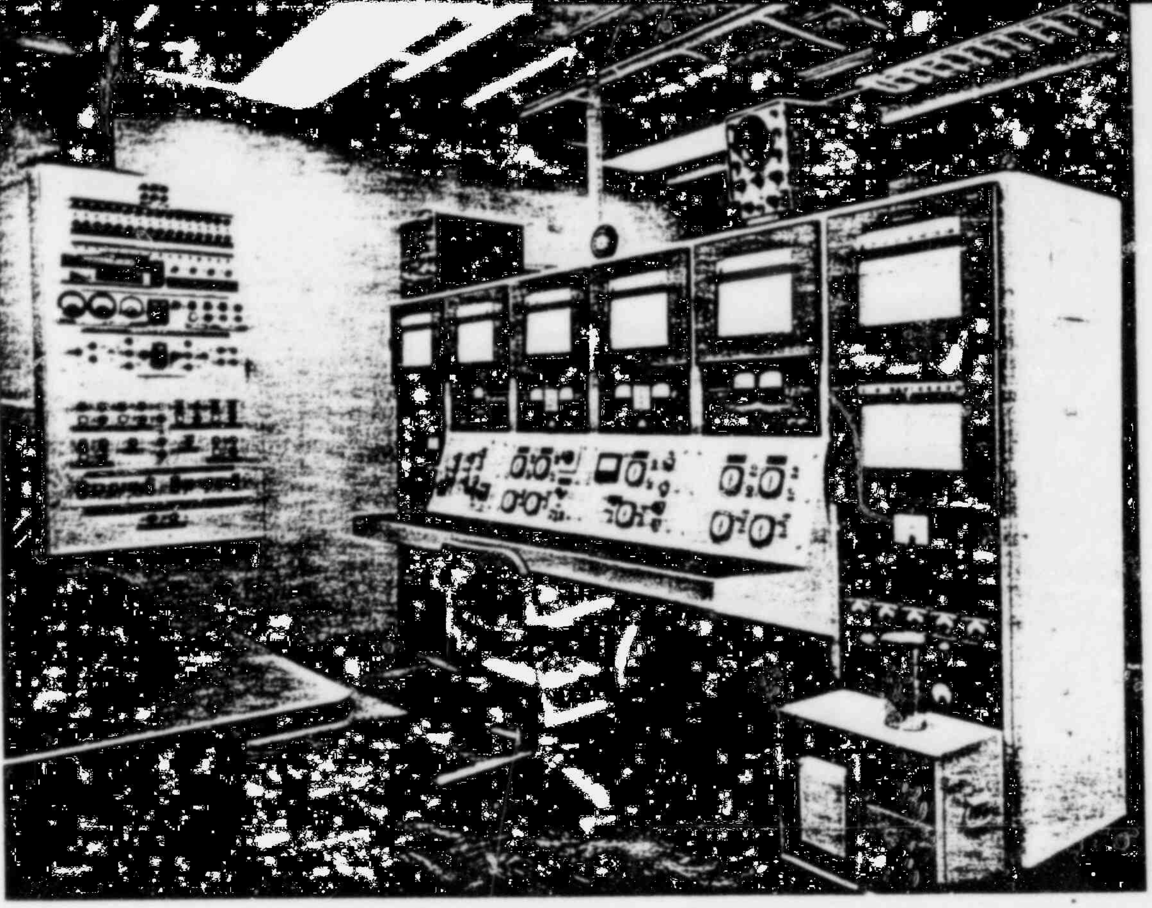


Figure 3.3 Fuel Storage Vault



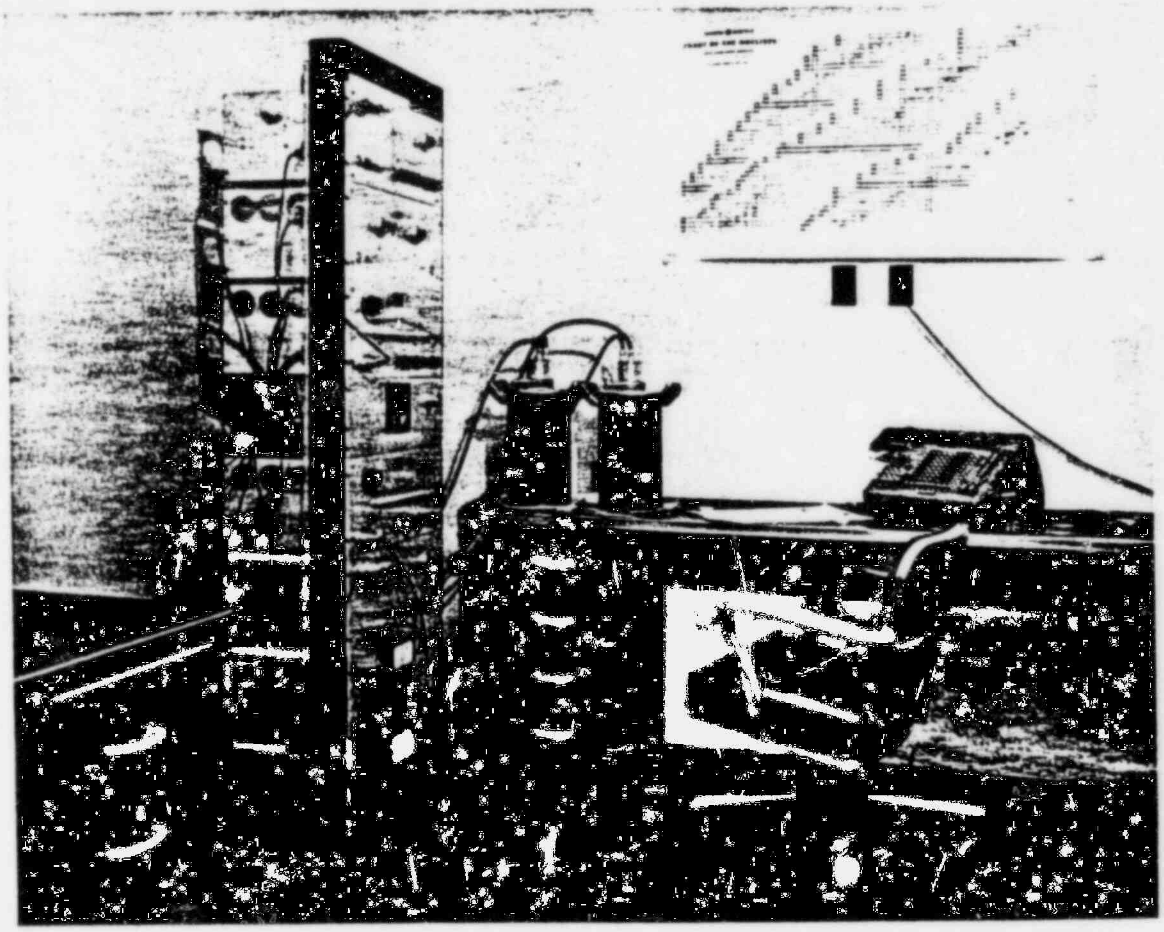


Figure 3.4 Control Room-Counting

4. Reactor

Reactor Tank

The reactor tank, storage tank, pumps, valves, and all system piping are of stainless steel. This allows the use of untreated, city-supplied water without inducing corrosion or other water damage. The reactor tank structure, Figure 4.1, is mounted at floor level and is supported by I beams bridging the reactor room pit. A welded steel catwalk structure and staircase provide access to the tank rim seven feet above floor level. With a seven feet inner diameter, the tank capacity is about 2000 gallons.

The storage tank, mounted horizontally and strapped to the depressed section of the reactor room pit, is seven feet in diameter and ten feet long. A covered manhole and several vent pipes are provided on the uppermost surface of the tank in addition to the feed and dump line ports. The tank capacity is about 3500 gallons.

Control Rod Drives

The overhead control rod drives, seven in number, used at this Facility are mounted on the tank as shown in Figure 4.1 . One such drive is shown in further detail in Figure 4.2 . The drives are supported by rigid cantilevers with three degrees of freedom to allow positioning of the rods at any point in the tank. Structurally, the drives consist of 1/20 horsepower motor, gear box, magnetic clutch, drive shaft, pinior gear, and control rod rack. Control rod position is determined by a pair of geared anti-backlash

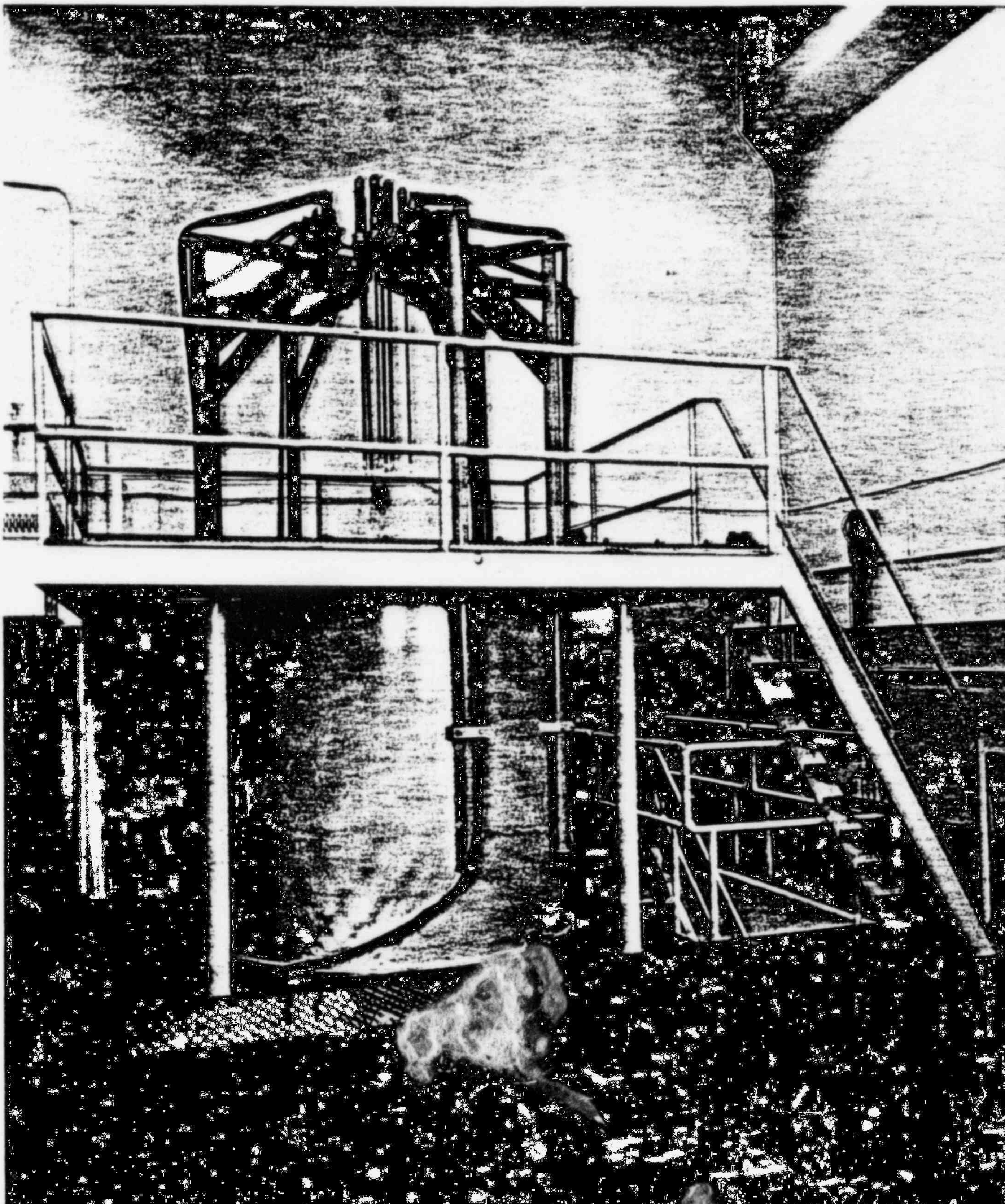
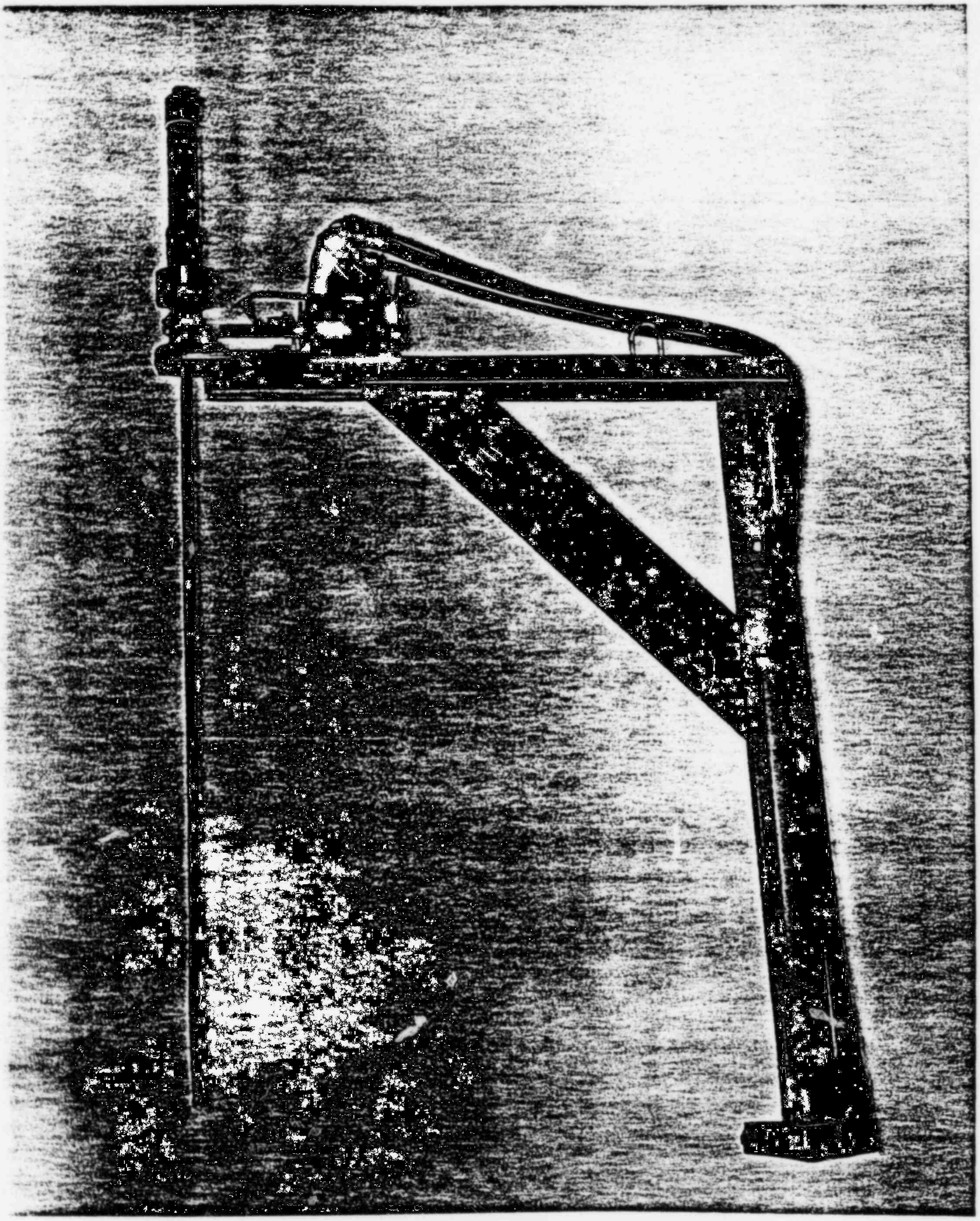


Figure 4.1 Reactor Tank Structure



Control Rod Drive
Figure 4.2

synchromotors. Electrically the control rods operate on demand from the control room with power supplied to the magnetic clutches from the safety amplifiers. A minimum holding current is adjusted for each drive individually to minimize magnet decay time and therefore rod drop-time. This current is interrupted on receipt of any scram signal or on power failure.

Source and Source Drive

A five curie neutron source of encapsulated Pu - Be is used during reactor operation. Source emission rate is approximately 10^7 neutrons/second. The source is inserted into and withdrawn from the reactor via an attached 1/4 inch rod by means of a friction drive motor. In the withdrawn position the source is enclosed in a 6 inch by 8 inch paraffin block for shielding purposes.

Reactor Core

The core and support structure were designed for flexible critical experiments using variable arrays of assemblies made up of flat fuel plates. The fuel is in the form of UO_2, 93% enriched in U^{235}, fabricated in a stainless steel cermet and clad in stainless steel. The nominal active core dimensions are 22 inches in height and an equivalent diameter of about 16.5 inches. The grid structure can accommodate up to 37 stationary fuel assemblies and 7 control rod assemblies with fuel followers. The support structure consists of a three-tiered table of grid plates mounted on four posts set in the floor of the reactor tank as shown in Figure 4.3.

Fuel Assemblies

The flat fuel plates are made up into box type assemblies with cell dimensions of 2.9375 inches square by 22 inches high. They may contain a maximum of 18 plates for stationary assemblies or 16 plates for control rod fuel followers. The side plates are 0.027 inches stainless steel, turned in at the edges, with grooved polystyrene inserts

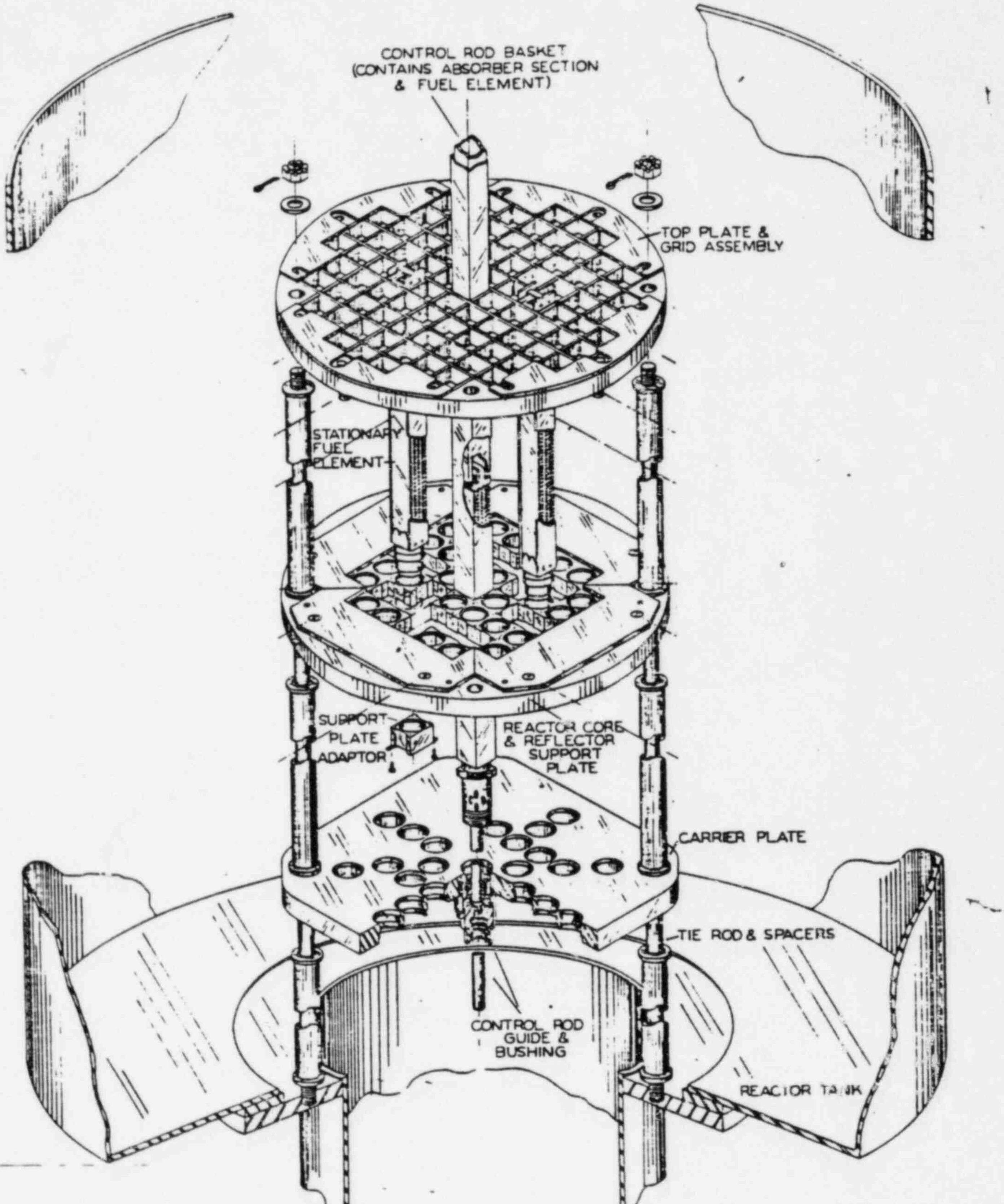


FIG. 4.4 - CORE SUPPORT STRUCTURE

to maintain the plate center to center spacing at 0.163 inches. For reduced density loadings, which is frequently the case, some plates may be omitted or may be dummy plates without fuel.

Control Rods

Seven control rods are provided, and removable inserts in the grid plates permit many choices of lattice positions. Each rod consists of a 2.75 inch square tube which passes through the core and rests on a hydraulic buffer on the bottom carrier plate of the support structure. A fuel follower is inserted in the bottom section of the tube, below the core, and a box type absorber, 2.619 inches square, is inserted above it, within the core.

Four types of absorbers are available. Seven rods are enriched boron in iron; five contain Eu_2O_3 in a stainless steel cermet; three contain an alloy of silver-cadmium-indium; and one is simply stainless steel. All are clad in stainless steel, have the same dimensions, and are approximately equivalent in reactivity effect, except for the one of stainless steel.

Control Instrumentation

The Facility control instrumentation is based upon a fail-safe philosophy and multiple channel scram triggers in this approach, the probability of a neutron level increase failing to cause scram is insignificantly small.

The neutron chambers are normally positioned about eight inches from the core boundary in water tight stainless steel tubes. Specifically, there are two BF_3 start-up detectors and four uncompensated ionization chambers.

Two of the ionization chambers feed Beckman micromicro-ammeters. They are linear instruments with scale changes provided to cover a range of 3×10^{-13} amps. to 3×10^{-7} amps. The remaining two ionization chambers feed log-N and period amplifiers. The range of these instruments is about six decades.

5. Reactor Coolant System

The maximum design power of 100 watts results in negligible heat up of the 2000 gallons of water in the reactor tanks.

The water temperature transients under accident conditions is analyzed in Section 14.

6. Engineered Safety Features

Engineered safety features shut down the critical chain reaction if limits are exceeded.

A SCRAM signal simultaneously drops control rods into core and opens a moderator dump valve.

The engineered safety features and set points are described in Section 15, Technical Specifications.

7. Instrumentation and Control

The core has the capability of seven control rods, at least three of these will be used for any critical core assembly. Each drive gear box contains a lead screw actuating upper and lower limit switches, normally set for 22" travel, and synchro transmitters for coarse (0-22") and fine (0.1", legible to 0.001") position indication. The drive switches and synchro receivers are mounted on the control room console. When there is a reactor scram, the rod drives' clutch magnet current is interrupted and all rods drop. Additionally, the moderator is dumped when it is not bypassed. The control rods and moderator dump are to operate within the limits of section 3.1 of the 'Technical Specifications.'

The nuclear instrumentation for control of the reactor consists of the following neutron flux detectors;

2 counters (1 minimum) - BF_3 or fission chambers

4 ion chambers, uncompensated - 2 minimum linear amplifiers
1 minimum log. amp. & period

Their minimum operating ranges and scram settings are as follows:

counters: 1 to 10^4 cps log ratemeter, no scram

lin. amp.: 3×10^{-13} to 3×10^{-7} amp., scram 90% each scale

log. amp.: 5×10^{-11} to 1.5×10^{-4} amp., scram 5 sec period

The neutron source yields about 10^7 neutrons/second, which is sufficient to maintain the logarithmic count ratemeters and linear amplifiers on scale at all times when the reactor is subcritical. The linear amplifiers and logarithmic amplifiers cover all power ranges above critical up to 135 watts.

In accordance with section 3.3 of the 'Technical Specifications' there is an area gamma monitoring system. Four scintillation detectors are used, one at each of the following locations; control room, reactor room near the fuel vault, reactor deck, and outside the reactor room window. Portable radiation monitors are also available. There is a "cutie-pie", a G-M tube, and a portable neutron survey meter for this purpose. Additionally, the counting room contains sodium-iodide photomultiplier tubes for checking activated foils and samples, wipe tests, water sample residues, etc.

Whenever the reactor is to be operated the particulate activity of the reactor room atmosphere is monitored. The air monitor counts the beta-gamma activity on a filter paper through which a continuous 5 cfm sample of air is drawn from the stack duct. It provides audible and visual alarms if the count rate goes above 2000 cpm.

The safety system channels that operate during reactor operation are specified in section 3.1 of the Technical Specifications. This indicates the channel's function and range of operation.

8. <u>Electric Power Systems</u>

Off-site electric power is provided by Niagara Mohawk.

There is no on-site emergency power supply.

9. Auxiliary Systems

The reactor is not dependent upon auxiliary systems for safe shutdown.

The following auxiliary systems are involved in reactor control and are controlled from an auxiliary control panel.

- Fast and slow fill and drain controls and lights for moderator control.
- Key locked switch for optional automatic dumping of the moderator on any scram signal. This feature is always used when approaching criticality with the moderator controls.
- Moderator dump and reset buttons.
- Power circuit breakers.
- 400 cycle MG set voltage, current, and frequency meters and push button control.
- Source drive controls and limit switch lights
- Sump pump, agitator (used for maintaining uniform tank temperature during temperature coefficient runs), two 15KVA immersion heater, air compressor and unit heater fan controls.

10. Experimental Systems

The standard experimental programs are described in "A Manual of Experiments for the Rensselaer Polytechnic Institute Critical Facility, 1975".

All new experiments or classes of experiments that raise an unreviewed safety question shall be reviewed and approved by the Nuclear Safety Review Board in accordance with Section 6.3 of the Technical Specifications.

13. Conduct of Operations

Operations will be performed in compliance with the Operating Procedures.

Emergencies will be coped with in compliance with the Emergency Procedures.

14. Accident Analysis

14.1 GENERAL SUMMARY

This analysis was performed to evaluate the response of an uncontrolled rod withdrawal event which inserts a positive ramp reactivity worth 0.32% ($0.41) into the RPI Critical Facility Core 2 within 0.5 seconds during which the facility operates at a steady-state thermal power of 100 watts. The 0.5 second insertion time is a conservative assumption comparing to the technical specification of maximum rate of reactivity insertion, $10^{-5}sec^{-1}$.

The reactor is an open tank type, using 93% enriched uranium fuel plates. The reactor core since 1963 contained 6.01 Kg U-235, until it was reconfigurated to a critical mass less than 5 Kg, NCR's proposed "Formula Quantity" in 1980. The core reconfiguration causes the necessity of re-evaluating the safety on postulated events.

The analysis was based on the RETRAN-02 computer code. Fourteen control volumes (11 for reactor core and 3 for water tank) and eighteen junctions were used in the calculation. Two non-conducting heat exchangers were used to model the heat transfer on tank surface.

The analysis assumed that the operator can lower the water level in the reactor tank by opening the dump valve to obtain the reactivity decrease when water level is below the top of reactor core. It takes 12 seconds to dump the water level to the top of the core level according to the valve capacity and tests. The technical specification value of this dumping time is 60 seconds.

14.2 FACILITY DESCRIPTION

The RPI Critical Facility is described in References 1 and 2. The parameters relevant to this analysis are given here. The reactor core consists of 25 square fuel assemblies supported by grid plates in the center of a water tank as shown in Figure 14.1. With seven feet inner diameter and seven feet height, the tank capacity is about 2000 gallons.

The reconfigurated core is shown in Figure 14.2 with the corresponding plate locations. The active core is roughly 15 inches square by 22 inches high. The configuration of the core is non-uniform, but in good symmetry. The core contains 4.952 Kg of U-235, with an excess reactivity of 0.0032 at $68^{o}F$.

The fuel is in the form of UO_2, 93% enriched in uranium, dispersed in stainless steel cement and clad in stainless steel. Fuel plate dimensions are listed in Table 14.1. Figure 3. shows six plates inserted in an assembly.

Table 14.2 lists the nuclear and physical characteristics of RPI Core 2. Kinetics parameters of the core are shown in Table 14.3 which shows that all parameters of Core 2 satisfy the limitation of technical specification. Table 14.4 lists the measured data of feedback coefficient. The fuel temperature coefficient of reactivity $(\alpha_t)_f$ given in this table is obtained from results of experiments performed at the Kyoto University Reactor[3] which is the same type reactor as RPI Critical Facility.

A "scram" signal can initiate a rapid shutdown by simultaneously inserting control rods and dumping moderator from the reactor tank through a five inch diameter pipe to moderator storage tank. The scram can be initiated either by safety channel actuated trips or manually by operator. Measured rate of dump is 12 seconds from normal level to the top of the core. Technical specification requires maximum time of 60 seconds from initiation of dump until negative reactivity is inserted. The measured data of reactivity decrease due to the moderator dump is listed in Table 14.5.

14.3 RETRAN MODEL

The analysis was based on the RETRAN-02 computer code, a new version released in May 1981. RETRAN-02 removes some limitations and extends the analysis capability of RETRAN-01. A detailed description of RETRAN-02 code will be found in Reference 4. A brief account of the relevant parts of the code is given here in order to describe the model used in this analysis.

The reactor kinetics may be computed by use of the point kinetics model or the space-time kinetics model. The space-time kinetics model is a one-dimensional two-energy group space-time kinetics model. If this model is selected, a cross-section data file must be supplied. Up to six delayed neutron groups can be used. The feedback reactivity in RETRAN-02 can be expressed in terms of tabulated water density and fuel temperature reactivity functions, as well as fuel temperature and moderator temperature reactivity coefficients. Control reactivity is input in tabular form and can be initiated by the trip logic to simulate the transient rod reactivity and scram reactivity.

Heat transfer and temperature distribution in the fuel elements are based on standard one-dimensional conduction models. Both plate and rod geometries can be modelled. Temperature dependent conductivity and heat capacity for the fuel and clad are supplied in tabular form. A number of surface heat transfer models are available to cover different flow regimes including critical heat flux conditions.

The fluid system is broken down into a number of control volumes interconnected by junctions. Conservation equations of mass, momentum and energy must be satisfied. The effects of gravity, kinetic and potential energy, pressure, wall friction, junction losses and heat transfer are included in the equations.

14.4 CORE MODEL

Because of non-uniform fuel plate locations, the Core 2 was divided into five regions: three fuel regions (regions 2, 3 and 4) and two by-pass regions (regions 1 and 5) as shown in Figure 14.4. The fuel plate arrangement was assumed uniform in each region and the related thermal-hydraulic properties of each region are given in Table 6. Where the equivalent hydraulic diameter was taken as

$$D_H = \frac{4 \times \text{Plate Width} \times \text{Plate Pitch Distance}}{2 \times (\text{Plate Width} + \text{Plate Pitch Distance})} \qquad (4.1)$$

Three axial volumes were designed for each fuel region and a single volume for each by-pass region as shown in Figure 14.5. Totally, the reactor core was represented by eleven control volumes and each volume associated with one heat conductor element. The power fraction of heat conductors was provided according to radial and axial neutron flux distributions. The radial flux distribution was measured and the axial flux distribution was assumed a chopped sine shape.

14.5 FLUID VOLUMES AND JUNCTIONS

The schematic of the facility RETRAN model is shown in Figure 6. Besides the core volumes described in prior section, three control volumes were presented for the water in lower (volume 6), upper (volume 7) and side (volume 8) parts of the tank. Eighteen junctions were used to interconnect the 14 fluid volumes and describe the natural circulation through the core and water tank. The thermal-hydraulic properties of control volumes and the mass flow rates of junctions in steady-state operation were specified in Section 14.8 as the input data of RETRAN code.

Two non-conducting heat exchangers were used to model the heat transfer on the tank surface by means of air convection and thermal radiation. Two heat sinks having constant temperature, 68^{0}F were connected to volumes 7 and 8 to remove heat according to the heat transfer coefficient and the temperature difference.

14.6 HEAT CONDUCTORS

A total of 13 heat conductors were used in the model. Eleven heat conductors represent the fuel plates, one per fluid volume in the core. These conductors have internal heat generation. Two non-conducting heat exchangers were used to account for heat dissipation from tank surface by air convection and radiation.

CORE HEAT CONDUCTORS

The reactor fuel was modelled with eleven heat conductors, one per volume. The heat conductors were allowed to interact only with the adjacent fluid volumes. The rectangular geometry, two region representation of the fuel plates was used with six nodes in the fuel and five nodes in the cladding. No power generation in the clad was assumed.

Power fraction of each core heat conductors was weighted with the volume of fuel, V, multiplied local neutron flux ,ϕ, as

$$PF_i = \frac{\sum_{\text{conductor } i} (V\phi)_{\text{fuel plate}}}{\sum_{\text{core}} (V\phi)_{\text{fuel plate}}} \qquad (6.1)$$

The radial flux variation was measured and shown in Table 6 of Reference 2. To the variation of flux along the axis, a chopped sine curve was assumed

$$\phi(Z) = \phi_o \ \mathrm{SIN} \ \frac{\pi(Z + 0.05H)}{1.1H} \qquad (6.2)$$

where ϕ_o is the peak value of neutron flux and H is the height of fuel plate. PF_i of each heat conductor was given in Table 7. PF_i in transient condition were assumed same with that in steady-state operations.

The heat transfer coefficient for single-phase liquid natural convection was given by the Collier correlation. Thermal properties of UO_2 and stainless steel were specified as a function of temperature and provided in tabular form according to Reference 5.

NON-CONDUCTING HEAT EXCHANGERS

Heat transfer on the tank surface by means of air convection and radiation was modelled as volume-associated heat exchangers which do not consider conduction. Two heat sinks having constant secondary temperature, 68^oF were connected to the reactor tank of volumes 7 and 8 to remove all heat generation in steady-state operations.

The heat transfer coefficient on the tank surface was therefore calculated automatically by RETRAN code as the heat removal rate divided by the product of initial mass flow rate and the temperature difference between the tank volume and sink volume.

14.7 REACTIVITY CALCULATION

The nuclear model of the analysis is based on the point kinetics equations, one prompt neutron group, six delayed neutron groups, and eleven delayed gamma emitters. The feedback reactivities was expressed in terms of tabulated moderator density and fuel temperature reactivity functions, and moderator temperature reactivity coefficient. Moderator density weighting factors, moderator temperature weighting factor and Doppler weighting factor were also provided for each fluid volume and heat conductors in the core to account for the variation of local reactivity effects. Control reactivities were input in tabular form and were initiated by the trip setting times of the red withdrawal and the moderator dump. The prompt and delayed energy are released in core heat conductors according to a power fraction given in Table 7.

The void coefficient and the moderator temperature coefficient of reactivity have been well measured in RPI Critical Facility reconfigurated core.(2) The fuel temperature coefficient of reactivity is obtained referring to the experimental results performed at the Kyoto University Reactor.(3) The worth of one dollar (β_{eff}) was taken to be 0.0078 since initial operation of RPI Critical Facility, and was applied to the reconfigurated core. Prompt neutron lifetime was determined to be 8.2×10^{-5} seconds for highly-enriched MTR reactor.(6)

VOID COEFFICIENT OF REACTIVITY

The average coefficient of reactivity of each fuel assembly listed in Table 4, were measued by the method described in Reference 1. These values yield a negative core average void coefficient of $-7\times10^{-6}\ \Delta P/cm^3$ weighted by the I_j of each assembly. I_j is defined as

$$I_j = S_j^+ . S_j \ / \ \Sigma_{j'} \ S_{j'}^+ . S_{j'} \tag{7.1}$$

and the values obtained from NODER program(2) are shown in Table 8.

The core average void coefficient, $-7\times10^{-6}\ \Delta P/cm^3$ was translated to a water density reactivity function and input to the program in tabular form. Water density weighting factors, WF_i were provided to the code for each control volume in the code as

$$WF_i = \frac{\sum_{j\varepsilon\ \mathrm{volume}_i} (V_m\ \alpha_v)_j}{\sum_{j\varepsilon\ \mathrm{core}} (V_m\ \alpha_v)_j} \tag{7.2}$$

where V_m is the volume of water and α_v is the void coefficient of reactivity in various locations. Table 9 gives the calculation results of WF_i and the negative value in volume 11 (water column in the central assembly) means a positive void coefficient in this volume. This positivity is not expected to have much effect on the overall reactivity change because of good heat transfer and small density change in this region.

MODERATOR TEMPERATURE COEFFICIENT OF REACTIVITY

The isothermal moderator temperature coefficient of reactivity, $(\alpha_T)_{m,iso}$ was measured and represented for RPI reconfigurated core as [2]

$$(\alpha_T)_{m,iso} = 7.82 \times 10^{-3} \left(1 - \frac{T^oF}{61}\right) \$/^oF \tag{7.3}$$

The moderator temperature coefficient of reactivity of each control volume in the core was obtained from

$$(\alpha_T)_{m,j} = (\alpha v)_j \left[\left(\frac{\alpha V_j}{\alpha T_j}\right) + C\ I_j \right] \tag{7.4}$$

due to the temperature change T_j in control volume j only. The constant C in equation (7.4) can be calculated from the relation of

$$(\alpha_T)_{m,iso} = \sum_j \left[(\alpha v)_j \left(\frac{\alpha V_j}{\alpha T_j}\right) + C\ I_j \right] \tag{7.5}$$

with known values of $(\alpha T)_{m,iso}$ and $(\alpha v)_j$, $\left(\frac{\alpha V_j}{\alpha T_j}\right)$, I_j in control volume j. The calculation results of $(\alpha T)_{m,j}$ are listed in Table 9.

FUEL TEMPERATURE COEFFICIENT OF REACTIVITY

Reactor noise measurements were carried out on the Kyoto University Reactor (KUR), a tank type reactor operating with MTR type 90% enriched uranium fuel, natural convection, for two different core configurations. Fuel temperature coefficients were determined to be $-1.5\times10^{-5}\ \Delta P/^oC$ and $-2.0\times10^{-5}\ \Delta P/^oC$ for two core configurations[3] in these noise measures. The less negative value was used in this analysis.

Doppler weighting factors, DF_i were calculated for each core heat conductor as shown in Table 9, to examine the local fuel temperature effects on the reactivity as

$$DF_i = \frac{\sum_{j\varepsilon \text{ conductor}} (IV_f)_j}{\sum_{j\varepsilon \text{ core}} (IV_f)_j} \qquad (7.6)$$

where V_f is the volume of metal in each heat conductor.

14.8 STEADY STATE INITIALIZATION

The RETRAN overall balance equations used to describe the fluid flow are developed in Section II, Computer Code Manual, Volume 1.[4] The balance equations, together with the initial conditions, represent an initial value problem which is solved according to the numerical technique described in the Manual. For steady-state fluid flow, initial conditions are required which result in zero time derivatives for each of the balance equations.

MASS BALANCE

The steady-state continuity condition,

$$\sum_{j\text{-in}} W_j - \sum_{j=\text{out}} W_j = 0 \qquad (7.1)$$

must be satisfied for any volume. If the inconsistency in the mass balance is significant, the problem is terminated since it would be a fruitless exercise to perform the steady-state initial value calculations for the momentum and energy balance equations using unacceptable initial values for the junction mass flow rates.

MOMENTUM BALANCE

The steady-state momentum is satisfied by computing volume pressures and acceptable junction mass flow rates. Because of no pumping force, the natural convection of the coolant around the fuel plates is the only mechanism for heat removal in the RPI Critical Facility core. Therefore, the buoyancy force due to density change is equal to the friction force in the flow channel of each volume during steady-state operation.

The single phase steady-state momentum balance over a coolant channel j in the core can be stated as

$$\Delta P_{core\ j} = \frac{g}{g_c} H \left(\bar{\rho}_{\ell,DC} - \bar{\rho}_{\ell,\ core\ j}\right) \tag{8.2}$$

$$= \left(\frac{f_w H}{D_H} + K_{in} + K_{out}\right) \frac{G^2}{2 g_c \bar{\rho}_{\ell,\ core\ j}}$$

where H is the height of the core, D_H is the hydraulic diameter of the coolant channel, G is the mass flux of circulation flow, f_w is the friction factor of coolant channel, K_{in} and K_{out} are entrance and exit pressure loss coefficients, $\bar{\rho}_{\ell,DC}$ and $\bar{\rho}_{\ell,core\ j}$ are average liquid densities in downcomer and coolant channel in core region j. $\bar{\rho}_{\ell,core\ j}$ can be expressed as

$$\bar{\rho}_{\ell,core\ j} = \frac{1}{H} \int_0^H \rho_\ell \left(T_j(z)\right) dz \tag{8.3}$$

and approximated by

$$\bar{\rho}_{\ell,core\ j} = \frac{1}{H} \int_0^H \left(\rho_o + \beta\, \rho_o \left(T_j(z) - T_o\right)\right) dz \tag{8.4}$$

where ρ_o is the liquid density corresponding to the coolant inlet temperature To. By utilizing the heat removal equation on core region j

$$\left(T_j(z) - To\right) \left(G\, A_x\, C_p\right)_j = \int_0^z \dot{q}_j(z)\, dz \quad , \tag{8.5}$$

equation (8.4) can be expressed as

$$\bar{\rho}_{\ell,\text{ core } j} = \rho_o \left[1 + \frac{1}{2} \frac{\beta \dot{Q}_j}{(G A_x C_p)_j}\right] \tag{8.6}$$

assumed a homogeneous linear heat generation, where $\dot{Q}_j$ is the total heat generation in core region j.

$$\dot{Q}_j = \int_0^H \dot{q}_j (z)\, dz \tag{8.7}$$

Based on equations (8.2) and (8.6), the steady-state mass flow rate through the coolant channel in region j, $W_j = G_j A_x$ was calculated as listed in Table 10. These mass flow rates were input to the code as the initial conditions of junction flow. According to the steady-state continuity, the circulation flow through junctions 78 and 86 were given to be 0.3235 lb/sec.

ENERGY BALANCE

The energy balance of each control volume in steady-state is obtained by setting the time derivatives to zero.

$$\sum_{j \varepsilon \text{ out}} W_j \left[h + \frac{1}{2} \frac{1}{g_c} \left(\frac{W}{\rho A_x}\right)^2 + \frac{g}{g_c} \hat{z}\right]_j - \sum_{j \varepsilon \text{in}} W_j \left[h + \frac{1}{2} \frac{1}{g_c} \left(\frac{W}{\rho A_x}\right)^2 + \frac{g}{g_c} \hat{z}\right]_j = \dot{Q}_L \tag{8.8}$$

The steady-state junction enthalpy, h_j can be obtained from equation (8.8) according to the junction flow W_j given in Table 10. The volume average enthalpy then can be computed by using the following relation

$$\bar{h}_k = h_j - \frac{1}{2} \frac{1}{g_c} \left(\frac{W}{\rho A_x}\right)_k^2 + \frac{1}{2} \frac{1}{g_c} \left(\frac{W}{\rho A_x}\right)_j^2 - \frac{g}{g_c} \hat{z}_k + \frac{g}{g_c} \hat{z}_j - \Delta h_Q \tag{8.9}$$

Based on equations (8.8), (8.9) and mass flow rates obtained from momentum equations, the average enthalpy of each control volume can be calculated as the results shown in Table 11. The steady-state enthalpy in lower tank (volume 6) is assumed 36.0 Btu/lb (68^0F). These values of volume enthalpy were input to the code as the initial conditions of control volume energy.

14.9 RESULTS

Large reactivity insertion over short period of time was studied for finite reactivity ramp. Reactivity insertion of 0.41 $ which is the excess reactivity of RPI core 2, with insertion of 0.5 seconds, were analyzed. The core can obtain the negative reactivity by dumping the moderator during the events. The analyses were performed for two cases: the negative reactivity begins at 12 seconds and 60 seconds after the beginning of event according to the measured rate of moderator dump and the limitation of technical specification. The reactivity changes of these two cases are shown in Figures 6 and 10, respectively.

For the 12 seconds case, the reactor power transient for initial power of 100 watts is given in Figure 7, which shows a peak power of 910 watts. The fuel temperatures as functions of time for control volumes 11 and 22 are shown in Figure 14.8. Control volume 22 is the "hot channel" which has a peak fuel temperature of 69.3^0F at 13.25 seconds. Figure 14.9 shows the coolant temperatures of volumes 11 and 22. The increase of coolant temperature is less than 0.2^0F.

For the 60 seconds case, the transients of reactor power, fuel temperatures, and coolant temperature are shown in Figures 14.11, 14.12 and 14.13. The peak power of the reactor was predicted to be 0.11 MW. The maximum fuel temperature and coolant temperature in the "hot channel", volume 22 are 129°F and 83°F, respectively.

In both cases, the resulting peak fuel and coolant temperature are significantly below the melting point of fuel and the boiling point of water, so that no core melting and bulk boiling will occur throughout the transients.

4.10 REFERENCES

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Table 14.1 Fuel Plate Dimensions and Loading

	Stationary Assembly Plate	Moveable Assembly Plate
Plate Thickness, in.	0.030	0.030
Plate Length, in.	23.0	23.0
Plate Width, in.	2.77	2.55
Fuel Thickness, in.	0.020	0.020
Fuel Length, in.	21.75	21.125
Fuel Width, in.	2.54	2.32
Clad Thickness, in.	0.005	0.005
Fuel Loading, gm. U-235	28.62	25.02
Poison Loading, gm. B-10	0.020	0.018

Table 14.2 Nuclear and Physical Characteristics of the RPI Core 2

Effective Delay Neutron Fraction $\beta = 0.0078$

Effective Neutron Lifetime $\ell^* = 8.2 \text{x} 10^{05}$ sec

Delay Neutron Data

Group #	$\beta i/\beta$	λi
1	0.041	3.01
2	0.115	1.14
3	0.396	0.301
4	0.196	0.111
5	0.219	0.0305
6	0.033	0.0124

Reactor Power P = 100 watts

Axial Power Shape chopped sine

Coolant Temperature T = 68°F

Table 14.3 Kinetics Parameters of RPI Core 2 and Technical Specifications

Kinetics Parameter	Core 2 Value	Technical Specification
Excess Reactivity at 68°F	0.0032	<0.0078
Reactivity with One Rod Stuck	-0.017	<-0.005
Shutdown Margin	0.0271	>0.02
Core Average Isothermal Temperature Coefficient of Reactivity ($\bar{\alpha}_T$)	<0 for T>61°F	<0 for T>90°F
$\int_{50°F}^{T} dT\, \bar{\alpha}_T$	6.05×10^{-5}	$<8\times10^{-4}$
Core Average Void Coefficient of Reactivity ($\bar{\alpha}_V$)	$-7\times10^{-6}/cm^3$ at 57°F	$<-3\times10^{-6}/cm^3$
Reactivity Worth of Standard Fuel Assembly	≤0.025	<0.039

Table 14.4 Measured Feedback Coefficient
for RPI Core 2

Core Average Void Coefficient of Reactivity $\bar{\alpha}_v = 7\text{x}10^{-6}/\text{cm}^3$

Assembly Average Void Coefficient of Reactivity

Assembly Coordinates	Average Void Coefficient ($\Delta\rho/\text{cm}^3$)
(44)	$-2.4\text{x}10^{-6}$
(34)	$-10.\text{x}10^{-6}$
(33)	$-8.8\text{x}10^{-6}$
(24)	$-5.5\text{x}10^{-6}$
(23)	$-4.9\text{x}10^{-6}$
(22)	$-3.6\text{x}10^{-6}$

Central Water Region Value of Void Coefficient

$$(\alpha_v) \text{ center} = +5.5\text{x}10^{-6}$$

Moderator Temperature Coefficient of Reactivity

$$(\alpha_T)_m = 6.10\text{x}10^{-5} \left(1 - \frac{T°F}{61}\right) \text{ per °F}$$

Fuel Temperature Coefficient of Reactivity*

$$(\alpha_T)_f = -9.72\text{x}10^{-6} \text{ per °F}$$

* The value is obtained from the measurements performed at the Kyoto University Reactor which is the same type reactor as RPI Critical Facility.

Table 14.5 Measured Data of Reactivity Decrease due to Water Dump

Water Level below Top of Core (in.)	Reactivity Decrease ($)
2	-0.53
2.50	-0.77
3	-1.39
3.50	-1.76
4	-1.96
4.75	-2.27
6.75	-3.16
8.75	-3.39
10.80	-3.53

Notes: 1) Measured rate of dump is 12 seconds from normal level to the top of the core.

2) Technical specification requires maximum time of 60 seconds from initiation of dump until negative reactivity is inserted.

Table 14.6 Thermal-hydraulic Parameters of Core Regions

Regions	1	2	3	4	5
Water Volume (ft^3)	0.045	0.205	0.757	0.632	0.521
Flow area (ft^2)	0.023	0.107	0.395	0.330	0.272
Hydraulic dia. (ft)	0.142	0.019	0.041	0.060	0.142
Volume of (ft^3) heat conductor	0.001	0.052	0.084	0.020	0.006
Power fraction	0.007	0.240	0.368	0.098	0.042
Heat transfer (ft^2) area	0.8	41.6	67.2	37.6	4.8

Table 14.7 Power Fractions of Volumes and Heat Conductors in Core 2

Volumes of Heat Conductors	Power Fraction
11	0.007
21	0.060
22	0.120
23	0.060
31	0.092
32	0.184
33	0.092
41	0.086
42	0.171
43	0.086
51	0.042

Table 14.8 I_j Values of Assemblies for Core 2 obtained from NODER Solution

Assembly Coordinates	I_j
(44)	0.0686
(34)	0.0506
(33)	0.0475
(24)	0.0382
(23)	0.0310
(22)	0.0211

Table 14.9 Reactivity Feedback Parameters of Volumes and Heat Conductors in Core 2

Volumes of Heat Conductors	Moderator Density Factor (WF_i)	Doppler Factor (DF_i)	Moderator Temperature Coefficient ($)
11	-0.019*	0.010	0.75×10^{-4}
21	0.026	0.063	-0.72×10^{-4}
22	0.099	0.244	-3.56×10^{-4}
23	0.026	0.063	-1.08×10^{-4}
31	0.075	0.086	-2.10×10^{-4}
32	0.292	0.334	-10.12×10^{-4}
33	0.075	0.086	-3.15×10^{-4}
41	0.042	0.016	-1.29×10^{-4}
42	0.164	0.060	-6.15×10^{-4}
43	0.042	0.016	-1.93×10^{-4}
51	0.178	0.022	-5.16×10^{-4}

* The negative value means a positive void coefficient of reactivity in this volume.

Table 14.10 Steady State Mass Flow Rate through Coolant Channel in Core Region

Core Region	Mass Flow Rate (lb/sec)
1	0.0105
2	0.0176
3	0.0905
4	0.1170
5	0.0879

Table 14.11 Average Enthalpy of Each Fluid Volumes

Volumes	h (Btu/lb)
6	36.0000
11	36.0200
21	36.1204
22	36.4264
23	36.7324
31	36.0345
32	36.1336
33	36.2327
41	36.0055
42	36.0301
43	36.0547
51	36.0130
7	36.1513
8	36.0756

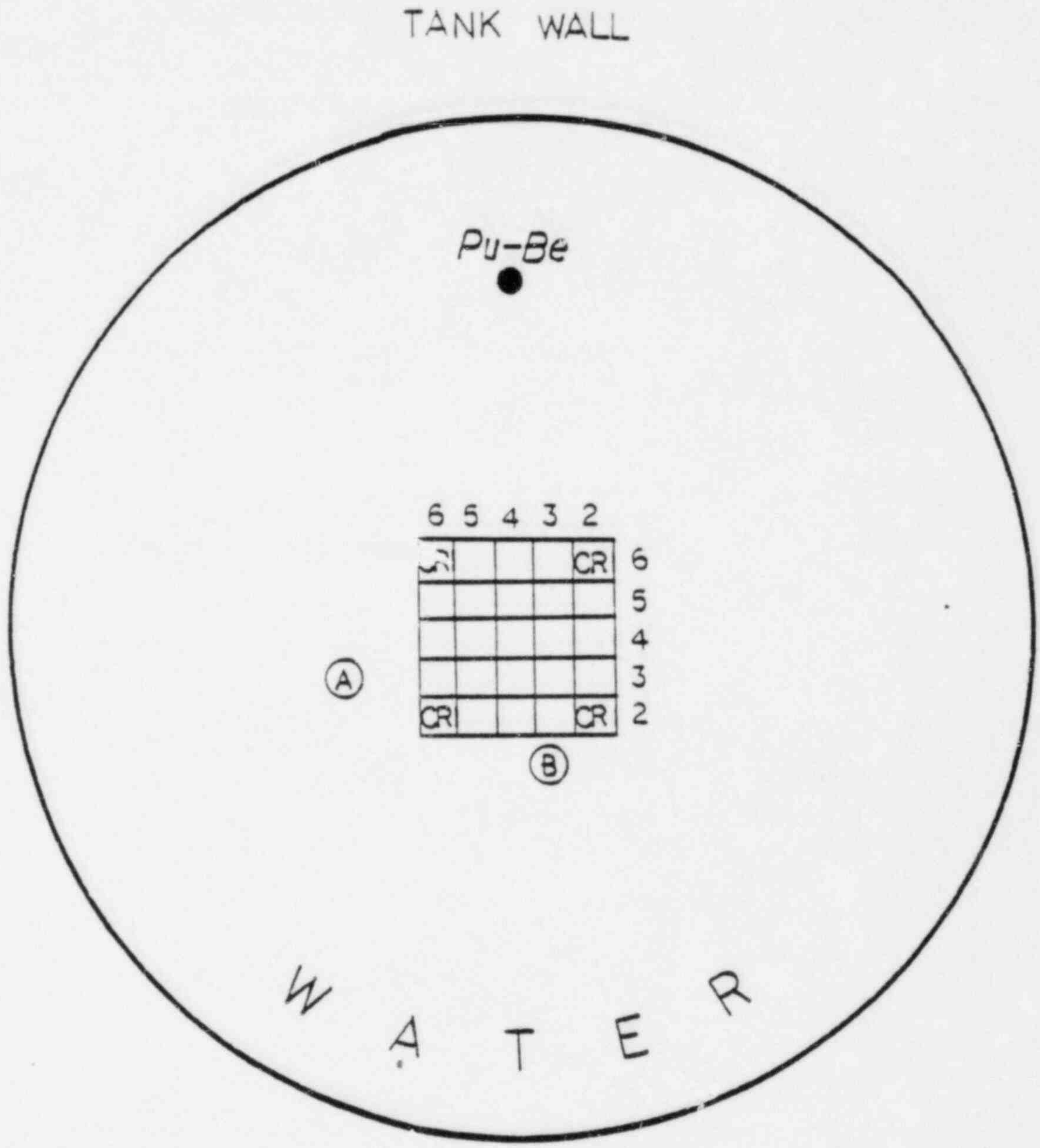
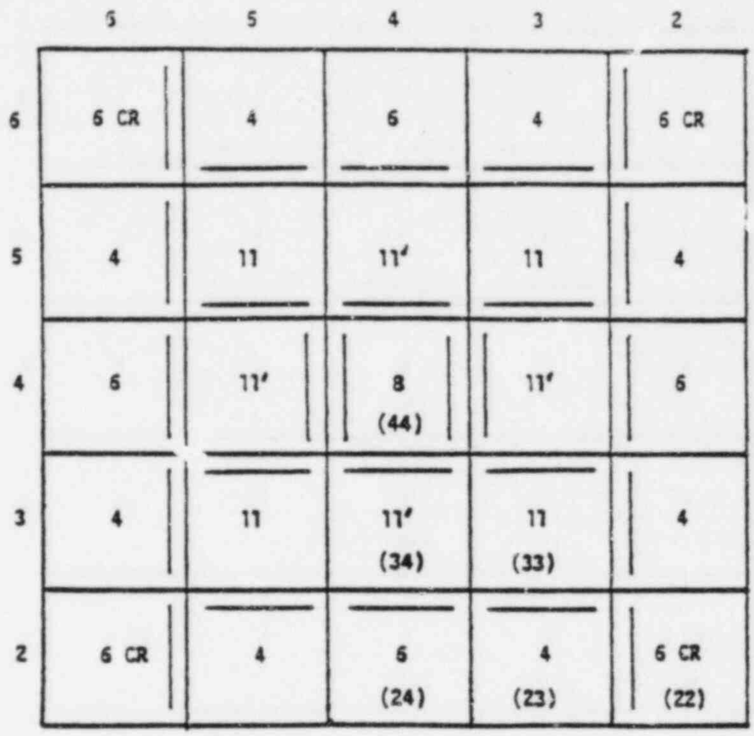


Figure 14.1 Reactor Top Schematic



Assembly Type (Fuel Plates)	Assembly Plate Slot Number 1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18
8	1	0	1	1	1	0	0	0	0	0	0	0	0	1	1	1	0	1
11	1	1	1	1	1	0	1	0	1	0	1	0	1	0	1	0	1	0
11'	1	1	0	1	1	0	1	0	1	0	1	0	1	0	1	0	1	1
6	1	0	1	0	0	1	0	0	1	0	0	0	1	0	0	1	0	0
4	1	0	0	1	0	0	0	1	0	1	0	0	0	0	0	0	0	0
6 CR*	1	0	1	0	1	0	1	0	1	0	1	0	0	0	0	0		

"1" indicates presence of a fuel plate

Figure 14.2 Fuel and Control Rod Arrangement

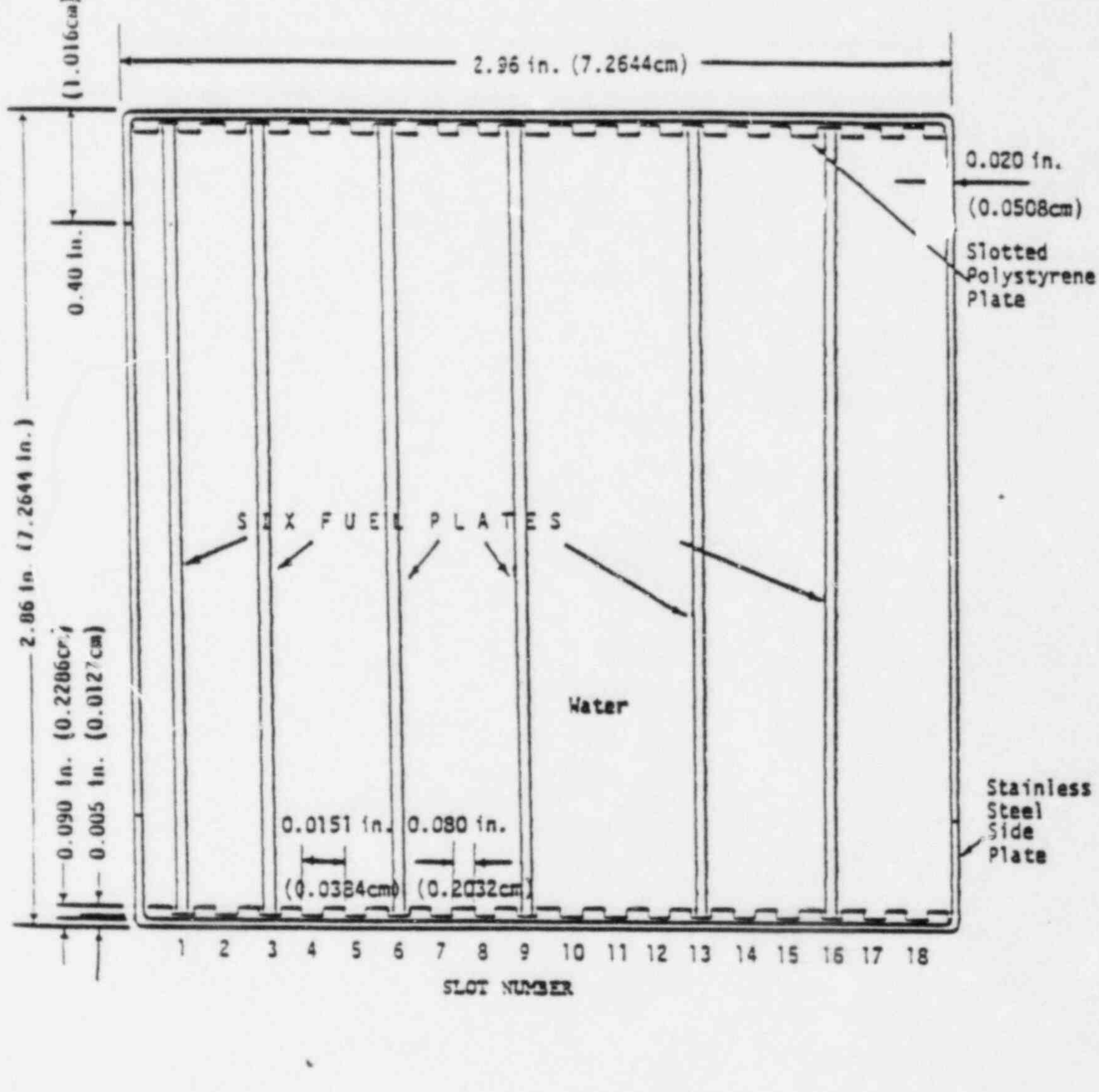


Figure 14.3 Fuel Plate Options

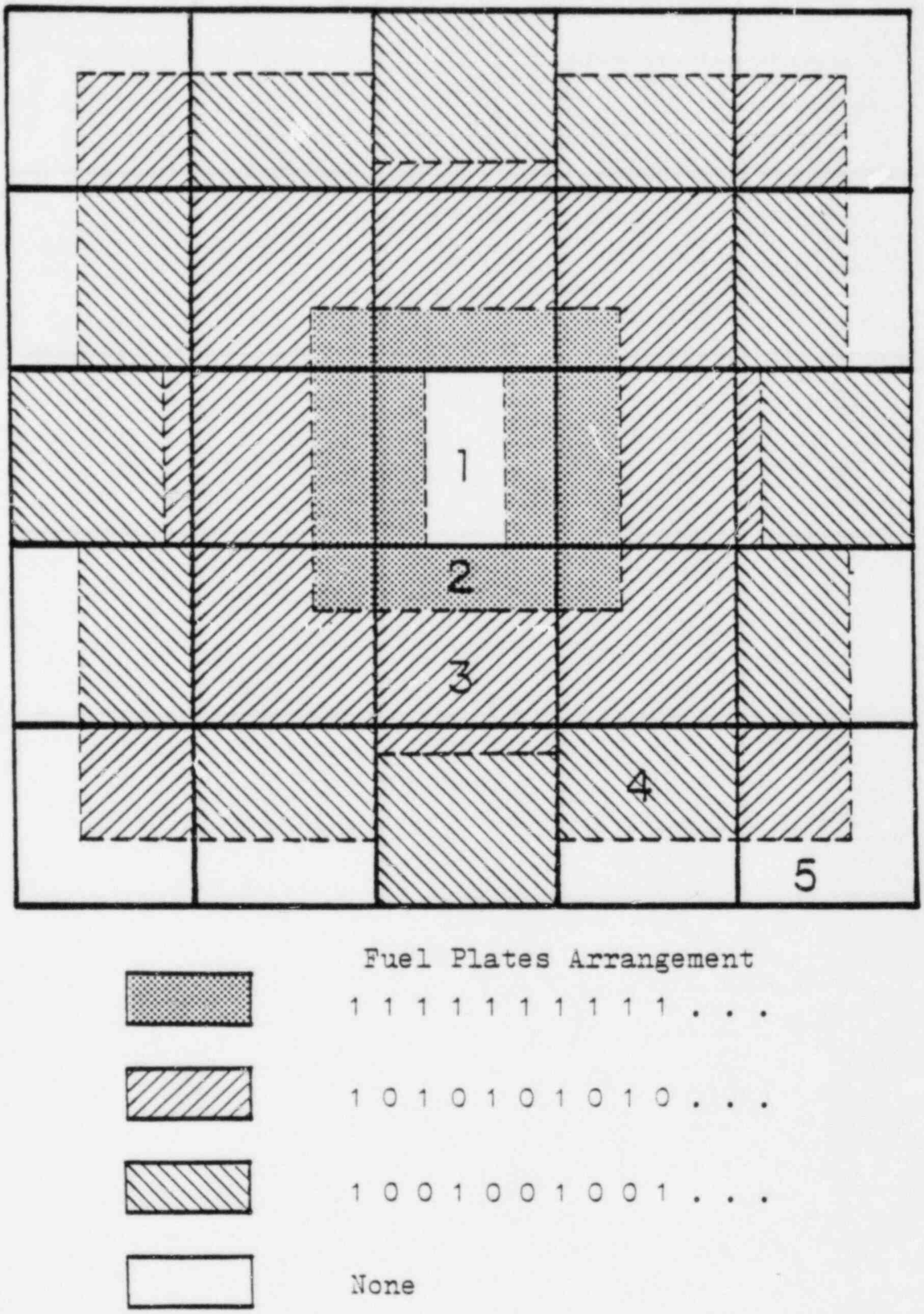


Figure 14.5a RETRAN Volumes

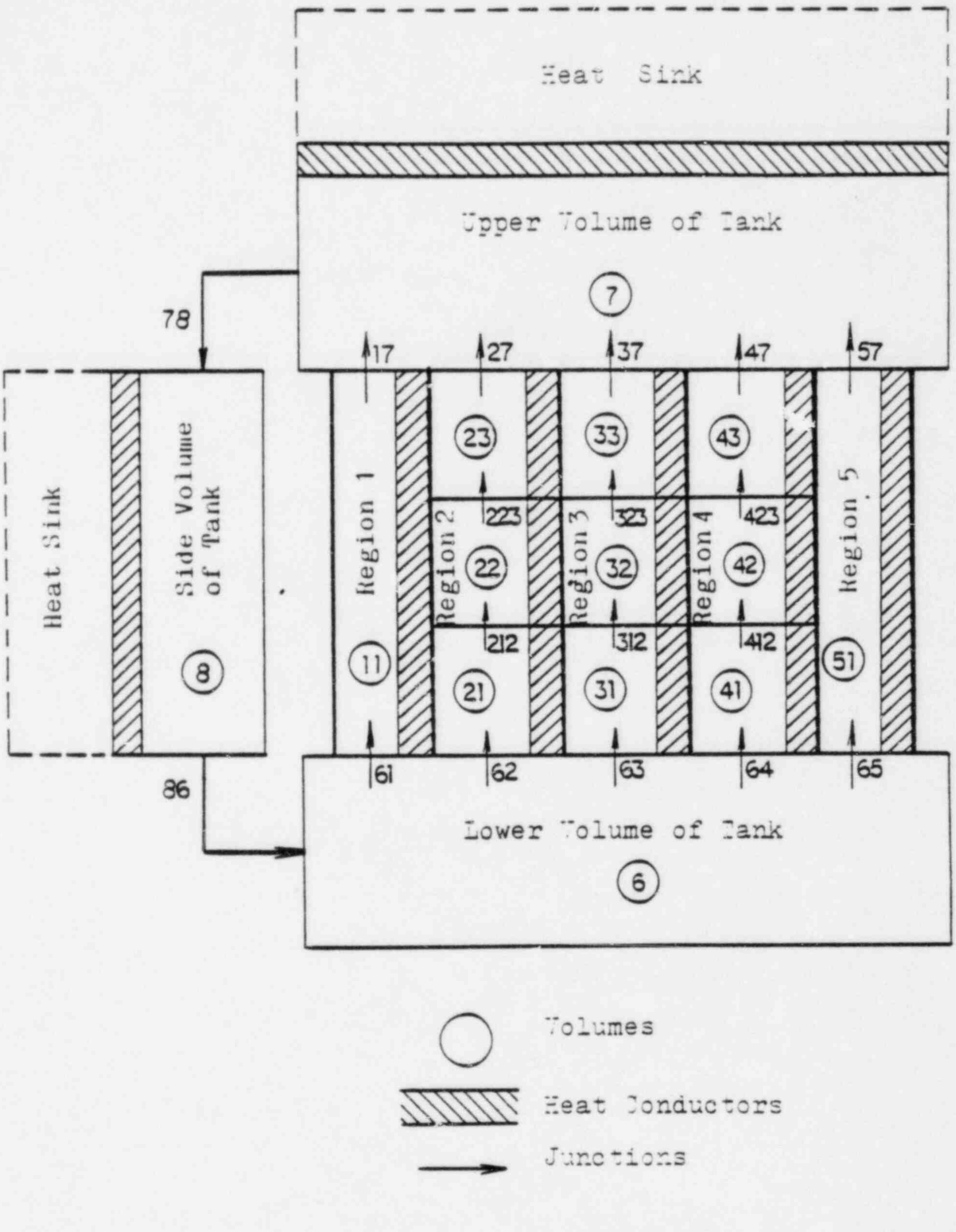


Figure 14.5b RETRAN Volumes

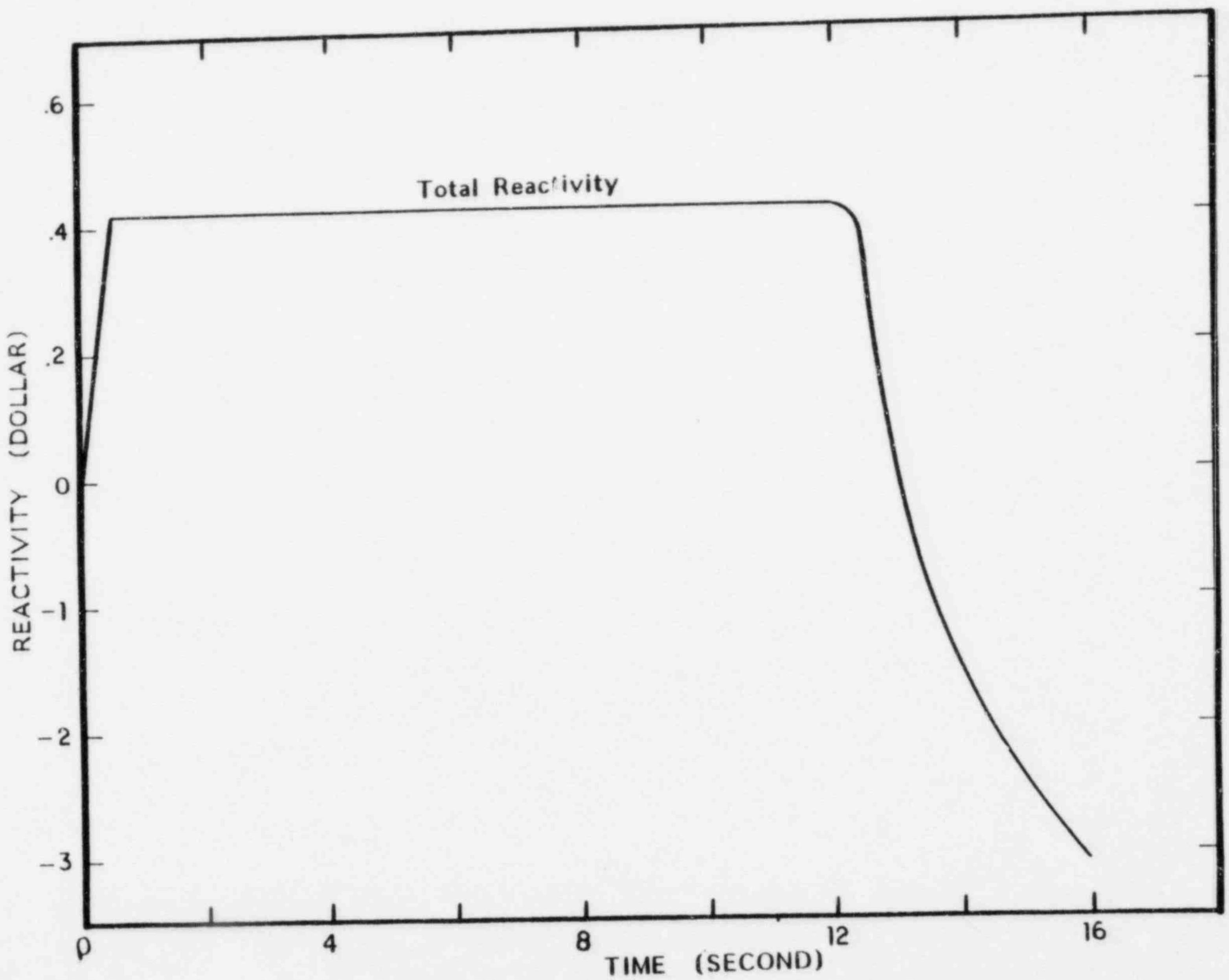


Figure 14.6 Reactivity Transient for Water Dump Reactivity Decrease at 12 Seconds after Rod Withdrawal Initiation.

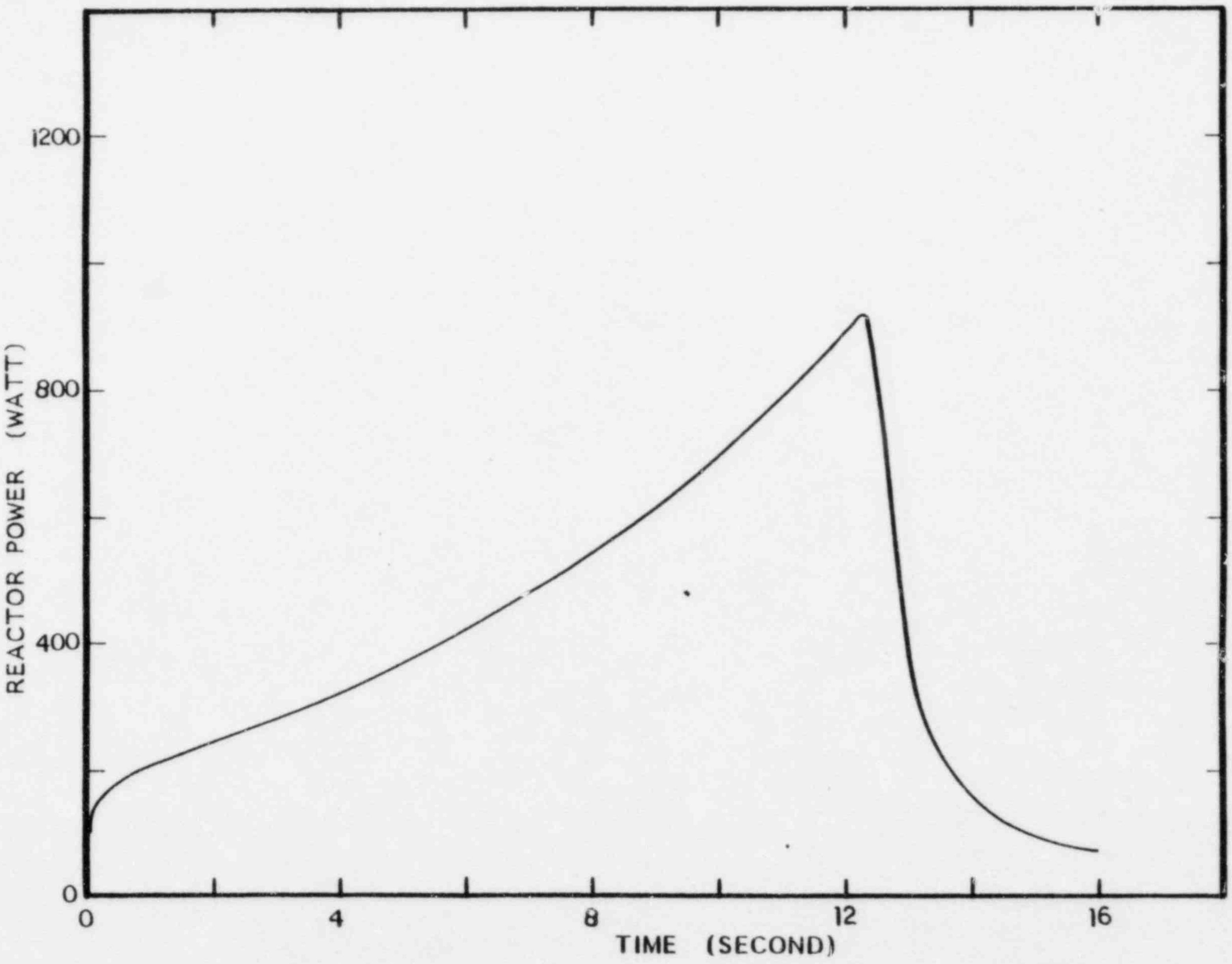


Figure 14.7 Reactor Power Transient for Water Dump Reactivity Decrease at 12 Seconds After Rod Withdrawal Initiation.

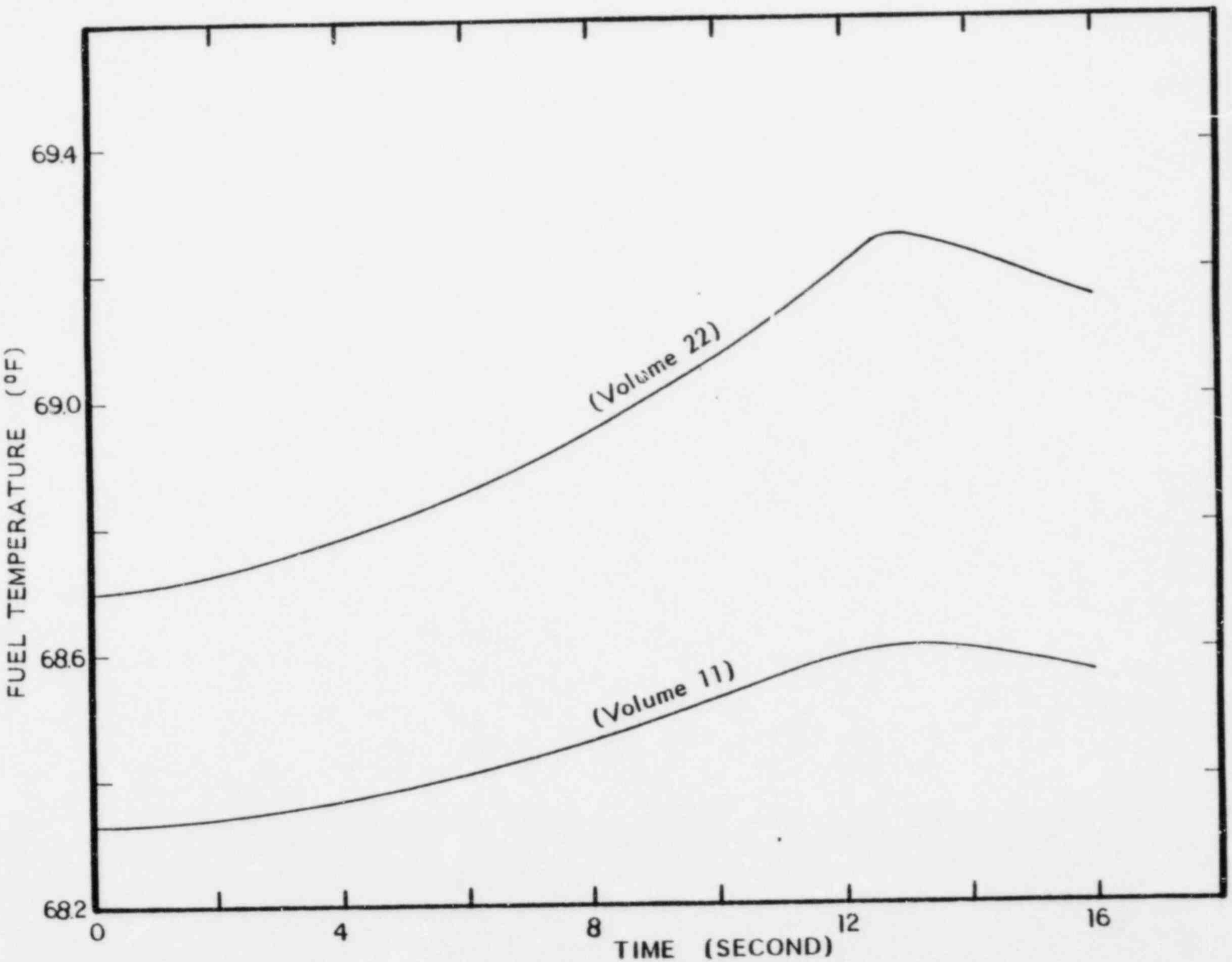


Figure 14.8 Fuel Temperature Transient for Water Dump Reactivity Decrease at 12 Seconds After Rod Withdrawal Initiation.

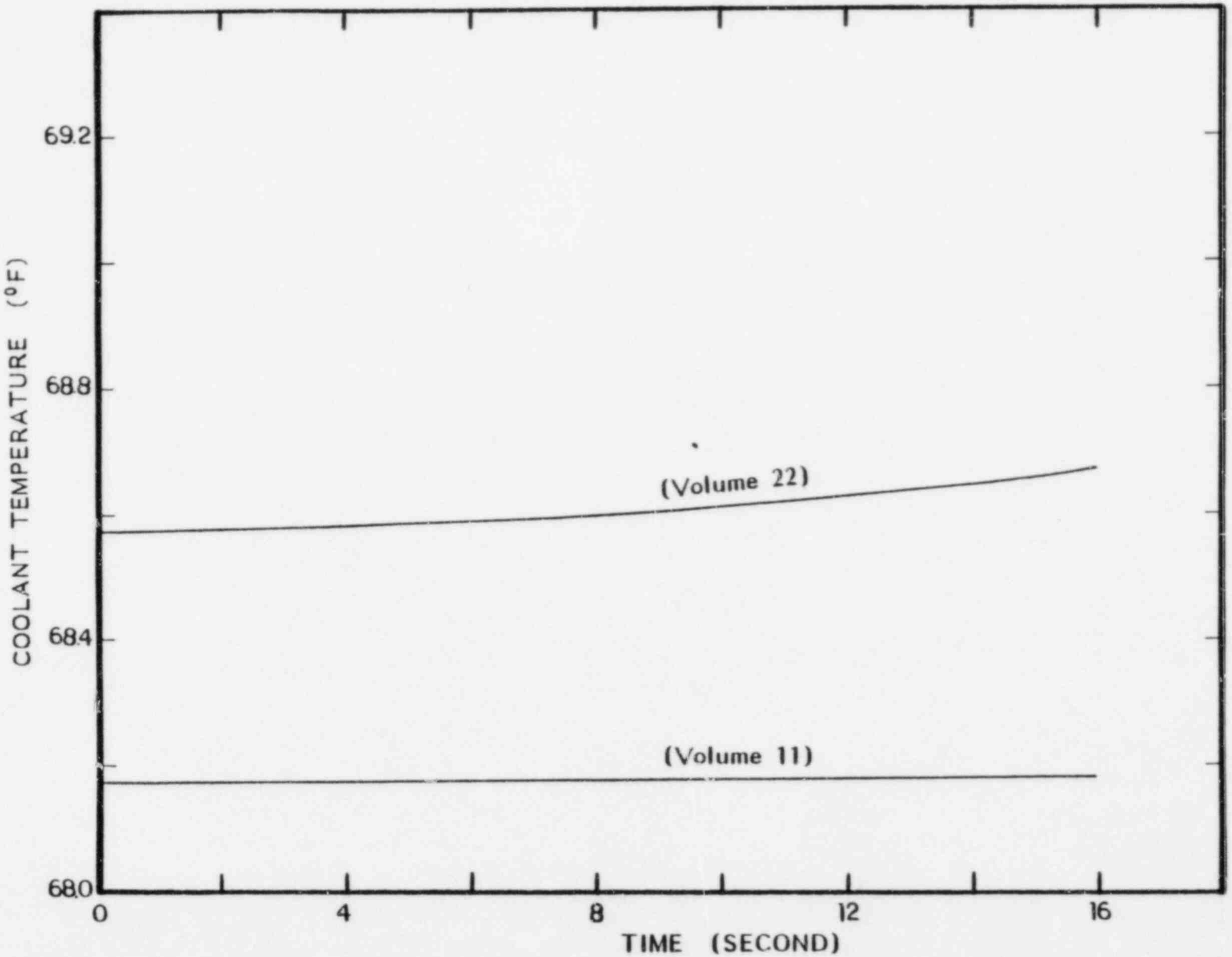


Figure 14.9 Coolant Temperature Transient for Water Dump Reactivity Decrease at 12 Seconds after Rod Withdrawal Initiation.

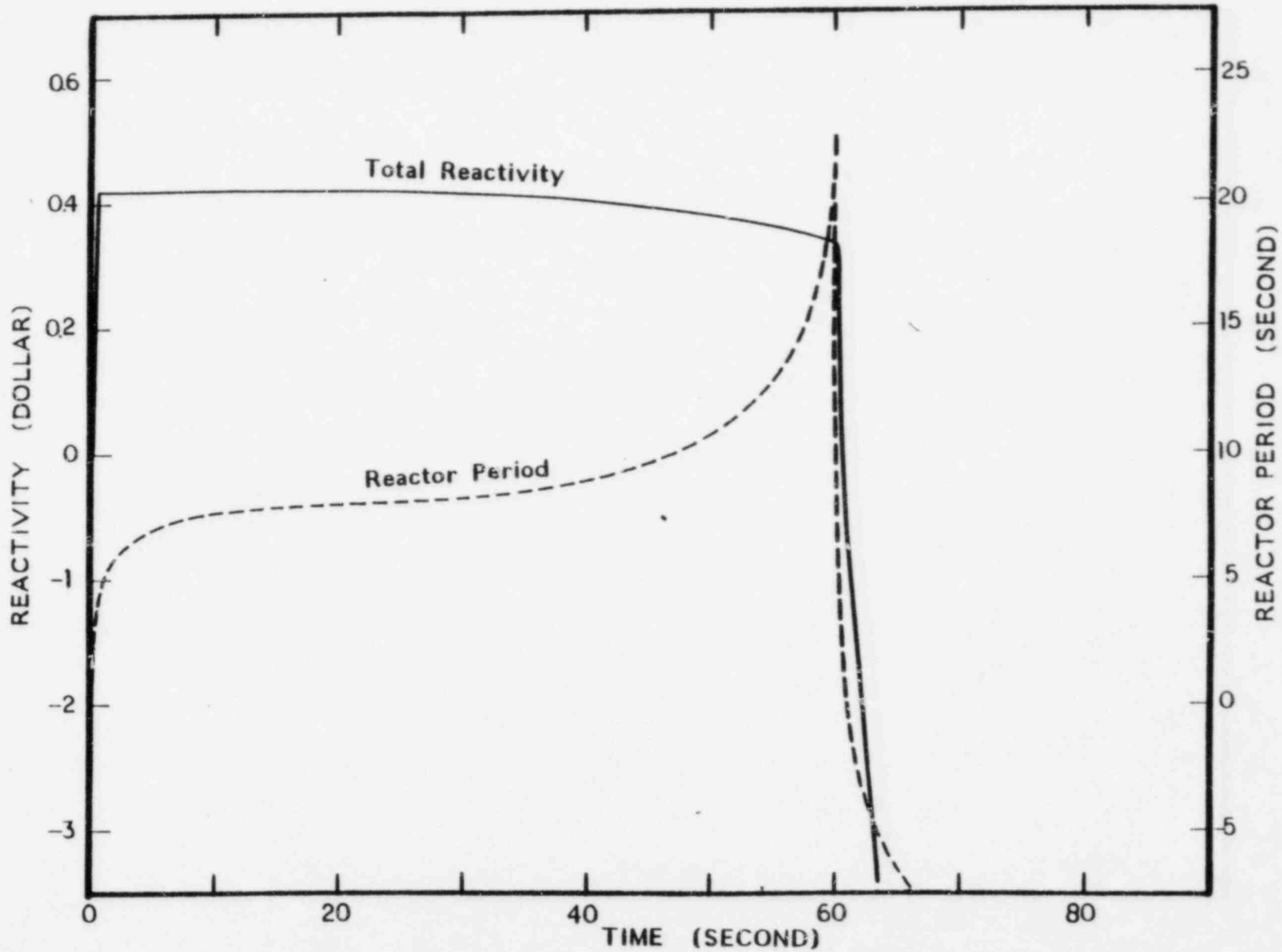


Figure 14.10 Reactivity Transient for Water Dump Reactivity Decrease at 60 Seconds after Rod Withdrawal Initiation.

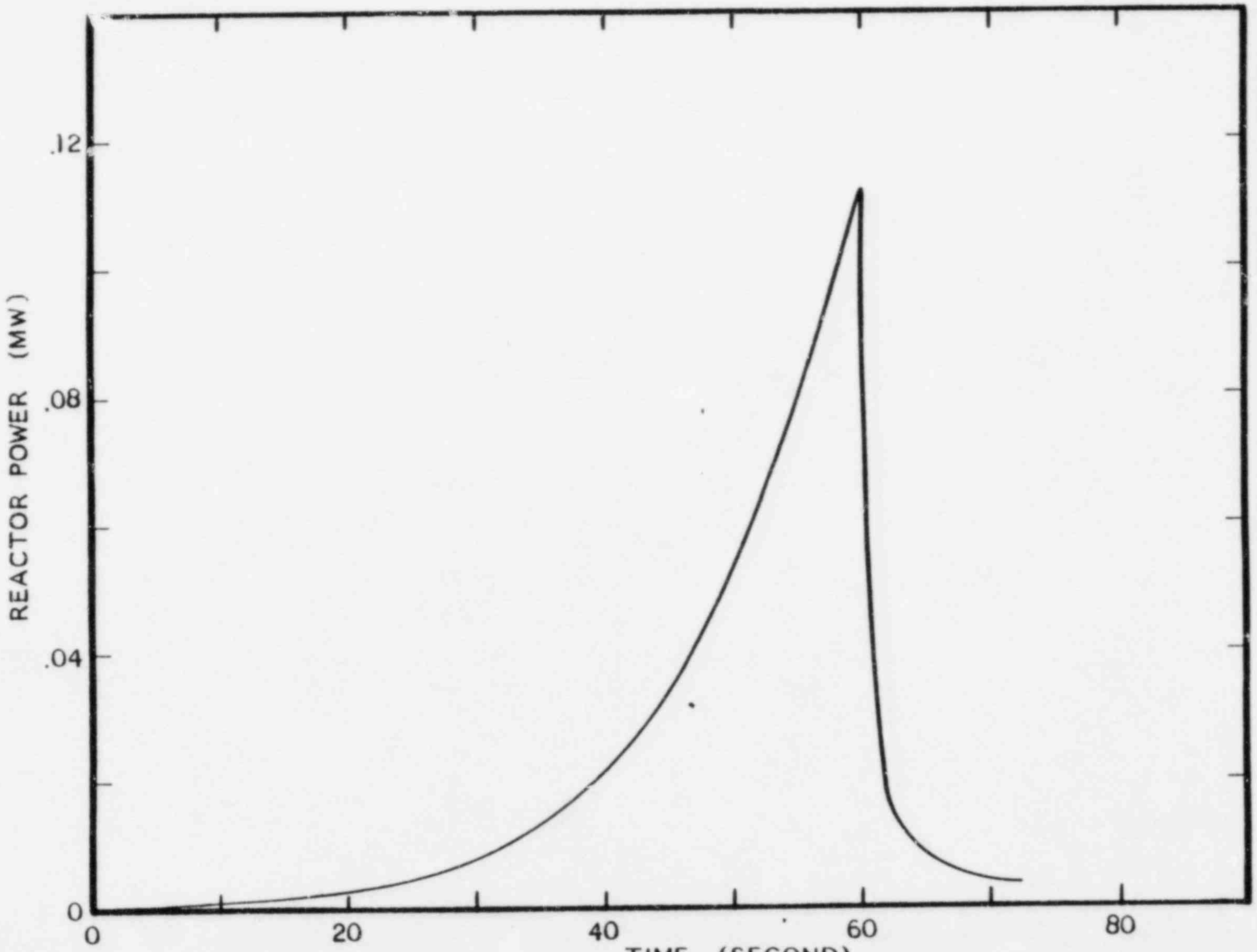


Figure 14.11 Reactor Power Transient for Water Dump Reactivity Decrease at 60 Seconds after Rod Withdrawal Initiation.

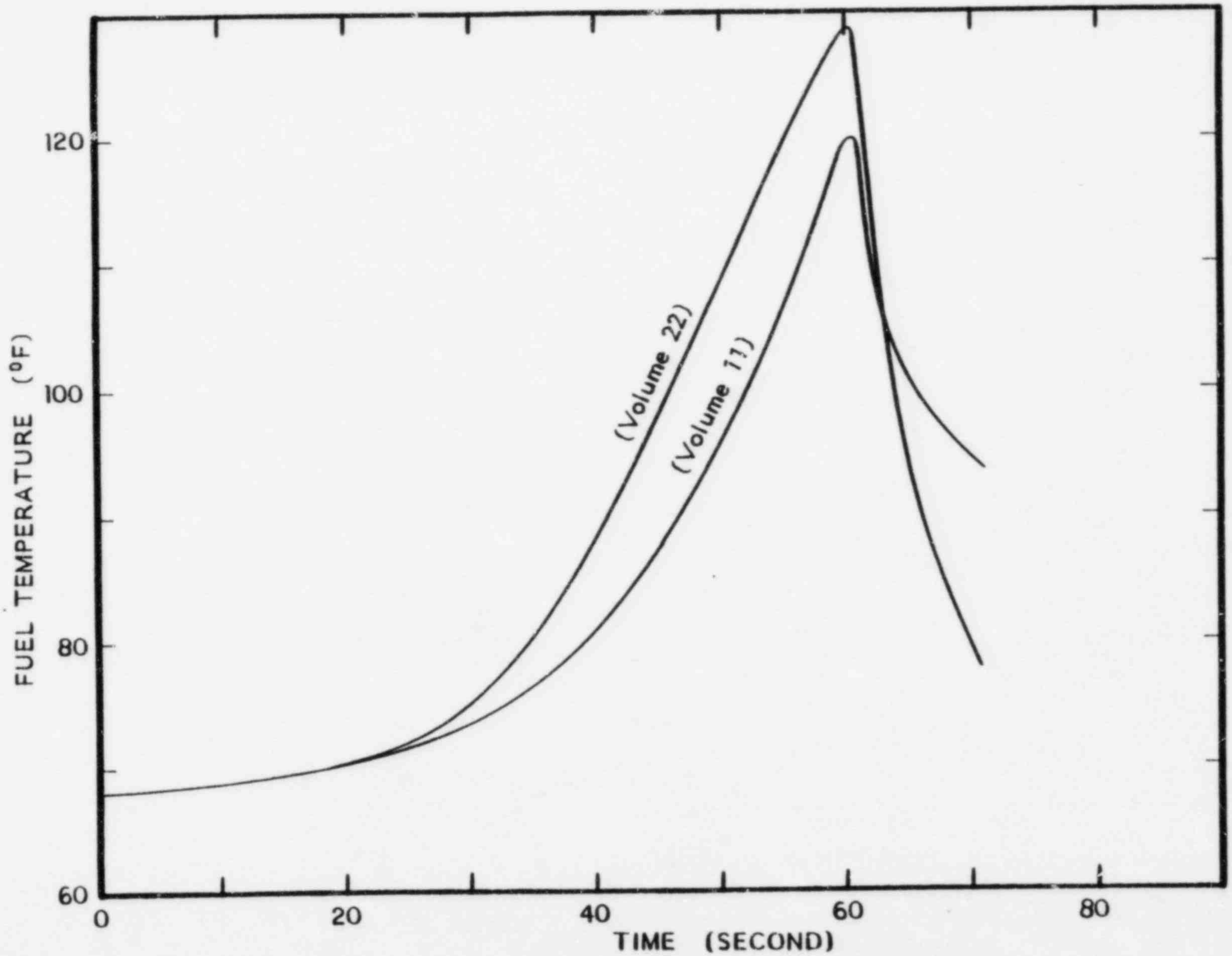


Figure 14.12 Fuel Temperature Transient for Water Dump Reactivity Decrease at 60 Seconds after Rod Withdrawal Initiation.

15. Technical Specifications

The attached Technical Specifications are unchanged from October 2, 1975 with the exception of Section 6.0, Administrative Controls, which has been modified to comply with subsequently issued administrative requirements.

APPENDIX A

TO

FACILITY LICENSE NO. CX-22

TECHNICAL SPECIFICATIONS

AND BASES

FOR THE

RENSSELAER POLYTECHNIC INSTITUTE

CRITICAL EXPERIMENTS FACILITY

SCHENECTADY, NEW YORK

DOCKET NO. 50-225

(CHANGE NO. 3)

January, 1983

TABLE OF CONTENTS

1.0 DEFINITIONS

The terms Safety Limit (SL), Limiting Safety System Setting (LSSS), and Limiting Condition for Operation (LCO), and Surveillance Requirements are as defined in 50.36 of 10 CFR Part 50.

A. Excess Reactivity - The available reactivity above a cold clean critical configuration which may be added by manipulation of controls.

B. Safety Channel - A measuring channel in the reactor safety system.

C. Reactor Safety System - Combination of safety channels and associated circuitry which forms the automatic protective system for the reactor or provides information which requires manual protective action to be initiated.

D. Channel Check - Qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

E. Channel Test - The injection of a simulated signal into the instrument primary sensor to verify the proper instrument response alarm and/or initiating action.

F. Channel Calibration - The correlation of channel outputs to known input signals and other known parameters. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip.

G. Operable - A system or component is capable of performing its intended function in its required manner.

H. Operating - A system or component is performing its intended function in its required manner.

I. Reactor Scram - A gravity drop of the control rods accompanied by the opening of the moderator dump valve. The moderator dump valve may be bypassed by a senior operator licensed pursuant to 10 CFR 55, if the cause of scram is known, all control rods are verified to have scrammed, and it is deemed wise to retain the moderator shielding in the reactor tank. The scram can be initiated either manually or automatically by the safety system.

J. Source - A neutron-emitting radioactive material, other than reactor fuel, which is positioned in or near the assembly to provide an external source of neutrons.

K. Review and Approve - The reviewing group or person shall carry out a review of the matter in question and may either approve or disapprove it; before it can be implemented, the matter in question must receive approval from the reviewing group or person.

L. Control Rod Assembly - A control mechanism consisting of a top absorber section and a lower fuel follower section.

1. Control Rod Absorber Section - These may contain either enriched boron in iron, EuO_3 in a stainless steel cermet, stainless steel, or an alloy of silver-cadmium-indium. All absorber sections except the one containing silver-cadmium-indium are clad in stainless steel. All are of the same dimensions, nominally 2.6 inches square, with their poisons uniformly distributed. The absorbers, when fully inserted, shall extend above the top and to within one inch of the bottom of the active core.

2. Control Rod Follower Section - An array of up to 16 stainless steel plates or stainless steel clad fuel plates containing 93 percent enriched fissionable oxides of uranium in a stainless steel cermet.

M. Reportable Occurrence - The occurrence of any facility condition that:

1. Causes a Limiting Safety System Setting to exceed the setting established in Section 2 of the Technical Specifications,

2. Exceeds a Limiting Condition for Operation as established in Section 3 of the Technical Specifications,

3. Causes any uncontrolled or unplanned release of radioactive material from the restricted area of the facility,

4. Results in safety system component failures which could, or threaten to, render the system incapable of performing its intended safety function as defined in the Technical Specifications or SAR,

5. Results in abnormal degradation of one of the several boundaries which are designed to contain the radioactive materials resulting from the fission process,

6. Results in uncontrolled or unanticipated changes in reactivity of greater than 0.5% delta k/k,

7. Causes conditions arising from natural or offsite manmade events that affect or threaten to affect safe operation of the facility, or

8. Results in observed inadequacies in the implementation of administrative or procedural controls such that the inadequacy causes or threatens to cause the existence or development of an unsafe condition in connection with the operation of the facility.

N. Reactor Shutdown - The control rod(s) are inserted and the reactor is shutdown by at least 2% delta k/k. The reactor is considered to be operating whenever this condition is not met and there are 12 or more fuel elements loaded in the core.

O. Reactor Secured - (1) The full insertion of all control rods has been verified, (2) the console key is removed, and (3) no operation is in progress which involves moving fuel elements in the reactor vessel, the insertion or removal of experiments from the reactor vessel or control rod maintenance.

P. True Value - The actual value at any instant of a process variable.

Q. Measured Value - The value of the process variable as it appears on the output of a measuring channel.

R. Measuring Channel - The combination of sensor, lines, amplifiers, and output devices which are connected for the purpose of measuring the value of a process variable.

S. Experiment - (1) An apparatus, device, or material placed in the reactor vessel and/or (2) any operation designed to measure reactor characteristics.

T. Secured Experiment - Any experiment, experimental facility, or component of an experiment is deemed to be secured, or in a secured position, if it is held in a stationary position relative to the reactor. The restraining forces must be equal to or greater than those which hold the fuel elements themselves in the reactor core.

U. Unsecured Experiment - Any experiment, experimental facility, or component or an experiment is deemed to be unsecured if it is not and when it is not secured. Moving parts of experiments are deemed to be unsecured when they are in motion.

V. Movable Experiment - A movable experiment is one which may be inserted, removed, or manipulated while the reactor is critical.

W. Readily Available on Call - The Licensed Senior Operator on duty shall remain within a 15 mile radius or 30 minutes travel time of the facility, whichever is closer, and the operator-on-duty shall know the exact location and telephone number of the LSO on duty.

X. Surveillance Frequency - Unless otherwise stated in these specifications, periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus or minus 25%. In cases where the elapsed interval has exceeded 100% of the specified interval, the next surveillance interval shall commence at the end of the original specified interval.

Y. Surveillance Interval - The surveillance interval is the calendar time between surveillance tests, checks, calibrations, and examinations to be performed upon an instrument or component when it is required to be operable. These tests may be waived when the instrument, component, or system is not required to be operable, but the instrument, component, or system shall be tested prior to being declared operable.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits - Reactor Power

Applicability

Applies to variables associated with reactor power.

Objective

To establish the maximum power level and annual integrated power below which fuel cladding is preserved and fission product inventories are acceptably limited.

Specification

1. The thermal power level shall not exceed 270 watts.

2. The integrated thermal power for any 365 consecutive days shall not exceed 200 kilowatt-hours.

Bases

Since a critical experiment does not contain significant quantities of radioactivity, a definite safety limit is not required to protect the integrity of the physical barriers which guard against the uncontrolled release of radioactivity. A steady state thermal power level of 100 watts and an integrated thermal power of 200 kW-hrs appropriately limit the quantity of radioactivity available for release and provides adequate flexibility for the performance of training and educational operations.

The 270 watt limit is stipulated because at that level the change in moderator and fuel temperature will not result in damage to reactor components or compromise the integrity of the fuel clad.

Measurements have shown that during normal steady state operation the average moderator temperature increase is negligible and the clad and fuel temperatures remain far below their failure points. In general, the operating power level is kept as low as practicable, consistent with experimental and educational operation requirements and normally below 20 watts.

2.2 Limiting Safety System Settings - Reactor Power

Applicability

Applies to the settings for instruments monitoring parameters associated with the reactor power limits.

Objective

To assure protective action before safety limits are exceeded.

Specification

The limiting safety system settings on reactor power shall be as follows:

Maximum Power Level	135 watts
Minimum Flux Level	2.0 counts/sec
Minimum Period	5 seconds

Bases

The maximum power level trip setting of 135 watts corresponds to a reading of not greater than 90% on the last scale of either linear power channel as established by activation techniques. This safety margin is sufficient to account for uncertainties in this power calibration and instrumentation. The minimum flux level has been established to prevent a source-out start-up. The interlock set point on the source level channel is 2 cps. The specified minimum flux level will assure that this interlock is satisfied.

The minimum 5-second period is specified so that the automatic safety system channels have sufficient time to respond before safety limits are exceeded.

Administrative control of annual integrated thermal power shall be used to meet the safety limit of 200 kilowatt-hours in any consecutive 365 days.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Control and Safety Systems

Applicability

Applies to all methods of changing core reactivity available to the reactor operator.

Objective

To assure that available shutdown method is adequate and that positive reactivity insertion rates are within those analyzed in the Hazards Summary Report (hereinafter safety analysis report).

Specifications

1. The excess reactivity of the reactor core above cold, clean critical shall not be greater than 3.9% delta k/k. The maximum number of fuel assemblies contained in the reactor vessel shall be 45. The maximum reactivity worth of any clean fuel assembly shall be "3.9%" delta k/k.

2. There shall be a minimum of three operable control rods. The reactor shall be subcritical by more than 0.5% delta k/k with the most reactive control rod fully withdrawn.

3. The maximum control rod reactivity rate shall be less than 0.084% delta k/k/sec up to 10 times source level and 0.033% delta k/k/sec at all higher levels.

4. The total control rod drop time for each control rod from its fully withdrawn position to its fully inserted position shall be less than or equal to 900 milliseconds. This time shall include a maximum magnet release time of 50 milliseconds.

5. The auxiliary reactor scram (moderator-reflector water dump) shall add negative reactivity within one minute of its activation.

6. The normal moderator-reflector water level shall be established not greater than 10 inches above the top grid of the core.

7. The minimum safety system channels that shall be operating during the reactor operation are listed in Table I.

TABLE I

MINIMUM SAFETY SYSTEM CHANNELS

REACTOR CONDITIONS AND RANGES	CHANNELS	MINIMUM NUMBER	FUNCTIONS
Startup 2 cps -- 10^4 cps	Log Count Rate[a]	1	Minimum Flux Level
Power - 10^{-4} - 150% Full Power	Linear Power	2	High Neutron Level Scram
10^{-3} - 300% Full Power	Log-N Period[b]	1	Neutron Level Indication Period Scram
	Manual Scram[c]	2	Reactor Scram
	Building Power	1	Loss of Power Scram
	Reactor Door Scram[d]	1	Reactor Scram

(a) May be bypassed when linear power channels are reading greater than 3×10^{-10} amps.

(b) During steady state operation this safety channel may be bypassed with the permission of the reactor supervisor.

(c) The manual scram shall consist of a regular manual scram at the console and a manual electrical switch which shall disconnect the electrical power of the facility from the reactor causing a loss of power scram.

(d) The reactor door scram may be bypassed during maintenance checks and radiation surveys with the specific permission of the Operations Supervisor provided that no other scram channels are bypassed.

8. The interlocks that shall be operable during reactor operations are listed in Table II.

Bases

A minimum number of three control rods is specified to assure that there is adequate shutdown capability even for the stuck control rod condition. Normally there are more than three control rods available for shutdown capability.

The normal core loading will consist of no more than 45 fuel assemblies seven of which are control rod assemblies with fueled followers. The maximum reactivity worth of any clean fuel assembly will be 3.9% delta k/k and the excess reactivity of the reactor core above cold, clean critical will be less than 3.9% delta k/k.

The maximum reactivity insertion rates, far from the near criticality, are specified to assure that the reactivity addition rate is less than that analyzed in the design basis accident (DBA). The maximum control rod withdrawal rate and the moderator-reflector water addition rate are controlled by these limitations.

The insertion time of less than 900 milliseconds for each control rod from its fully withdrawn position is specified to assure that the insertion time does not exceed that assumed when establishing the minimum period in Specification 2.2 as a limiting safety system setting.

The auxiliary reactor scram is specified to assure that there is a secondary mode of shutdown available during reactor operations. The requirement that negative reactivity be introduced in less than one minute following activation of the scram is established to minimize the consequences of any potential power transients.

The normal moderator-reflector water level of the reactor is established at not greater than 10 inches above the top grid of the core to assure that the moderator-reflector water dump, back up scram will introduce negative reactivity within the time assumed in the safety analysis by loss of reflector at the top of the core.

The safety system channels listed in Table I provide a high degree of redundancy to assure that human or mechanical failures will not endanger the reactor facility or the general public.

The interlock system listed in Table II assures that only authorized personnel can operate the reactor and the proper sequence of operations is performed.

TABLE II

INTERLOCKS

Interlocks	Action if Interlock Not Satisfied
Reactor Console Keys (2) "On"	Reactor Scram
Reactor Period 15 sec[a]	Prevents Control Rod Withdrawal
Neutron Flux 2 cps	Prevents Control Rod Withdrawal
Failure of 400 Cycle Synchro Power Supply	Prevents Control Rod Withdrawal
Failure of Line Voltage to Recorders	Prevents Control Rod Withdrawal
Moderator-Reflector Water Fill On	Prevents Control Rod Withdrawal
Water Level in Reactor Tank 10 ± 1" Above Core Top Grid	Stops Water Fill
Turning the "calibrate" switch on the[a] Log-N Period Amplifier to Other Than the "Operate" position.	Prevents Control Rod Withdrawal

(a) These interlocks are available on only 1 of the 2 Log-N period Amplifiers and, therefore, may be bypassed with the permission of the Operations Supervisor if that one amplifier is out of service.

3.2 Reactor Parameters

Applicability

These specifications apply to core parameters and reactivity coefficients.

Objective

The purpose of these specifications is to assure that the reactor is operated within the range of parameters that have been analyzed.

Specifications

1. Above 90°F the isothermal temperature coefficient of reactivity shall be negative. The net positive reactivity insertion from the minimum operating temperature to the temperature at which the coefficient becomes negative shall be less than 0.08% delta k/k.

2. The void coefficient of reactivity shall be negative, when the moderator temperature is above 90°F, within all standard fuel assemblies and have a minimum average negative value of 0.3×10^{-5} delta k/k/cc within the boundaries of the active fuel region.

3. The minimum operating temperature shall be 50°F.

Bases

The minimum absolute value of the temperature coefficient of reactivity is specified to assure that an adequate inherent negative reactivity effect takes place when the reactor temperature increases above 90°F. Above a moderator temperature of 90°F the minimum average negative value of the void coefficient of reactivity is specified to assure that the negative reactivity insertion due to void formation is greater than that which was calculated to occur in the SAR. The minimum operating temperature of 50°F establishes the temperature range for which the net positive reactivity limit can be applied.

3.3 Radiation Monitoring

Applicability

These specifications apply to the minimum radiation monitoring requirements for reactor operations.

Objective

The purpose of these specifications is to assure that adequate monitoring is available to preclude undetected radiation hazards or uncontrolled releases of radioactive material.

Specifications

1. The minimum complement of radiation monitoring equipment required to be operating for reactor operation shall include:

   a. A criticality detector system which monitors the main fuel storage area and also functions as an area monitor. This system shall have a visible and an audible alarm in the control room.

   b. An area gamma monitoring system which shall have detectors at least in the following locations: (1) Control room, (2) Reactor room near the fuel vault, (3) Reactor room (high level monitor), and (4) Outside the reactor room window.

   c. Instruments to continuously sample and measure the particulate activity in the reactor room atmosphere shall be operating whenever the reactor is to be operated.

   d. The radiation monitors required by 3.3.1 a, b, and c, may be temporarily removed from service if replaced by an equivalent portable unit.

2. Portable detection and survey instruments shall be provided.

Bases

The continuous monitoring of radiation levels in the reactor room and other stations assures the warning of the existence of any abnormally high radiation levels. The availability of instruments to measure the amount of particulate activity in the reactor room air assures continued compliance with the requirements of 10 CFR Part 20. The availability of required portable monitors provide assurance that personnel will be able to monitor potential radiation fields before an area is entered.

In all cases, the low power levels encountered in operation of the critical assembly minimizes the probable existence of high radiation levels.

3.4 Experiments

Applicability

These specifications apply to all experiments placed in the reactor tank.

Objective

The objective of these specifications is to define a set of criteria for experiments to assure the safety of the reactor and personnel.

Specifications

1. No new experiment shall be performed until a written procedure which has been developed to permit good understanding of the safety aspects is reviewed and approved by the Nuclear Safety Review Board and approved by the Facility Supervisor. Experiments that fall in the general category but with minor deviations from those previously performed may be approved directly by the Facility Supervisor.

2. No experiment shall be conducted if the associated experimental equipment could interfere with the control rod functions or with the safety functions of the nuclear instrumentation.

3. For experiments with an absolute worth greater than 0.25% delta k, the maximum reactivity change for withdrawal and insertion shall be 0.15% delta k/k/sec. Moving parts worth less than 0.25% delta k/k may be oscillated at higher frequencies in the core.

4. The maximum positive step insertion of reactivity which can be caused by an experimental accident or experimental equipment failure of a moveable experiment shall not exceed 0.5% delta k/k.

5. Experiments shall not contain a material which may produce a violent chemical reaction and/or significant airborne radioactivity.

6. Experiments containing known explosives or highly flammable materials shall not be installed in the reactor.

7. All experiments which corrode easily and are in contact with the reactor coolant shall be encapsulated within corrosion resistant containers.

8. The radioactive material content of any singly encapsulated experiment shall be limited such that the complete release of all gaseous, particulate, or volatile components directly to the reactor room will not result in exposures in excess of 10% of the equivalent annual exposures stated in 10 CFR 20 for persons remaining in unrestricted areas for two hours or in restricted areas during the length of time required to evacuate the restricted area.

9. The radioactive material content of any doubly encapsulated experiment shall be limited such that the postulated complete release from the encapsulation or confining boundary of the experiment could not result in exposure of any person occupying an unrestricted area continuously for a period of two hours from the time of release in excess of 500 mRem whole body or 1.5 Rem thyroid or an exposure in excess of 5 Rem whole body or 30 Rem thyroid for persons located within the restricted area during the length of time required to evacuate the restricted area.

Bases

The basic experiments to be performed in the reactor programs are described in the Safety Analysis Report (SAR). The present programs are oriented toward reactor operator training, the instruction of students, and with such research and development as is permitted under the terms of the facility license. To assure that all experiments are well planned and evaluated prior to being performed, detailed written procedures for all new experiments must be reviewed by the NSRB and approved by the Facility Supervisor.

Since the control rods enter the core by gravity and are required by other technical specifications to be operable, no equipment should be allowed to interfere with their functions. To assure that specified power limits are not exceeded, the nuclear instrumentation must be capable of accurately monitoring core parameters.

All new reactor experiments are reviewed and approved prior to their performance to assure that the experimental techniques and procedures are safe and proper and that the hazards from possible accidents are minimal. A maximum reactivity change is established for the remote positioning of experimental samples and devices during reactor operations to assure that the reactor controls are readily capable of controlling the reactor.

All experimental apparatus placed in the reactor must be properly secured. In consideration of potential accidents, the reactivity effect of movable apparatus must be limited to the maximum accidental step reactivity insertion analyzed in the SAR.

Restrictions on irradiations of explosives and highly flammable materials are imposed to minimize the possibility of explosion or fires in the vicinity of the reactor.

To minimize the possibility of exposing facility personnel or the public to radioactive materials, no experiment will be performed with materials that could result in a violent chemical reaction, produce airborne activity, or cause a corrosive attack on the fuel cladding or primary coolant system.

Specifications 8 and 9 will assure that the quantities of radioactive materials contained in experiments will be limited such that their failure will not result in exposures to individuals in restricted or unrestricted areas to exceed the maximum allowable annual exposures stated in 10 CFR Part 20.

4.0 SURVEILLANCE REQUIREMENTS

4.1 Reactor Control and Safety

Applicability

These specifications apply to the surveillance of the safety and control apparatus and instrumentation of the facility.

Objective

The purpose of these specifications is to assure that the safety and control equipment is operable and will function as required in Specification 3.1.

Specifications

1. The total control rod drop time and magnet release time shall be measured semiannually to verify that the requirements of Specification 3.1, Item 3, are met.

2. The moderator-reflector water dump time shall be measured semiannually to verify that the requirement of Specification 3.1, Item 4, is met.

3. All instrument channels, including safety system channels, shall be calibrated annually.

4. A channel test of the safety system channels (intermediate, and power range instruments) and a visual inspection of the reactor shall be performed prior to reactor startup. The interlock system shall be checked to satisfy rod drive permit. These systems shall be rechecked following a shutdown in excess of 8 hours.

5. The moderator-reflector water height shall be checked visually before reactor startup to verify that the requirements of Specification 3.1, Item 5, are met.

Bases

Past performance of control rods and control rod drives and the moderator-reflector water fill and dump valve system have demonstrated that testing semiannually is adequate to assure compliance with Specification 3.1, Items 3, 4, and 5.

Visual inspection of the reactor components, including the control rods, prior to operation is to assure that the components have not been damaged and that the core is in the proper condition. Since redundancy of all safety channels is provided, random failures should not jeopardize the ability of the overall system to perform its required functions. The interlock system for the reactor is designed so that its failure places the system in a safe or non-operating condition. However, to assure that failures in the safety channels and interlock system are detected as soon as possible, frequent surveillance is desirable and thus specified. All of the above procedures are enumerated in the daily startup check-list.

Past experience has indicated that, in conjunction with the daily check, calibration of the safety channels annually assures the proper accuracy is maintained.

4.2 Reactor Parameters

Applicability

These specifications apply to the verification of control rod reactivity worths, temperature and void coefficients of reactivity, and reactor power levels which are pertinent to the reactor control.

Objective

The purpose of these specifications is to assure that the analytical bases are and remain valid and that the reactor is safely operated.

Specifications

The following parameters shall be determined during the initial testing of an unknown or previously untested core configuration:

a. control rod bank reactivity worth

b. temperature and void coefficients of reactivity

c. reactor power measurement

d. shutdown margin

Bases

Measurements of the above are parameters made when a new reactor configuration is assembled. Whenever the core configuration is altered considerably to an unknown or untested configuration the core parameters are evaluated to assure that they are within the limits of these specifications and the values analyzed in the SAR. During the initial test period of the reactor, measurements and calculations of core parameters will be for standard assemblies which are to be utilized in the reactor's operational program.

4.3 Radiation Monitoring

Applicability

These specifications apply to the surveillance of the area and air radiation monitoring equipment.

Objective

The purpose of these specifications is to assure the continued validity of radiation protection standards in the facility.

Specification

The criticality detector system, area gamma monitors, and the mobile particulate air monitor shall be checked daily if the reactor is operated, tested monthly, and calibrated semiannually.

Bases

Experience has demonstrated that calibration of the criticality detectors, air gamma monitors, and the mobile air monitoring instrument semiannually is adequate to assure that significant deterioration in accuracy does not occur. Furthermore, the operability of these radiation monitors is included in the daily pre-startup checklist.

5.0 DESIGN FEATURES

5.1 Site

The facility is located on a site situated on the south bank of the Mohawk River in the City of Schenectady. A radius of 30 feet shall define the restricted area for the site and a minimum radius of 50 feet shall define the exclusion area.

5.2 Facility

The facility is housed in the reactor building. The security of the facility is maintained by the use of two fences; one at the site boundary and the other, defining the restricted area, around the reactor building itself.

5.3 Reactor Room

The reactor room is a 12 inch reinforced concrete enclosure with approximate floor dimensions of 40 x 30 feet. The height from the ground floor to the ceiling shall be about 30 feet. The roof is a steel deck covered by 2 inches of lightweight concrete, five plies of felt and asphalt, with a gravel surface. Access to the reactor room is through a sliding fireproof steel door which also contains a smaller personnel door. Near the center of the room is a pit 14-1/2 x 19-1/2 feet and 12 feet deep with a floor of 18 inch concrete. This part contains the 3500 gallon water storage tank and other piping and auxiliary equipment.

5.4 Reactor

5.4.1 Reactor Tank

The stainless steel lined reactor tank has a capacity of approximately 2000 gallons of water. The tank nominal dimensions are 7 feet in diameter and 7 feet high. The tank is supported at floor level above the reactor room by 8 inch steel I beams. There are no side penetrations in the reactor tank.

The reactor tank is connected to the water storage tank via a six inch quick dump line. Therefore, it is required that the storage tank be vented to the atmosphere such that its freeboard volume can always contain all water in the primary system.

5.4.2 Reactor Core

The stainless steel reactor core structure is comprised of upper, center, and lower grid plates. The active core is situated between the upper and center grid plates and is about 22 inches in height and 22 inches in equivalent diameter. The core normally contains 38 stationary fuel assemblies and 7 control rod assemblies with fuel followers. The entire support structure is mounted on four posts set in the floor of the reactor tank.

5.4.3 Standard Fuel Assemblies

A stationary fuel assembly shall be composed of a maximum of 18 fuel plates of stainless steel clad 93% enriched UO_2 - SS cermet. The box-type fuel assemblies are 2.9 x 2.9 x 22 inches in dimensions. The center-to-center spacing of the fuel plates is maintained by grooved polystyrene inserts at 0.163 inch. A control rod fuel follower will be of similar fuel enrichment but limited to a maximum of 16 plates per assembly. For reduced loadings, plates may be omitted or dummy plates used.

5.4.4 Control Rod Assemblies

The control rod assembly shall consist of a control rod absorber section and a control rod follower. The length of the control rod poison section is 22 inches and is nominally 2.619 inches square. The poison and fuel follower are inserted in a stainless steel square tube, 2.75 inches square, which passes through the core and rests in a hydraulic buffer on the bottom grid plate of the support structure.

The drive mechanism is a motor and gear box coupled by a magnetic clutch to a rack and pinion attached to the top of the rod from an overhead cantilever mount.

Seven control rods are provided; removable inserts in the grid plates permit several choices of lattice positions.

5.5 Water Handling System

The water handling system allows remote filling and emptying of the reactor tank. It provides for a water dump by means of a fail safe butterfly type gate valve when a reactor scram is initiated. The filling system shall be controlled by the operator who must satisfy the sequential interlock system before adding water to the tank. A pump is provided to add the moderator-reflector water from the storage dump tank into the reactor tank. Slow and fast fill rates of about 10 gpm and 50 gpm, respectively, are provided. A nominal six-inch valve is installed in the dump line and has the capability of emptying the reactor tank on demand of the operator or when a reactor scram is initiated, unless bypassed with the approval of the licensed Senior Operator on duty. A valve is installed in the bottom drain line of the reactor tank to provide for completely emptying the reactor tank.

5.6 Fuel Storage and Transfer

When not in use, the fuel plates shall be stored within the storage vault located in the reactor room. The vault shall be closed by a locked door and shall be provided with a criticality monitor near the vault door. The fuel shall be stored in cadmium clad steel tubes with no more than 1 Kg fuel per tube mounted on a steel wall rack. The center-to-center spacing of the storage tubes together with the cadmium clad steel tubes assures that the infinite multiplication factor is less than 0.9 when flooded with water.

All fuel transfers shall be conducted under the direction of a licensed senior operator.

Operating personnel shall be familiar with health physics procedures and monitoring techniques and shall monitor the operation with appropriate radiation instrumentation.

For a completely unknown or untested system, fuel loading shall follow the inverse multiplication approach to criticality and, thereafter, meet Specification 4.2. Should any interruption of the loading occur (more than four days), all fuel elements except the initial loading step shall be removed from the core in reverse sequence and the operation repeated.

For a known system, up to a quadrant of elements may be removed from the core or a single stationary fuel assembly be replaced with another stationary assembly only under the following conditions:

1. The net change in reactivity has been previously determined by measurement or calculation to be negative or less than 0.5% delta k/k positive.

2. The reactor is subcritical by at least 2% delta k/k in reactivity.

3. There is initially only one vacant position within the active fuel lattice.

4. The nuclear instrumentation is on scale and the dump valve is not bypassed.

5. The critical rod bank position is checked after the operation is complete.

Tech. Spec.

6.0 Administrative Controls

6.1 Organization

6.1.1 Structure

The organization for the management and operation of the reactor facility shall include the structure indicated in Figure 6.1.

Level 1: The Facility Director is responsible for the facility license and site administration.

Level 2: The Facility Supervisor is responsible for the reactor facility operation and management.

Level 3: Licensed senior operators are responsible for daily reactor operations.

Level 4: Licensed operators are the operating staff.

A health physicist who is organizationally independent of the RPI operations group shall provide advice as required by the RPI Operations Supervisor in matters concerning radiological safety.

6.1.2 Responsibility

The Operations Supervisor of the Rensselaer Polytechnic Institute Critical Experiment Facility shall be responsible for the safe operation of the facility He shall be responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, including these technical specifications.

In all matters pertaining to the operation of the reactor and these technical specifications, the Operations Supervisor shall report to and be directly responsible to the Facility Director.

6.1.3 Staffing

a) The minimal staffing when the reactor is not shutdown as described in these specifications shall be:

   1) An operator or senior operator licensed pursuant to 10CFR55 be present at the controls.

2) A licensed senior operator shall be present or readily available on call.

3) The identity of and method for rapidly contacting the licensed Senior Operator on duty shall be known to the operator.

b) A list of reactor facility personnel by name and telephone number shall be readily available in the control room for use by the operator. The list must include:

1) Management personnel
2) Radiation safety personnel
3) Other operations personnel

c) Events requiring the direction of the Facility Supervisor:

1) All fuel or control rod relocations within the reactor core.
2) Recovery from unplanned or unscheduled shutdown.

6.1.4 Selection and Training of Personnel

The selection, training, and requalification of operations personnel shall meet or exceed the requirements of American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4-1977, Sections 4-6.

Additionally, the minimum requirements for the Operations Supervisor are at least four years of reactor operating experience and possession of a Senior Operator License for the RPI Critical Facility. Years spent in baccalaureate or graduate study may be substituted for operating experience on a one-for-one basis up to a maximum of two years.

6.1.5 Review and Audit

A Nuclear Safety Review Board (NSRB) shall review and audit reactor operations and advise the Facility Director in matters relating to the health and safety of the public and the safety of facility operations.

6.1.5.1 Composition and Qualification

The NSRB shall have at least four members of whom no more than the minority shall be from the line organization shown in Figure 6.1. The board shall be made up of senior personnel who shall collectively provide a broad spectrum of expertise in reactor technology. Qualified and approved alternates may serve in the absence of regular members.

6.1.5.2 Charter and Rules

The Review Board shall function under the following rules:

a) The Chairman of the NSRB shall be approved by the Facility Director.
b) The Board shall meet at least semiannually.
c) The quorum shall consist of not less than a majority of the full Board and shall include the Chairman or his designated alternate.
d) Minutes of each Board meeting shall be distributed to the Director, NSRB members, and such others as the Chairman may designate.

6.1.5.3 Review Function

The following items shall be reviewed:

a) Proposed experiments and tests utilizing the reactor facility which are significantly different from tests and experiments previously performed at the facility.
b) Reportable occurrences.
c) Proposed changes to the Technical Specifications and proposed admendments to facility license.

6.1.5.4 Audit Function

The audit function shall include selective (but comprehensive) examination of operating records, logs, and other documents. Where necessary, discussions with cognizant personnel shall take place. In no case shall the individual immediately responsible for the area audit in the area. The following areas shall be audited:

a) Reactor operations and reactor operational records for compliance with internal rules, regulations, procedures, and with licensed provisions, procedures, and with licensed provisions, including technical specifications.
b) Existing operating procedures for adequacy and to assure that they achieve their intended purpose in light of any changes since their implication.
c) Plant equipment performance with particular attention to operating anomalies, abnormal occurrences, and the steps taken to identify and correct their causes.

6.2 Procedures

Written procedures shall be prepared, reviewed and approved prior to initiating any of the activities listed in this section. The procedures, including applicable check lists, shall be reviewed by the NSRB and followed for the following operations:

a) Startup, operation, and shutdown of the reactor.
b) Installation and removal of fuel elements, control rods, experiments and experimental facilities.
c) Corrective actions to be taken to correct specific and forseen malfunctions such as for power failures, reactor scrams, radiation emergency, responses to alarms, moderator leaks and abnormal reactivity changes.
d) Periodical surveillence of reactor instrumentation and safety systems, area monitors, and continuous air monitors.
e) Implementation of the facility security plan.
f) Implementation of facility emergency plan in accordance with 10CFR50 Appendix E.
g) Maintenance procedures which could have an effect on reactor safety.

Substantive changes to the above procedures shall be made only with the approval of the NSRB. Temporary changes to the procedures that do not change their original intent may be made with the approval of the Operations Supervisor. All such temporary changes to the procedures shall be documented and subsequently reviewed by the Nuclear Safety Review Board.

6.3 Experiment Review and Approval

a) All new experiments or classes of experiments that raise an unreviewed safety question shall be reviewed by the Nuclear Safety Review Board. This review shall assure that compliance to the requirements of the license technical specifications shall be documented.
b) Substantive changes to previously approved experiments shall be made only after review and approval in writing by NSRB. Minor changes that do not significantly alter the experiment may be approved by the Operations Supervisor.

c) Approved experiments shall be carried out in accordance with established approved procedures.

d) Prior to review, an experiment plan or proposal shall be prepared describing the experiment including any safety considerations.

e) Review comments of the NSRB setting forth any conditions and/or limitations shall be documented in committee minutes and submitted to the Facility Supervisor.

6.4 Required Actions

6.4.1 Action to be taken in Case of Safety Limit Violations

a) The reactor shall be shut down, and reactor operations shall not be resumed until authorized by the Nuclear Regulatory Commission.

b) The safety limit violation shall be promptly reported to one authority or designated alternates.

c) The safety limit violation shall be reported to the Nuclear Regulatory Commission in accordance with Section 6.5.3.

d) A safety limit violation report shall be prepared. The report shall describe the following:

(1) Applicable circumstances leading to the violation including, when known, the cause and contributing factors.

(2) Effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public.

(3) Corrective action to be taken to prevent recurrence.

The report shall be reviewed by the NSRB and any follow-up report shall be submitted to the Commission when authorization is sought to resume operation of the reactor.

6.4.2 Action to be Taken in the Event of an Occurrence of the Type Identified in Section

a) Reactor conditions shall be returned to normal or the reactor shall be shutdown. If it is necessary to shutdown the reactor to correct the occurrence, operations shall not be resumed unless authorized by Facility Supervisor or designated alternate.

b) Occurrence shall be reported to Facility Supervisor or designated alternates and to the commission as required.

c) All such conditions including action taken to prevent or reduce the probability of a recurrence shall be reviewed by the NSRB.

6.5 Reports

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the director of the appropriate Regional Office of Inspection and Enforcement unless otherwise noted.

6.5.1 Operating Reports

A report covering the previous year shall be submitted by March 1 of each year. It shall include the following:

a) Operations Summary - A summary of operating experience occurring during the reporting period that relate to the safe operation of the facility including:

1. Changes in facility design,
2. Performance characteristics (e.g., equipment and fuel performance),
3. Changes in operating procedures which relate to the safety of facility operations,
4. Results of surveillance tests and inspections required by these Technical Specifications,
5. A brief summary of those changes, tests, and experiments which required authorization from the Commission pursuant to 10CFR50.59(a),and
6. Changes in the plant operating staff serving in the following positions:

    a) Facility Director
    b) Operations Supervisor
    c) Health Physicist
    d) Nuclear Safety Review Board Members

b) Power Generation - a tabulation of the integrated thermal power during the reporting period.

c) Shutdowns - A listing of unscheduled shutdowns which have occurred during the reporting period, tabulated according to cause, and a brief discussion of the preventive action taken to prevent recurrence.

d) Maintenance - A tabulation of corrective maintenance (excluding preventative maintenance) performed during the reporting period on safety related systems and components.

e) Changes, Tests, and Experiments - A brief description and a summary of the safety evaluation for all changes, tests, and experiments which were carried out without prior Commission approval pursuant to the requirements of 10CFR Part 50.59(b).

f) A summary of the nature, amount and maximum concentrations of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge.

g) Radioactive monitoring - A summary of the TLD dose rates taken at the exclusion area boundary and the site boundary during the reporting period.

h) Occupational Personnel Radiation Exposure - A summary of radiation exposures greater than 500 mRem (50 mRem for persons under 18 years of age) received during the reporting period by facility personnel (faculty, students, or experimenters).

6.5.2 <u>Non-Routine Reports</u>

a) Reportable Operational Occurrence Reports
Notification shall be made within 24 hours by telephone and telegraph to the Director of the appropriate Regional Office followed by a written report within 10 days to the Director of the Regional Office in the event of a reportable operational occurrence as defined in Section 1.0. The written report on these reportable operational occurrences, and to the extent possible, the preliminary telephone and telegraph* notification shall: (a) describe, analyze, and evaluate safety implications, (b) outline the measures taken to assure that the cause of the condition is determined, (c) indicate the corrective action (including any changes made to the procedures and to the quality assurance program) taken to prevent repetition

*Telegraph notification may be sent on the next working day in the event of a reportable operational occurrence during a weekend or holiday period.

of the occurrence and of similar occurrences involving similar components or systems, and (d) evaluate the safety implications of the incident in light of the cumulative experience obtained from the record of previous failures and malfunctions of similar systems and components.

b) Unusual Events

A written report shall be forwarded within 30 days to the Director of the appropriate Regional Office in the event of:

1. Discovery of any substantial errors in the transient or accident analyses or in the methods used for such analyses, as described in the Safety Analysis Report or in the bases for the Technical Specifications.

6.6 Operating Records

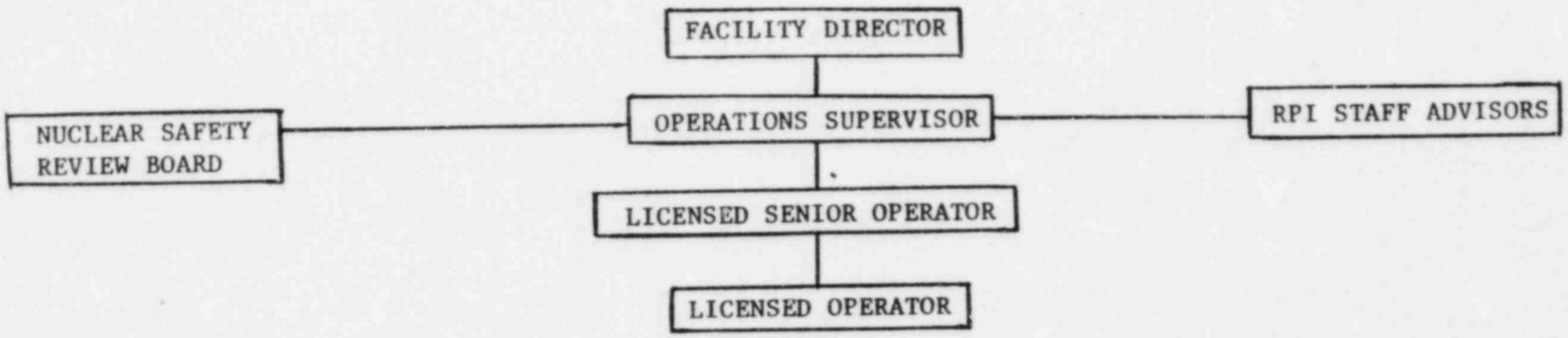
6.6.1 The following records and logs shall be maintained at the Facility or at Rensselaer for at least five years.

a. Normal facility operation and maintenance.
b. Reportable operational occurrences.
c. Tests, checks, and measurements documenting compliance with surveillance requirements.
d. Records of experiments performed.
e. Records of radioactive shipments.

6.6.2 The following records and logs shall be maintained at the Facility or at Rensselaer for the life of the Facility.

a. Gaseous and liquid radioactive releases from the facility.
b. TLD environmental monitoring surveys.
c. Radiation exposures for all RPI Critical Facility personnel (students, experimenters).
d. Fuel inventories, offsite transfers and inhouse transfers if they are not returned to their original core or vault location during the experimental program in which the original transfer was made.
e. Facility radiation and contamination surveys.
f. The present as-built facility drawings and new updated or corrected versions.
g. Minutes of Nuclear Safety Review Board meetings.

FIGURE 6.1


FACILITY DIRECTOR
NUCLEAR SAFETY REVIEW BOARD
OPERATIONS SUPERVISOR
RPI STAFF ADVISORS
LICENSED SENIOR OPERATOR
LICENSED OPERATOR