NUREG-0877

Final Environmental Statement

related to license renewal
and power increase for the
National Bureau of Standards Reactor

Docket No. 50-184

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

August 1982

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ABSTRACT

This Final Environmental Statement contains an assessment of the environmental impact associated with renewal of Operating License No. TR-5 for the National Bureau of Standards (NBS) reactor for a period of 20 years at a power level of 20 MW. This reactor is located on the 576-acre NBS site near Gaithersburg in Montgomery County, Maryland, about 20 mi northwest of the center of Washington, D.C. The reactor is a high-flux heavy-water-moderated, cooled and reflected test reactor, which first went critical on December 7, 1967. Though the reactor was originally designed for 20-MW operation, it has been operating for 14 years at a maximum authorized power level of 10 MW. Program demand is now great enough to warrant operation at a power level of 20 MW. No additional major changes to the physical plant are required to operate at 20 MW.

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SUMMARY AND CONCLUSIONS

This Final Environmental Statement was prepared by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission (NRC) (the staff).

(1) This action is administrative.

(2) The proposed action is the renewal of Facility License No. TR-5 for operation of the National Bureau of Standards (NBS) Reactor (Docket Number 50-184), located in Gaithersburg, Maryland, for a period of 20 years at a maximum power level of 20 MW.

(3) The environmental impacts and adverse effects can be summarized as follows:

(a) Gaseous radioactive waste, including approximately 1400 Ci ^{41}Ar and 900 Ci of tritium, will be released annually as a result of station operation.

(b) The annual liquid radioactive waste to the sanitary sewer system will include approximately 3 Ci of tritium and less than 4 mCi of β-γ.

(c) A very low probability of accidental radiation exposure to the public will be created.

(d) Approximately 350 mCi of low-level waste will be generated each year for shipment and disposal off site.

(e) Approximately 2.5 Ci of high-level waste from reactor experimental and maintenance activities, such as replacement of filters and resins, will be generated each year for shipment and disposal off site.

(f) Approximately 18 Ci of high-level waste will be generated each year as a result of fuel cutting operations for shipment and disposal off site.

(g) Each year, a maximum of 48 spent fuel elements will be shipped off site for reprocessing. Either two or four shipments will be required, depending on the capacity of the spent fuel shipping cask selected.

(h) A volume of water equal to 100,000 gpd (4×10^7 gal/yr) will be necessary for makeup to the secondary cooling system to replace water lost through evaporation and blowdown.

(i) Blowdown of 22 gpm (32,000 gpd) from the cooling tower basin to the sanitary sewer system will result in an annual discharge of approximately 100 lb of zinc, which is used for corrosion control in the secondary cooling system.

(j) An estimated 23 tons of minerals and salts and other dissolved solids will be deposited annually on the NBS grounds and immediate vicinity as a result of cooling tower drift. Approximately 95% of these are already in the water supplied by the WSSC.

(k) 5.4×10^6 kWh of electricity will be used annually.

(l) 6400 g of ^{235}U will be used annually.

(4) The principal alternatives considered were to

(a) Refuse to renew license; that is, allow it to expire June 30, 1985.

(b) Refuse to increase the authorized power level; that is, renew license at 10 MW.

(5) The following Federal, state, and local agencies were asked to comment on the Draft Environmental Statement:

Advisory Council on Historic Preservation
Argonne National Laboratory
Army Corps of Engineers
Brookhaven National Laboratory
County Executive, Montgomery County, MD
Department of Agriculture
Department of Commerce
Department of Energy
Department of Health and Human Services
Department of Housing and Urban Development
Department of the Interior
Department of Transportation
Environmental Protection Agency
Federal Energy Regulatory Commission
Maryland Department of State Planning
Mayor, City of Gaithersburg, MD
Metropolitan Washington Council of Governments
National Capital Planning Commission
Oak Ridge National Laboratory

Comments on the Draft Environmental Statement were received from the following persons and organizations:

Department of Agriculture
Department of Health and Human Services
Department of Housing and Urban Development
Department of the Interior
Environmental Protection Agency
Lochstet, William A. - University Park, PA
Maryland Historical Trust

(6) On the basis of the evaluation and analysis set forth in this statement, and after weighing the environmental, economic, technical, and other benefits against environmental costs and considering available alternatives, the staff concludes that the action called for is issuance of an operating license for a period of 20 years at a maximum power level of 20 MW.

FOREWORD

This environmental statement was prepared by the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Reactor Regulation (staff) in accordance with the Commission's regulations set forth in Title 10 of the Code of Federal Regulations Part 51 (10 CFR 51), which implement the requirements of the National Environmental Policy Act of 1969 (NEPA).

The NEPA states, among other things, that it is the continuing responsibility of the Federal government to use all practicable means, consistent with other essential considerations of national policy, to improve and coordinate Federal plans, functions, programs, and resources to the end that the Nation may

> Fulfill the responsibilities of each generation as trustees of the environment for succeeding generations.
>
> Ensure for all Americans safe, healthful, productive, and aesthetically and culturally pleasing surroundings.
>
> Attain the widest range of beneficial uses of the environment without degradation, risk to health or safety, or other undesirable and unintended consequences.
>
> Preserve important historic, cultural, and natural aspects of our national heritage, and maintain, wherever possible, an environment that supports diversity and variety of individual choice.
>
> Achieve a balance between population and resource use which will permit high standards of living and a wide sharing of life's amenities.
>
> Enhance the quality of renewable resources and approach the maximum attainable recycling of depletable resources.

Further, with respect to major Federal actions significantly affecting the quality of human environment, Section 102(2)(C) of the NEPA calls for preparation of a detailed statement on

> The environmental impact of the proposed action
>
> Any adverse environmental effects which cannot be avoided should the proposal be implemented
>
> Alternatives to the proposed action
>
> The relationship between local short-term use of man's environment and the maintenance and enhancement of long-term productivity
>
> Any irreversible and irretrievable commitments of resources which would be involved in the proposed action should it be implemented

An environmental report accompanied the National Bureau of Standards (NBS) application for renewal of Facility License No. TR-5 for a period of 20 years at a maximum power level of 20 MW. A "Notice of Availability of Applicant's Environmental Report and Notice of Intent to Publish an Environmental Impact Statement" (45 FR 85235) was published in the Federal Register. In accordance with the guidelines issued by the Council on Evironmental Quality, an open scoping meeting was held (March 20, 1981) to determine the scope of the issues to be addressed in the environmental impact statement and to identify the significant issues related to the proposed action.

The staff has met with the applicant to discuss items of information in the environmental report, to seek additional information from the applicant that might be needed for an adequate assessment, and generally to ensure that the staff has a thorough understanding of the proposed project. In addition, the staff has obtained information from other sources that assists in the evaluation and has visited the project site and surrounding vicinity. Members of the staff have met with state and local officials who are charged with protecting state and local interests. The principal staff persons who contributed to the review are

J. E. Fairobent	Accident Evaluation Branch
M. H. Fliegel	Hydrology and Geotechnical Engineering Branch
A. K. Ibrahim	Geosciences Branch
W. Pasedag	Accident Evaluation Branch
W. J. Pasciak	Radiological Assessment Branch
W. E. Rodak	Siting Analysis Branch
J. H. Wilson	Standardization and Special Projects Branch
M. L. Wohl	Accident Evaluation Branch

Los Alamos National Laboratory staff, under contract to NRC, have concurred in the selection of accidents evaluated herein that could lead to radiation exposure to the public and are evaluating the effect of these accidents on reactor safety. On the basis of all the foregoing and other such activities or inquiries as are deemed useful and appropriate, the staff made an independent assessment of the various impacts of the proposed project on the NBS site.

This evaluation led to the publication in January 1982 of the Draft Environmental Statement (DES) (NUREG-0877), prepared by the Office of Nuclear Regulation, which was circulated to Federal, state, and local governmental agencies for comment. Notices (47 FR 8273 and 47 FR 8402) were published in the Federal Register concerning the availability of the DES, and interested persons were invited to comment on it.

After receipt and consideration of comments on the DES, the staff prepared this Final Environmental Statement, which includes a discussion of questions and concern raised by the comments and their disposition.

Single copies of this Final Environmental Statement may be obtained as indicated on the inside front cover. James H. Wilson is the NRC Project Manager for this project. Should there be any questions regarding the content of this statement, Mr. Wilson may be contacted by calling (301) 492-9797 or by writing to

Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

1 INTRODUCTION

The action being considered is renewal of the operating license for the National Bureau of Standards (NBS) (applicant) reactor (Docket No. 50-184) for a period of 20 years at a maximum power level of 20 megawatts thermal. This reactor is located on the NBS 576-acre site near the City of Gaithersburg in Montgomery County, Maryland, about 20 mi northwest of the center of Washington, D.C. The reactor is a high-flux heavy-water-moderated, cooled, and reflected test reactor. It first went critical on December 7, 1967 after receiving a provisional operating license. Within 18 months, the reactor was operating routinely at 10 MW. Facility Operating License No. TR-5, authorizing operation at a maximum power level of 10 MW for a period of 15 years, was issued on June 30, 1970.

Through 1981, the reactor has accumulated about 740,000 MWh of operation. The highly automated experimental facilities associated with the reactor allow it to be operated and utilized 24 hours a day, 7 days a week. Routine shutdowns are scheduled every 6 weeks for partial refueling, with additional shutdowns scheduled during the summer and at Christmas time to accommodate staff vacations and for maintenance and testing. The reactor is normally on line between 70 and 75% of the time.

The NBS reactor is used for a wide variety of programs including materials research by neutron scattering, trace analysis by neutron activation analysis, neutron radiography, neutron flux standardization, neutron dosimetry using filtered beams in the keV region, radiation effects, and isotope production. The reactor has 11 radial-beam tubes, 2 through-beam tubes, a thermal column, 4 built-in pneumatic tubes, and provisions for up to 10 incore thimbles and 7 reflector thimbles. Twenty-five major experimental instruments, most of which can be used simultaneously, are installed at these reactor facilities. The facilities are used by more than 200 scientists and technicians from 18 NBS divisions and offices, from 18 other government agencies, and from 25 universities and industrial organizations. Even with the high degree of automation, which allows around-the-clock utilization, there is a 2- to 3-month waiting list for most of the experimental facilities.

The reactor was originally designed for 20-MW operation, but operated initially at 10 MW until program demand and operating experience were sufficient to justify full-power operation. The only parts of the reactor system not originally constructed for full 20-MW operation were a few elements of the process system. During the past 10 years, the process system has been upgraded as the components had to be replaced. The only significant modifications to the facility not described in the original Safety Analysis Report (SAR) (NBSR 9) submitted by the applicant were (1) the replacement of the original aluminum heat exchanger with two stainless-steel heat exchangers and (2) the replacement of the original cooling tower with a more efficient one and the associated modification of the secondary system piping. Consequently, no additional major changes to the physical plant are required to operate at 20 MW.

The material regarding the NBS reactor presented in this report has been taken from National Bureau of Standards Report (NBSR 9); the NBS Reactor Environmental

Report (NBSR 12); the Geology, Seismology, Hydrology Report on the NBS reactor site prepared by the NBS Geotechnical Engineering Group; and other material submitted by NBS personnel.

2 THE SITE AND ENVIRONS

2.1 Geography

The NBS site is a 576-acre tract of land in upper Montgomery County, Maryland, approximately 1 mi southwest of Gaithersburg, Maryland. The site is shown in Figure 2.1. The general area is a combination of residential and rural; the nearest population centers are Gaithersburg and Rockville, 5 mi southeast of the site. The site is located approximately 20 mi northwest of the center of the District of Columbia, in the Maryland Piedmont.

2.2 Topography and Surface Drainage

The topography in the vicinity of the reactor site is undulating, and the relief is moderate (Figure 2.2). Altitudes range from 300 ft above mean sea level (MSL) in the valley of Muddy Branch to 520 ft at Gaithersburg. On the site itself, the range of altitudes is from 365 ft to 465 ft above MSL. The reactor is located at an altitude of approximately 420 ft.

The site is generally in the Potomac River watershed. Drainage is to the south and to the west. Drainage to the south is by Muddy Branch and to the west by Long Draught Branch of Seneca Creek. Both streams are tributaries of the Potomac River. Muddy Branch, the easternmost of the tributaries, enters the Potomac near Katie Island, at a point about 5.5 mi above Lock 20 on the C&O Canal at Great Falls where the uppermost intake for the District of Columbia water supply system is located.

2.3 Geology

The site lies within the Piedmont physiographic province. The section of the Piedmont in which the site is located is underlain by gneiss and schist of the Wissahickon Formation of Precambrian to early Cambrian age (600 million years before present (mybp) to 550 mybp).

Based on the tectonic province concept set forth in Appendix A to Title 10 of the Code of Federal Regulations Part 100 (10 CFR 100) (U.S. General Services Administration, 1981) the site is within the Piedmont-New England tectonic province. There is no known major fault in the site vicinity. There is no known relationship between mapped faults and the moderate seismicity in the region; nor has a capable fault been identified in the eastern United States.

At the site, relatively sound rock was mapped by borings at depths ranging from 35 to 74 ft below ground surface. The surface material grades upward from sound rock to badly weathered rock then to saprolite. The saprolite grades from a fine silty sand with weathered rock fragments to silty clay in the upper 5 to 10 ft. The applicant reports that below a depth of about 10 ft beneath the ground surface, standard penetration counts range from 150 to 200 blows per ft, which indicates a very dense competent soil.

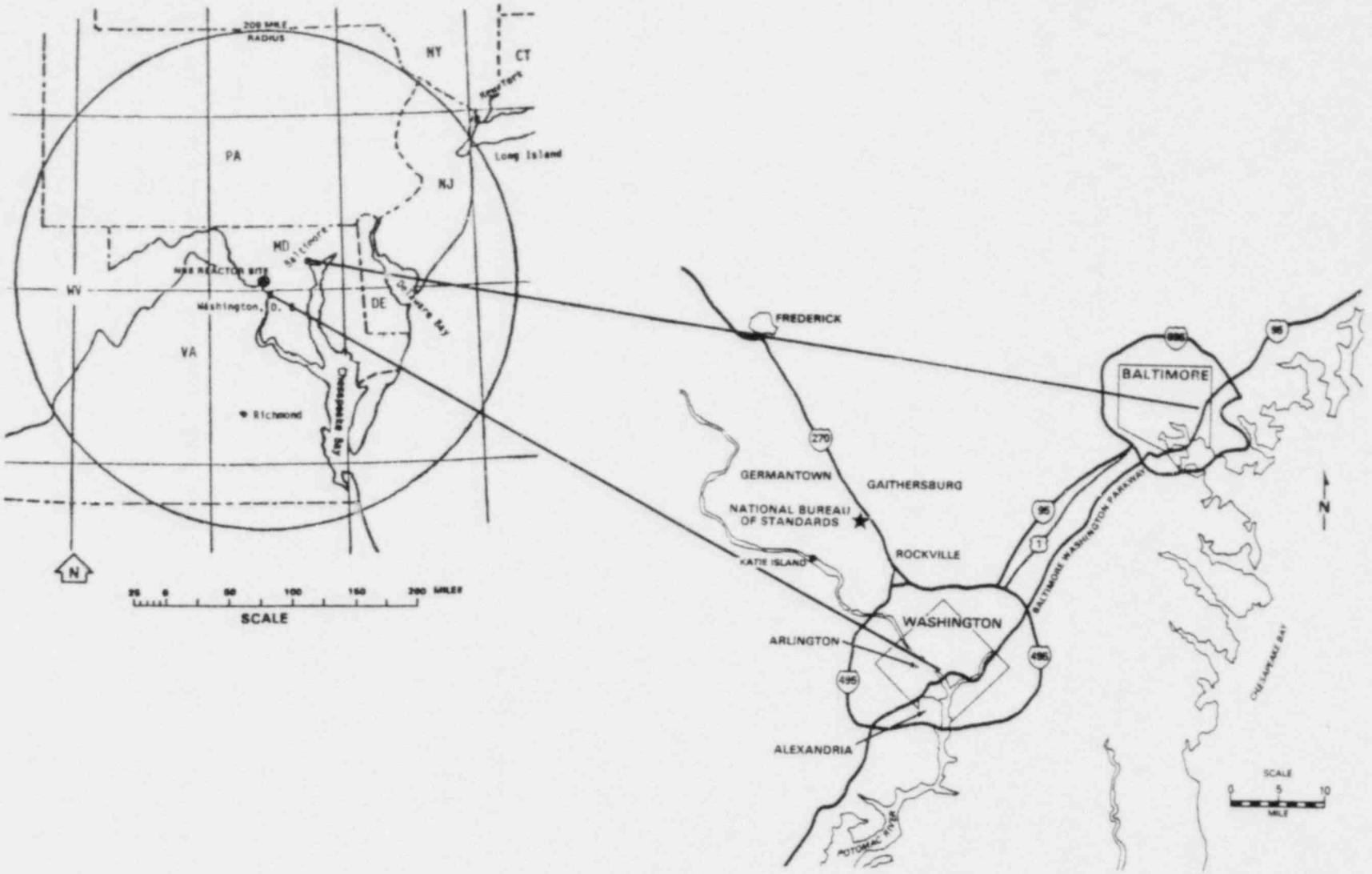


Figure 2.1 NBS regional site map

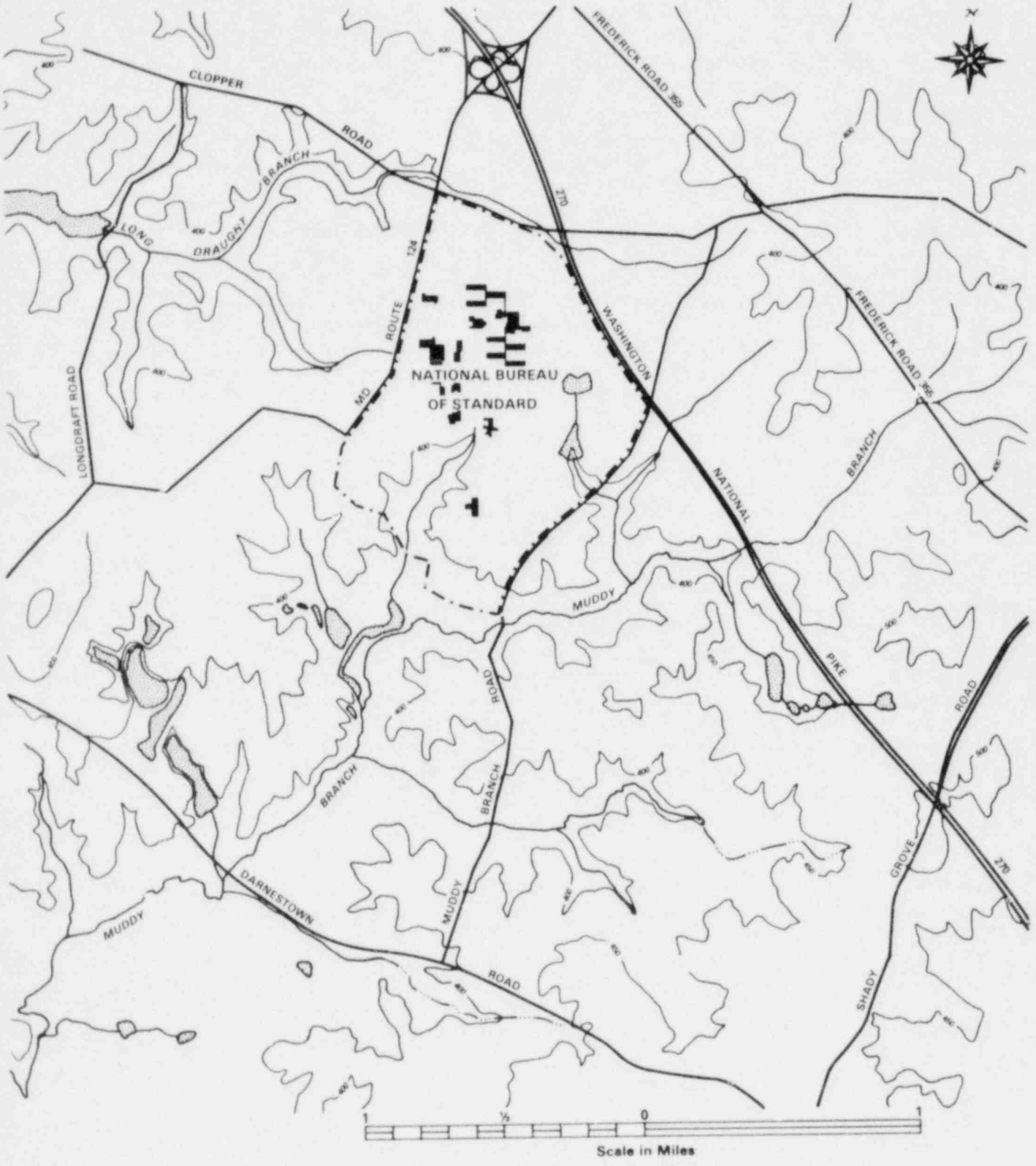


Figure 2.2 Contour map of NBS site and surrounding area

The applicant also reports that jointing in the upper portion of unweathered bedrock varies in spacing from a few inches to 2 or 3 ft. Open fractures decrease in frequency in their depth below the top of sound bedrock. Although no faults have been mapped in site bedrock, it is likely that minor faults exist because of the age of the rocks and the tectonic history of the region. However, these faults, if they do exist, are not capable in the context of Appendix A to 10 CFR 100. The applicant's findings are consistent with other evaluations of nuclear facilities located in the Piedmont tectonic province.

2.4 Seismology

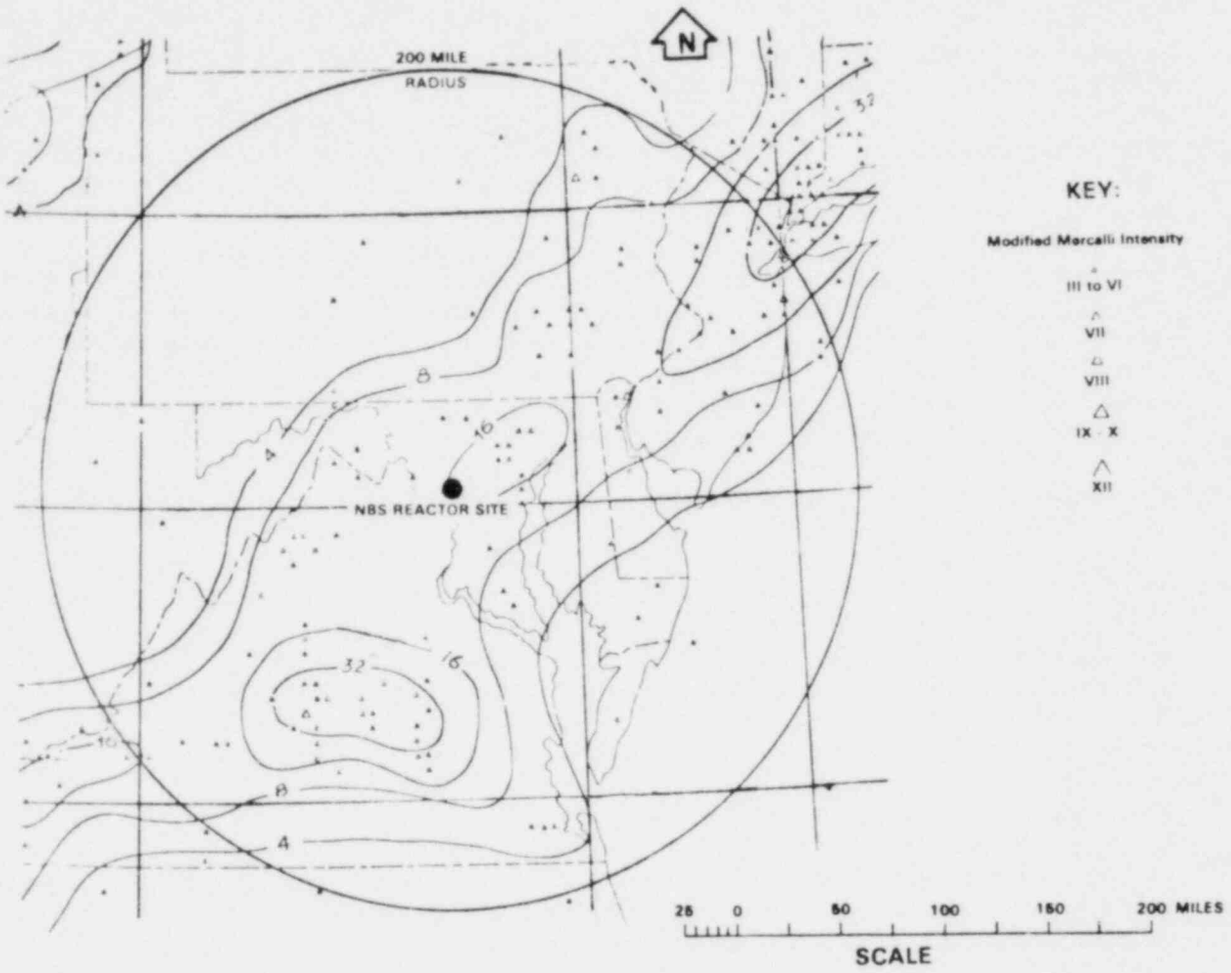
The applicant presents a geological history of the different structures and tries to correlate these structures with the trend of seismicity in the area. It was found that the level of seismicity is more uniform in the Southern Appalachian fold belt, but nowhere is it well-enough defined to indicate individually active structures. Figure 2.3 shows that most of the seismic activity is concentrated in an east-west trend in Central Virginia. The largest event in this cluster occurred near Richmond in 1875 and was a modified Mercalli intensity (MMI) VII event (Bollinger and Hopper, 1971).

For the region around the site, as for most of the eastern United States, more information is needed before a relationship between seismic activity and geologic structure can be determined. Therefore, the tectonic province approach was used. Based on a map by Hadley and Devine (1974), the applicant presented a seismotectonic map with boundaries that conform to his interpretation of the geological and seismological information available. The applicant divided the eastern United States into three tectonic provinces: Central Stable, Appalachian, and Coastal Plain. According to the applicant's method, the NBS reactor is located within the Piedmont belt of the Appalachian tectonic province. If the entire Appalachian Mountain region is considered to be a single tectonic province, then the effects of the 1897 Giles County, Virginia, earthquake of MMI VIII must be considered at the site. The applicant did not do this. However, based on past nuclear licensing reviews in the eastern United States, the staff considers the applicant's Piedmont belt to be a tectonic province by itself (referred to by the staff as the Piedmont-New England tectonic province). Thus, using a tectonic model, the staff agrees with the applicant's conclusion that the "...site is located in an area which has experienced only a minor amount of earthquake activity."

This is consistent with an earlier communication from the U.S. Coast and Geodetic Survey (Tison, 1967) in which they estimate that the maximum potential earthquake for the area would result in a maximum ground acceleration of 0.07 g at the NBS reactor site. Consequently, to be conservative, the confinement building and the reactor equipment within the building were analyzed (U.S. AEC, 1967 and NBSR 9B, 1966) for stresses developed by 0.1 g earthquake loadings and shown to remain intact and maintain their capability.

2.5 Hydrology

The source of the groundwater in the vicinity of the reactor site--and elsewhere in the Maryland Piedmont--is local precipitation, which averages about 40 in. per year. The precipitation is rather evenly distributed throughout the year



NOTES

1. This map has been abstracted from Hadley, J.B., and Devine, J.F. (1974). B. Earthquake Epicenters, 1800-1972 Seismotectonic Map of the Eastern United States. United States Geological Survey, Map MF-620, Sheet 2 of 3.

2. The center of each triangular symbol indicates the epicentral location of one or more seismic events, plotted to the nearest 0.1 degree of latitude and longitude. The intensity shown is maximum Modified Mercalli (MM) intensity in the epicentral area of the largest event at the plotted location. Most locations are based on observations of intensity rather than on instrumental records.

—8—

Seismic frequency contours represent the areal distribution of earthquake epicenters with epicentral intensity of MM III and greater, as indicated by the total number per 10⁴km² during the period 1800-1972. Contour intervals are 0-4, more than 4 but less than 8, more than 8 but less than 16, more than 16 but less than 32, more than 32 but less than 64, and more than 64. The contours are considerably generalized and are shown only as a guide for estimating regional seismicity. They have no value for precise location of seismic boundaries.

Figure 2.3 Regional map of earthquake epicenters

with the wettest month being July (about 4.5 in.) and the driest month October (about 2.8 in.). A zone of saturation is maintained in the subsoils by the precipitation that neither runs off directly nor evaporates. Generally the upper surface of the zone of saturation, or water table, is a subdued replica of the topography of the land surface. Hydraulic gradients that exist in this zone result in the general movement of groundwater to the streams. The rate of movement of water is variable; in the subsoil or saprolite zone the rate may be on the order of 0.1 to 1.0 ft per day.

The major use of groundwater within a 1-mi radius of the reactor site is for domestic and farm purposes. Five wells, located southwest of the center of Gaithersburg, are public-supply wells owned by the Washington Suburban Sanitary Commission; these wells formerly supplied the town with water but are not now used. Water for the town comes from surface water from Sanitary Commission reservoirs and is piped to NBS and the community.

Water-table contours based on measurement of the water level made January 20, 1961 in foundation borings in the vicinity of the reactor building showed a pronounced difference between the northwestward gradient and the southwestward gradient. This difference appears to be related to structural features of the rock, because the schistosity has a northeasterly trend in rock outcrops west and southwest of the reactor site. Groundwater flow parallel to the schistosity would meet with less resistance than that perpendicular to the schistosity. Thus it can be inferred with relative confidence that beneath the reactor building the groundwater flows in a generally southwestward direction.

Although a stream west and northwest of the reactor site is 500 to 600 ft nearer than one to the southwest, it seems unlikely that the path of easiest movement of groundwater would be directly across the schistosity. Ultimately, however, groundwater does move to a stream west of the site, because it is the principal drainage for the reactor site area.

On the basis of available information, the staff concludes that a liquid spilled or leaked at the site that entered the ground probably would move in a southwesterly direction toward the nearby stream at a velocity on the order of 1 ft per day or less. The hazard to nearby groundwater supplies, as currently developed, is small to negligible. Under certain conditions, such as frozen ground, a liquid spill could flow overland to the tributaries to the Potomac River. Depending on stream conditions, total time of travel would range from a few hours to nearly a day.

2.6 Meteorology

The climate of the region within which the NBS test reactor is located is continental, modified somewhat by the Appalachian Mountains to the west and the Atlantic Ocean to the east. The mean annual temperature in the area is about 55°F, ranging from about 32°F in January to about 77°F in July. Extreme temperatures of -15°F and 106°F have been reported in the area. Annual precipitation is about 40 in., well distributed throughout the year. The lowest monthly precipitation usually occurs in February and October (about 2.5 in.); July and August are usually the wettest months (between 4.0 and 4.5 in. of precipitation). A maximum monthly precipitation of 18.2 in. was reported at Dulles International Airport in June 1972. The maximum 24-hour rainfall of 11.9 in. was also reported at Dulles in June 1972 during the

passage of Hurricane Agnes. Annual snowfall at the NBS site is about 20 in., with the maximum monthly snowfall (5 to 7 in.) expected in February. Maximum monthly snowfall in the area was reported to be 35.2 in. in February 1899. The maximum 24-hour snowfall was reported to be 25 in. in January 1922.

About 40 thunderstorms can be expected each year, most frequently from May through August. Hail often accompanies severe thunderstorms. Hail with diameters 3/4 in. or larger occurs about once each year. During the period 1955-1967, 28 tornadoes were reported in a 2-degree latitude-longitude square containing the site. The computed recurrence interval for a tornado at the NBS site is about 2000 years. This recurrence interval is comparable to that calculated using reports of tornadoes for the period 1950-1976. July is the month with the highest frequency of tornado occurrences. Numerous tropical cyclones, storms, and hurricanes have affected the area. In the period 1871-1978, about 20 tropical cyclones, storms, and hurricanes have passed within 100 mi of the NBS site.

The average wind speed in the area is about 8 mph. The "fastest mile" wind speed reported at Washington National Airport was 76 mph, associated with the passage of Hurricane Hazel in October 1954. Figure 2.4 is a representative annual wind rose for the NBS site. Winds from the northwest quadrant (west-northwest, northwest, and north-northwest) occur about 30% of the time and winds from the south and south-southwest occur about 20% of the time.

Some meteorological measurements have been made at the NBS site. Wind speed and direction are currently measured at the top of a small tower located on the southwest corner of the roof of the building about 15 ft above the top of the building. Comparison of wind data at the NBS site with data collected at Washington National Airport showed reasonable agreement. There appear to be no unusual atmospheric dispersion characteristics associated with the NBS site.

2.7 Demographic and Socioeconomic Considerations

The NBS campus is a 576-acre site bounded on the east by a major interstate highway (I-270), on the north and west by Maryland Route 124, and on the southwest by Muddy Branch Road. Figure 2.5 shows the plan of the campus. A 1/4-mi circle is drawn around the reactor. The area adjacent to the reactor building is occupied by a parking lot, the reactor cooling tower, and roads. Thus, the area within a 500-ft radius of the reactor building stack is not readily available for the construction of new buildings, and planning for future development of the NBS site does not include any new buildings within 500 ft of the reactor stack.

If the Bureau should, at some future date, consider constructing a new building to be occupied on a regular basis within 500 ft of the reactor stack before construction would be approved, an analysis would be performed to determine that any radiation exposure to occupants of the building would meet the regulations applicable at that time.

Figure 2.6 shows a 1-mi and a 5-mi circle drawn around the reactor to aid the readers in estimating distances to various locations.

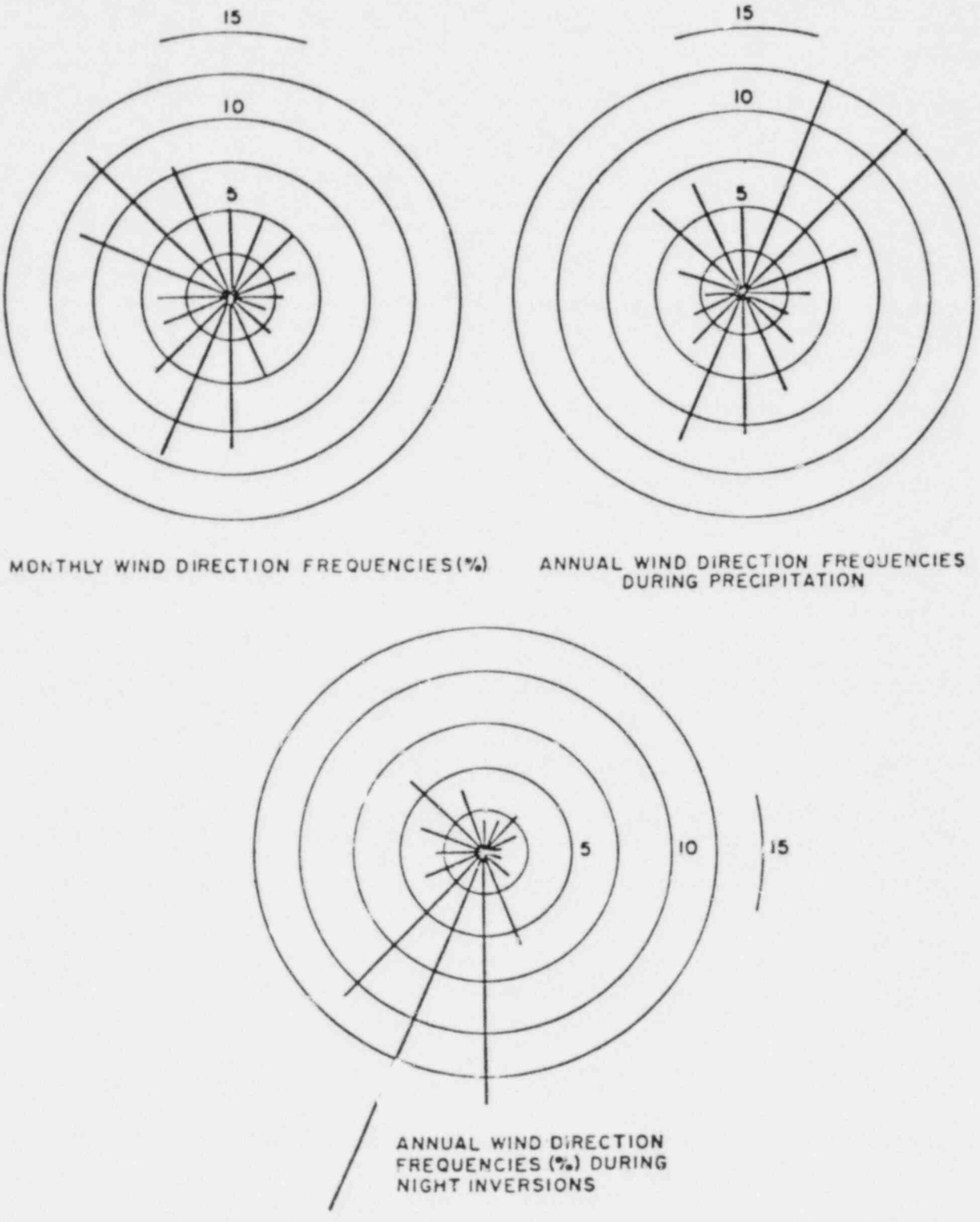


Figure 2.4 Annual wind rose

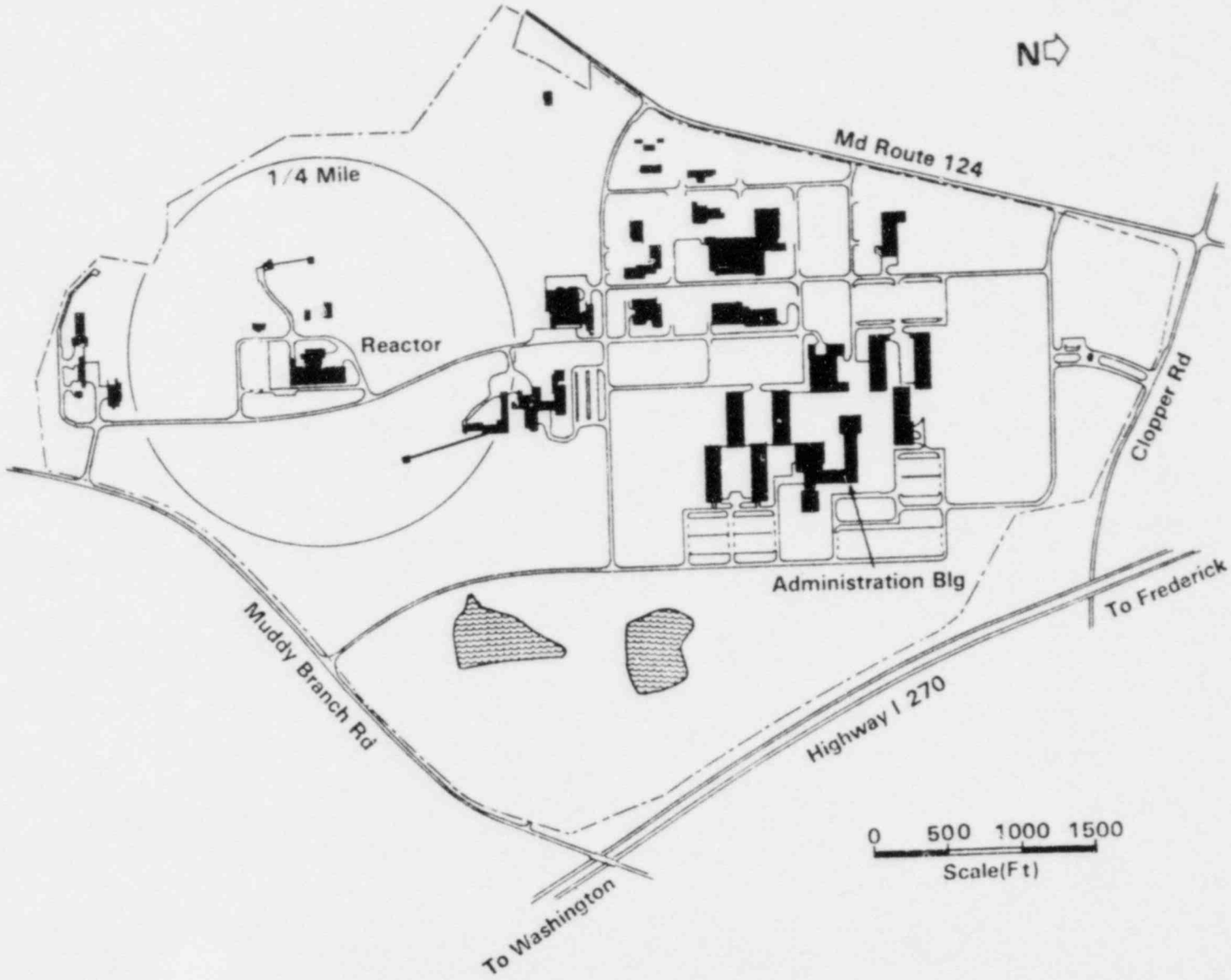


Figure 2.5 Plan of NBS site

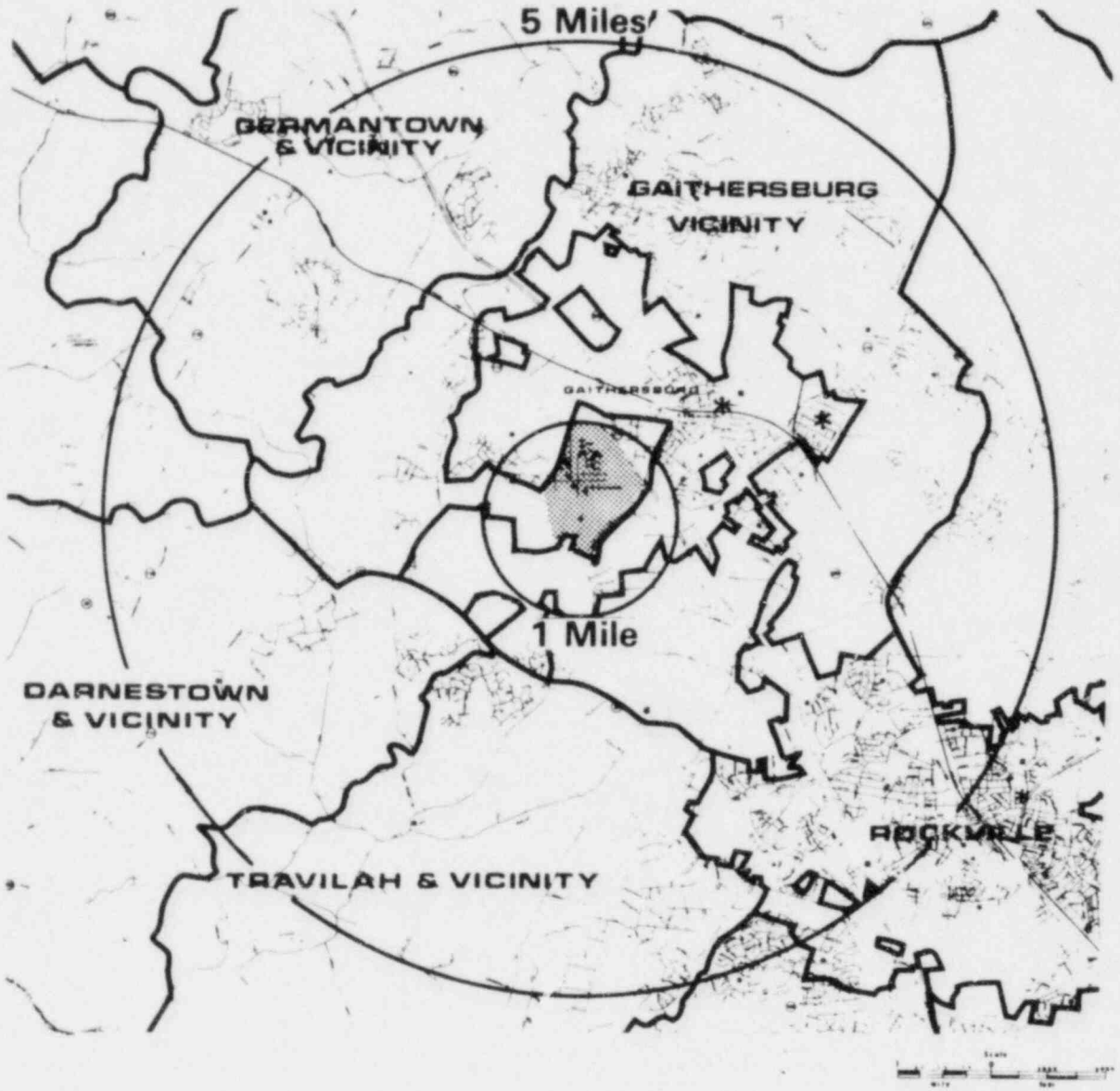


Figure 2.6 Location of NBS reactor in relation to surrounding area

The site boundary nearest to the NBS reactor is approximately 1/4 mi southwest of the reactor. The nearest offsite residential or commercial housing currently is about 1500 ft or 0.3 mi to the southeast of the reactor. Entrances to the NBS site are limited, and access to the reactor facility itself is subject to further security control. The area around the NBS site and reactor is a mixture of light industry and suburban development within gently rolling country side. The eastern site boundary, I-270, is a major commuter highway for Washington, D.C. Along this road are the offices of many Federal contractors and the divisions of several Federal agencies. The households within the 5-mi radius surrounding the site include many Federal employees and support or contracted staff. The NBS staff contributes only a very small share to the dynamics of the local economy.

The daytime population of the NBS site is about 3000, all of whom are under the control of NBS. The current and projected populations for the surrounding area out to 10 mi are shown in Table 2.1. The data are divided into 16 sectors of 22.5 degrees each, with the center of the first sector due north. The population is shown as a function of distance from the reactor and is projected through the year 2000.

The rapid population growth that has taken place in Montgomery County was anticipated when the NBS reactor was first licensed. This can be seen from Table 2.2, which compares the projections given in the applicant's original SAR (NBSR 9) with 1980 population figures. The figures in the SAR were conservative in that they projected about 50% greater population within a 3-mi radius than actually exists. The projected and actual figures for population within a 10-mi radius, however, are very close. These figures show that although the population of Montgomery County has grown rapidly, the growth was anticipated at the time that the reactor was approved for siting at its present location.

RADIUS (MI)	YEAR	N 1	NNE 2	NE 3	ENE 4	E 5	ESE 6	SE 7	SSE 8	S 9	SSW 10	SW 11	WSW 12	W 13	WNW 14	NW 15	NNW 16	TOTAL
0-1	1980	-	100	200	200	300	200	100	500	100	-	-	-	100	100	600	400	2900
	1985	-	200	300	400	600	600	500	800	100	200	300	300	200	100	700	400	5700
	1990	-	200	300	400	600	600	600	900	200	200	400	400	200	100	800	500	6400
	1995	-	300	300	400	600	700	700	1000	300	300	500	500	300	200	800	500	7400
	2000	-	300	300	500	700	800	800	1000	400	400	500	600	400	200	800	500	8200
1-2	1980	-	1200	3500	2600	1500	300	300	300	300	800	300	100	400	1200	1400	300	14500
	1985	300	1800	3700	3000	1700	600	1100	900	300	1000	300	300	500	1500	1600	800	19400
	1990	300	2300	3700	3000	2300	700	1300	1100	400	1500	300	500	1200	2000	1700	900	23000
	1995	500	2400	3700	3100	1300	900	1400	1200	400	2000	300	600	2000	2200	1800	1000	25800
	2000	700	2600	3800	3200	2500	1000	1500	1300	400	2500	400	700	2700	2500	1800	1100	28700
2-3	1980	500	800	2000	3400	1800	1500	400	400	200	900	1100	400	800	400	1000	900	16500
	1985	1100	800	1800	3400	1000	2200	900	700	500	1000	800	600	900	500	1200	1000	18400
	1990	1100	800	1800	3500	1100	2300	1000	800	500	1000	800	600	1000	900	1400	1200	19800
	1995	1300	900	1800	3500	1300	2400	1000	800	600	1100	900	600	1100	1100	1600	1500	21500
	2000	1500	1000	2000	3500	1500	2500	1000	900	700	1200	900	700	1200	1100	1900	1800	23300
3-4	1980	11100	4700	600	2100	1600	2900	6000	3200	800	400	200	700	400	400	1700	1800	38600
	1985	12600	5100	1200	2100	1700	2900	6000	3200	1100	700	400	900	500	700	2700	3100	44900
	1990	14300	5300	1300	2200	1800	3000	6100	6500	1300	900	500	1000	600	1100	3500	4000	53400
	1995	14700	5400	1400	2300	1800	3000	6100	6600	1500	1100	600	1100	700	1600	3600	4700	56200
	2000	15500	5400	1500	2300	1800	3100	6100	6600	1700	1100	800	1200	800	2200	4900	5800	60500
4-5	1980	7200	600	200	1900	1200	2000	9600	3800	1700	300	300	300	200	400	900	1200	37200
	1985	8500	7300	300	1900	1300	2100	9700	3800	1800	500	600	500	200	700	1700	2100	43000
	1990	11400	7500	400	2100	1600	2300	9900	3900	2000	800	700	600	300	1500	3300	2500	50800
	1995	12000	7600	500	2300	1700	2400	10000	4100	2100	1000	800	700	400	1800	4600	3400	55400
	2000	13000	7800	500	2300	1800	2500	10000	4100	2200	1100	1000	800	500	2000	4800	4000	59400
5-10	1980	2200	2700	2700	10000	8500	46500	44400	20600	25700	1800	1700	1100	1000	1400	2200	2700	175200
	1985	2900	3300	3800	13500	12000	46600	46600	21600	26500	2200	2000	1400	1400	1700	3200	4500	193200
	1990	3200	3700	4700	15000	16000	47500	49700	22600	27400	2600	2200	1600	1500	2200	4400	5800	210100
	1995	3400	4000	5100	16500	17500	49700	52700	24600	28600	3000	2400	2000	1900	2700	5500	8000	227600
	2000	3600	4300	5400	17500	18500	51800	55200	25900	29900	3200	2700	2200	2000	2900	5000	8900	240900
TOTAL	1980																	284900
	1985																	324600
	1990																	363500
	2000																	419100

Table 2.1 Population Distribution

Table 2.2 Comparison of NBSR 9 population projections with actual 1980 population

Radius	NBSR 9 Projection	Actual Population
0-1 mi	6,800	2,900
1-2 mi	21,250	14,500
2-3 mi	23,500	16,500
3-4 mi	30,700	38,600
4-5 mi	28,000	37,200
5-10 mi	189,300	175,200
Total	299,550	284,900

3 REACTOR DESCRIPTION

3.1 Description of Reactor Complex

The NBS reactor is a reactor-laboratory complex that gives the Bureau the means of performing research and developing standards on materials and nuclear processes.

The locations of the reactor facility and other principal buildings on the Bureau grounds are shown in Figure 2.5. The nearest site boundary from the reactor is 1/4 mi to the southwest.

The complex is shown in Figure 3.1. The front, or east portion, of the complex consists of offices, cold laboratories, warm laboratories, shops, and other special purpose spaces, all of which fall outside the nuclear confinement boundary. The north wing of this nonconfinement portion of the structure has a basement area wherein is located various equipment, emergency power units, and storage space. This basement area also is accessible through a truck ramp on the northwest or rear side of the building.

3.2 Description of Confinement or Reactor Building

The reactor--or confinement portion--of the structure attaches to the office and laboratory building on the latter's west or rear side. The reactor structure is a three-level building, 90 ft × 90 ft in cross section, designed to confine the results of any credible accident. In addition to housing the reactor, it provides space to carry out the scientific programs for which the reactor was designed. Although space is provided within the confinement building for both beam tube and incore irradiation experiments, the additional laboratories and offices required to support the scientific programs are not located within the confinement building.

The reactor confinement building is shown in Figures 3.2 through 3.4. The building has three main levels: (1) the basement, which houses the primary process equipment and the spent-fuel storage pool; (2) the first floor, which serves the reactor beam holes; and (3) the second floor, which provides access to the top of the reactor and where the control room is located. The confinement building also is divided into three ventilation regions, each of which is serviced by a separate ventilation system to minimize mixing of the air among the regions.

Figure 3.2 shows the basement plan. About half of the basement area is devoted to the process room, which is surrounded by a thick concrete wall. Access to this room is through a steel shielding door from the pool area. Directly under the reactor in the process room is the subpile room whose 3-ft-thick concrete walls and steel access door provide shielding for the relatively high-radiation area directly under the reactor.

On the south side of the basement is the storage pool. The pool is used to store spent-fuel elements until they are shipped off site. A canal leading from the subpile room allows fuel elements to be transferred directly from the reactor vessel into the storage pool without the use of transfer casks.

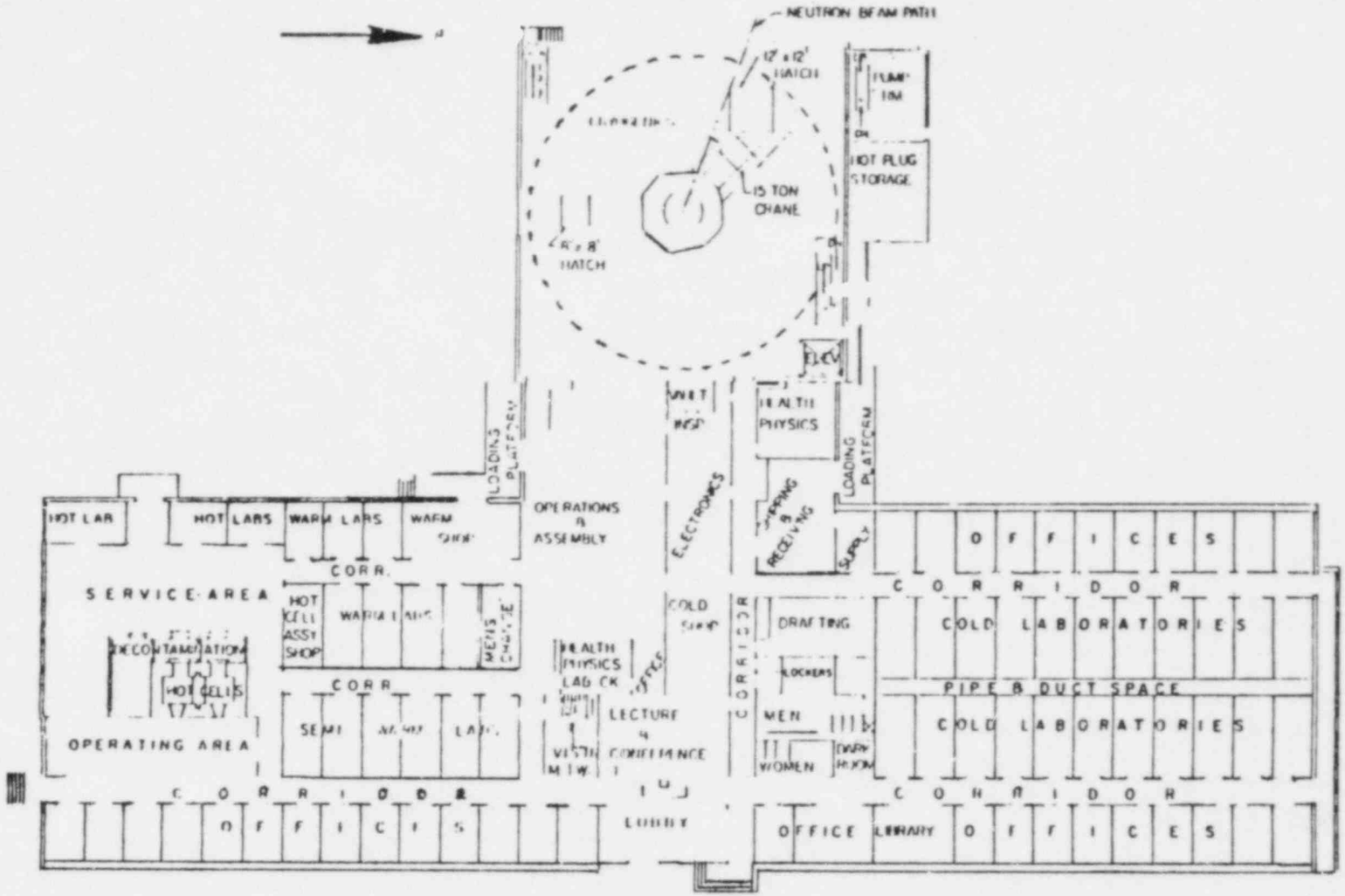


Figure 3.1 Reactor laboratory complex main-level floor plan

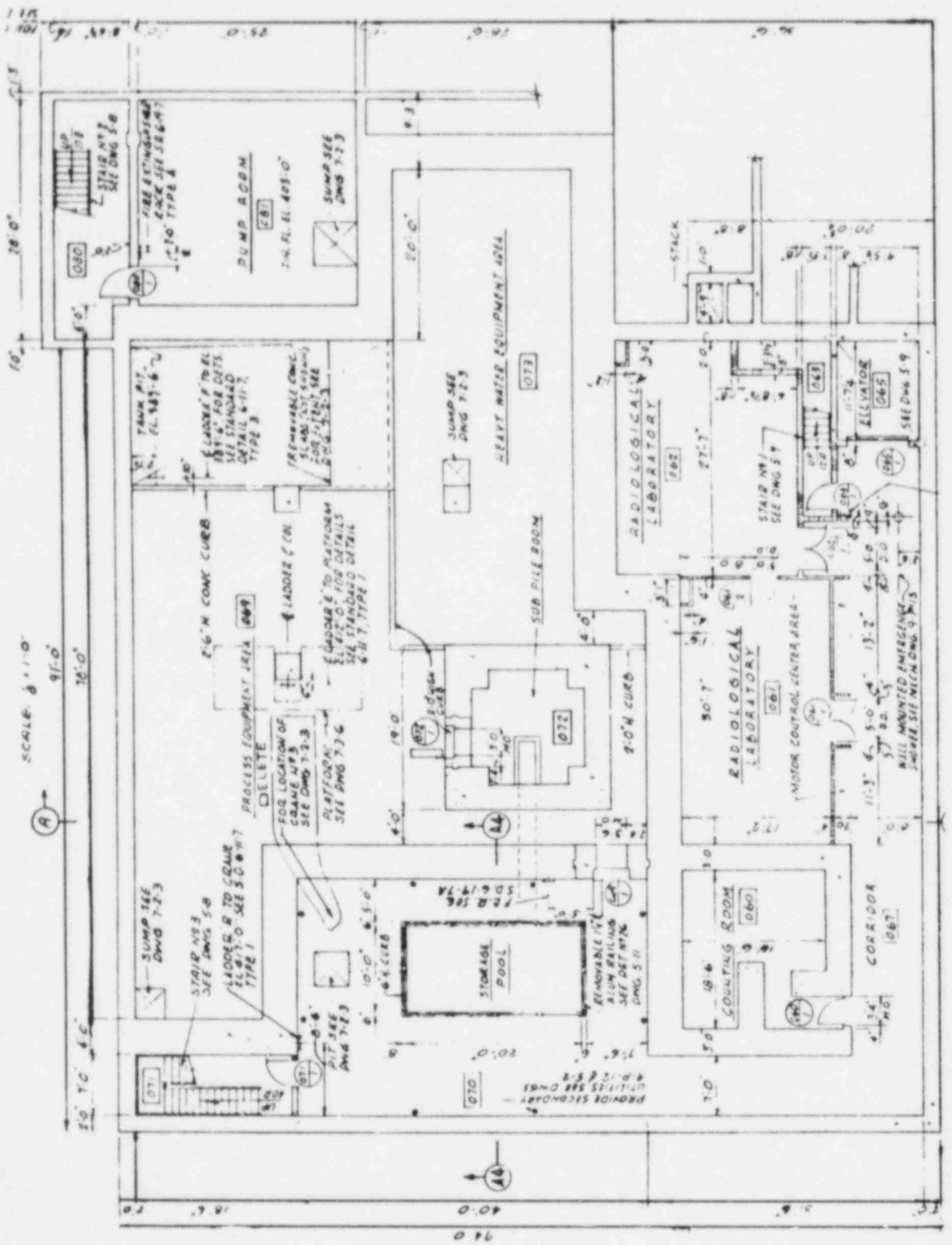


Figure 3.2 Basement plan

On the east side of the basement are two radiological laboratories and a counting room. They are used primarily in conjuction with the four pneumatic tubes which provide sample access to the reactor.

Figure 3.3 shows the first-floor plan. The experimental facilities using the beam tubes and thermal column are located on this floor. The facilities are serviced by a 15-ton annular crane. This floor is at the same level as the adjacent laboratories. Access to the reactor confinement building from the laboratories is through the two sets of doors shown on the east side of the confinement building. Each set of doors consists of two double doors separated by a short passageway. In an emergency, a sliding steel door closes and seals the door openings with an inflatable gasket.

The reactor, about 20 ft in diameter, is located in the center of the floor. The biological shield runs up to the ceiling and serves as support for the inner rail of the annular crane that services the area.

The second floor of the confinement building is shown in Figure 3.4. The top of the reactor shielding is flush with the floor. Utility and access trenches under the floor provide access to the radiation facilities that go into the core and reflector from the top of the reactor. The area is served by a 20-ton crane that is used to handle shielding casks for radioactive samples removed from the reactor. Two large square hatches in the floor provide access to the floor below, making it possible to move heavy equipment from one level to the other.

The control room also is located on the second floor and looks out over the reactor top. All the process instrumentation as well as the reactor instrumentation is located in the control room so that all aspects of reactor operation can be monitored from this single location.

3.3 Description of Reactor

An elevation drawing of the reactor is shown in Figure 3.5, and a plan view in Figure 3.6. The reactor is cooled, moderated, and reflected by heavy water. The reactor is unpressurized and operates at a low temperature (< 150°F). The fuel elements are arranged in a hexagonal array on 7-in. centers. This arrangement allows for vertical incore experimental facilities and for the use of semaphore-type shim arms. The reactor vessel is made of aluminum, as is the rest of the core structure and most of the primary cooling system.

Surrounding the vessel is the thermal shield, an iron and lead, light-water-cooled structure to protect the biological shield from excessive radiation heating. Surrounding the thermal shield and penetrated by numerous beam ports and the like is the high density concrete biological shield.

The large volume above the core is provided so that spent-fuel elements may be removed from the core, transferred to the transfer chute, and lowered into the storage pool below--all without having to remove the spent fuel from the reactor shielding. A large annular tank also is located in the region, and it is always filled with D_2O because its open top is just below the normal water level for

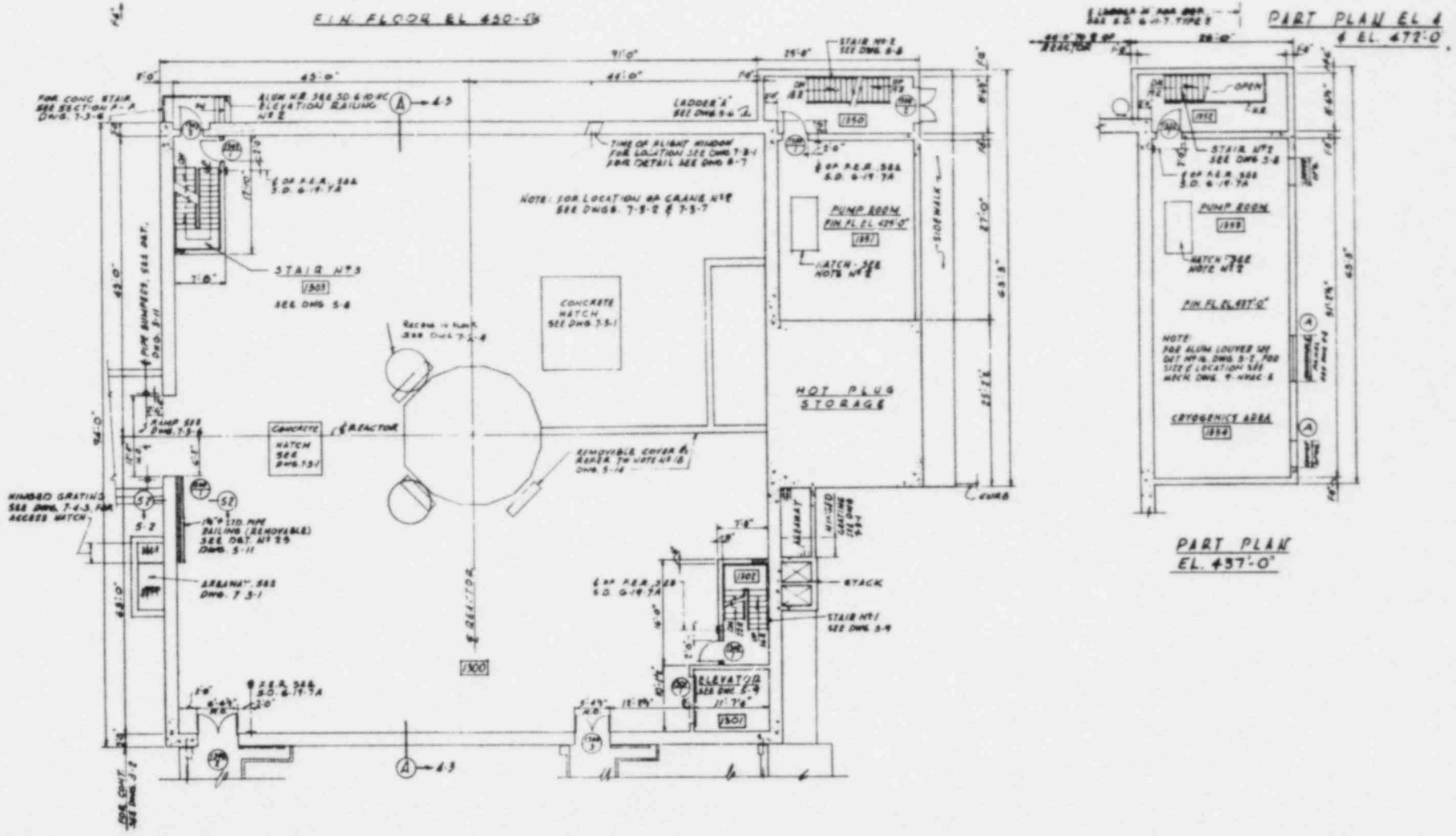


Figure 3.3 First-floor plan

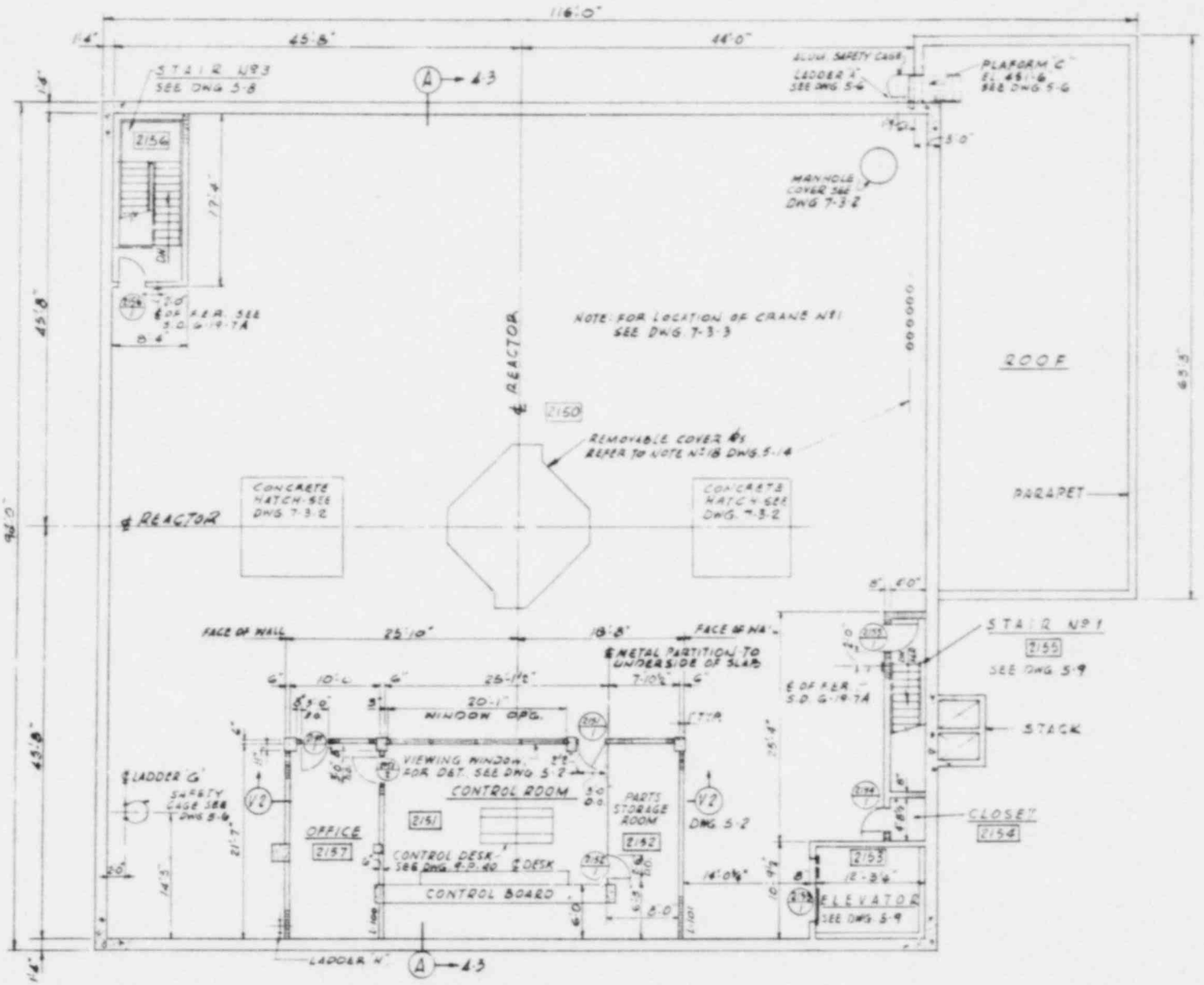


Figure 3.4 Second-floor plan

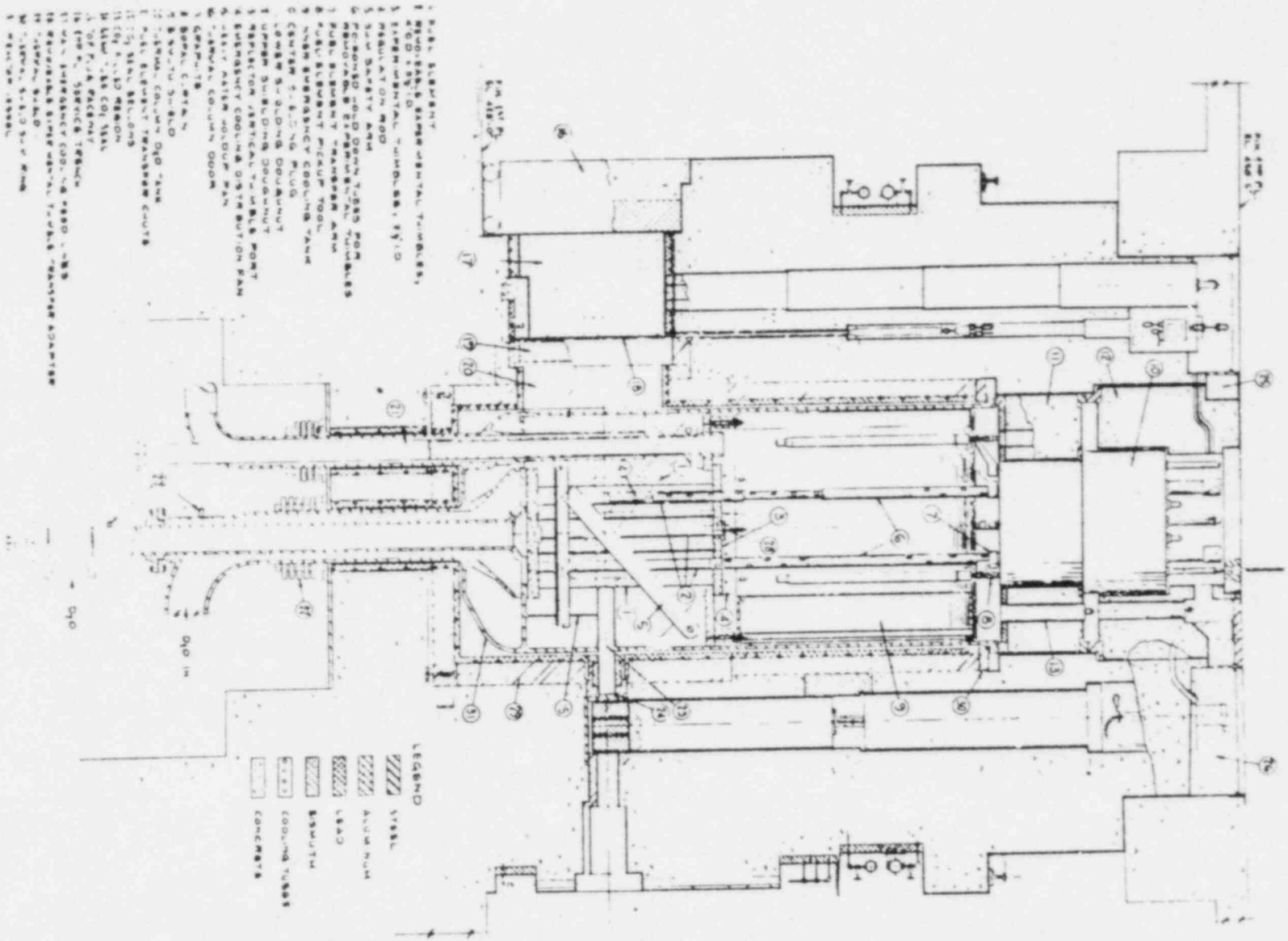


Figure 3.5 Elevation drawing of reactor

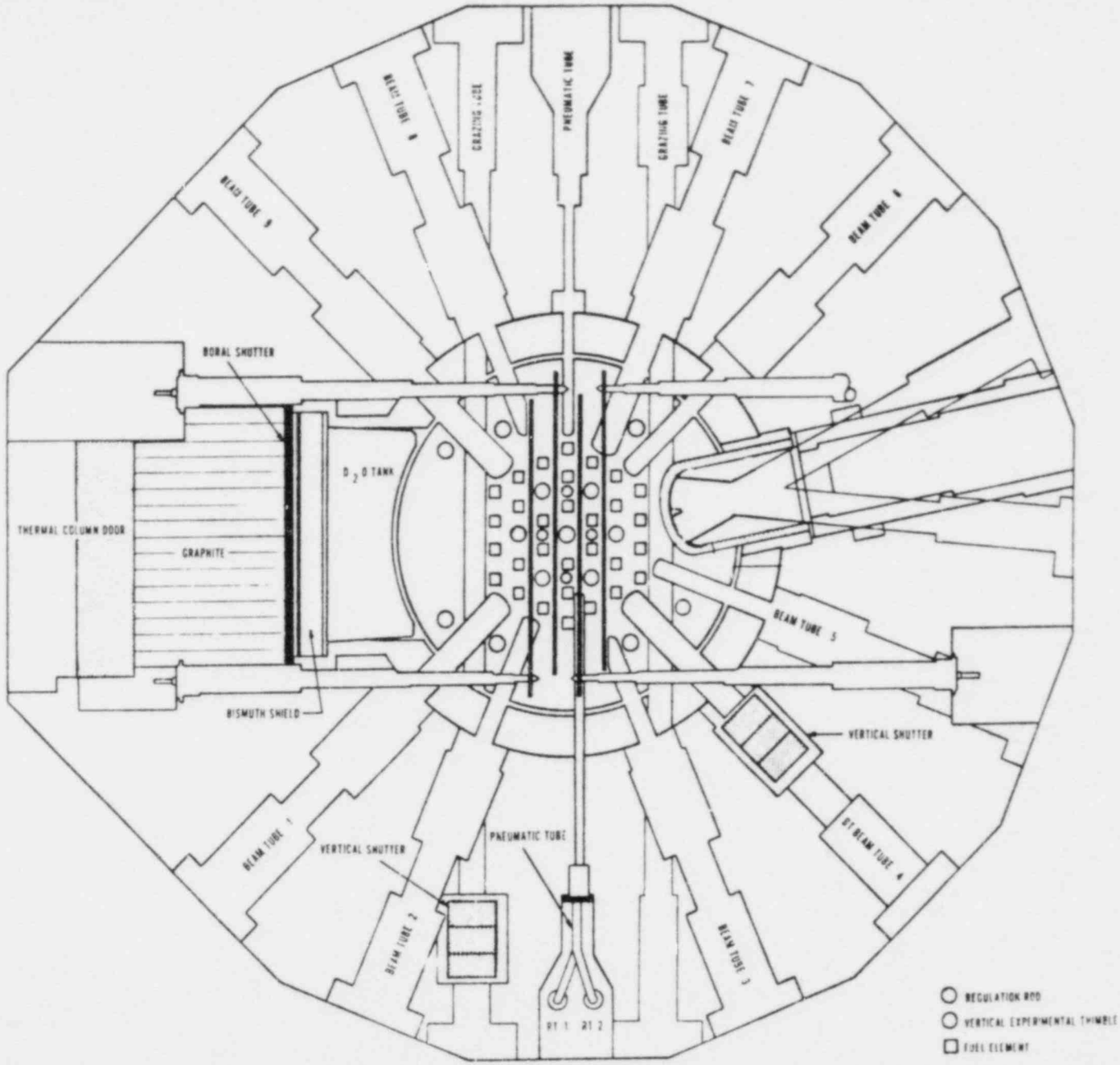


Figure 3.6 Plan view of reactor

reactor operation. This tank serves as the first step of the emergency core cooling system. The reactor is cooled by heavy water flowing up through the fuel elements and returning down through two outlet pipes at the bottom of the reactor vessel. The heat is removed from the primary system through a heat exchanger to a secondary (H_2O) cooling system that disperses its heat into the atmosphere through a cooling tower.

The core consists of an array of 30 curved-plate-type elements in which the high enriched uranium (93% ^{235}U) is combined with aluminum within the readily achievable or normal range of composition, and clad with aluminum. This array generates a well-moderated and thermalized reaction because the fuel element centers are far apart and allow for a high moderator-to-fuel ratio. The elements differ from the fuel elements usually found in other reactors, however, because the former are fueled above and below the midplane of the core and there is no fuel in the center 7 in. of the plates. Because the gap between the upper and lower regions is nearly the same as that between element centers, neutron transport along the vertical dimension takes place as readily as it does in the radial dimension. Reentrant-beam tubes welded to the reactor aluminum vessel terminate in the vicinity of the vertical fuel gap, thereby allowing extraction of neutron beams considerably freer of fast neutrons than can be obtained from the usual configuration.

One feature of the NBS reactor that makes it different from many other nonpower reactors is its method of handling fuel elements. For transferring elements to a different core position or for removing elements from the core, it is not necessary first to withdraw elements into a cask above the reactor top as in the usual CP-5 type system. Pickup and transfer tools operating through the top lid effect the shuffling and dropout under a blanket of either heavy water or helium. The above-core fuel-handling system and the bottom receiver constitute a significant engineering feature for reducing radiation exposure to workers during fuel handling.

3.4 Experimental Facilities

The reactor was designed to meet a variety of needs based on the broad spectrum of NBS activities. Consequently, the reactor has a large number of facilities of varying kinds. These include 11 radial-beam tubes, 2 through-beam tubes, a thermal column, 4 built-in pneumatic tubes, and provisions for up to 10 incore thimbles and 7 reflector thimbles. Experimental facilities currently available at the reactor include 10 fully automated neutron-scattering facilities, a small-angle neutron-scattering facility, 3 monoenergetic filtered beams in the keV region, a radiography facility, 2 standard fast-flux fields, 5 pneumatic tubes, 2 incore irradiation facilities, and a capture γ-ray facility. In addition, a fast neutron pneumatic tube is under development. Most of these instruments are used 24 hours a day, 7 days a week, and are highly automated. The reactor beam tubes are now heavily utilized, so the addition of a significant number of experimental facilities to meet increasing demand is not feasible. Doubling the reactor power, however, will make it possible both to do experiments more quickly and to do experiments of a more complex nature on the existing facilities, thereby making it possible to meet the great demand for access to these facilities.

3.5 Cooling Systems

3.5.1 Primary Cooling System

To minimize the loss of heavy water used as the primary cooling fluid, the primary loop is completely closed and sealed. The D_2O surface in the reactor vessel is swept by helium gas which is maintained at a pressure of less than 4 in. of water. This establishes the pressure in the core. The helium gas passes through a catalytic recombiner to recover any deuterium produced by radiolysis of the water. Three primary pumps rated at about 2300 gpm at 80 psig discharge pressure supply primary coolant. At 10 MW, two pumps drive the water through the reactor and a single 10-MW heat exchanger. A third primary pump is installed in the system so that any two of the three may be used. A second 10-MW heat exchanger has also been installed and checked out to serve as a backup at 10 MW. To preserve it in its new, unused condition, the connecting piping was disconnected. It will be used at 20-MW operation to provide the additional heat exchange capability required. If one heat exchanger should be out of operation, the reactor power would have to be reduced until the second heat exchanger was back on line. A fourth primary pump and its connecting piping also are on hand. Thus, the only modifications that must be made to the primary system are to reconnect the second heat exchanger and install the fourth pump. The increased volume of the system would require an increase of about 1000 gal in the heavy water inventory, but no significant increase in the loss of heavy water is anticipated.

At 10 MW, the primary water inlet temperature is about 100°F and the outlet temperature about 112°F. At 20 MW, these temperatures are expected to increase by no more than 10 or 20F° because the primary flow will be significantly increased.

3.5.2 Secondary Cooling System

The secondary cooling system has been upgraded and modernized. The original cooling tower was approaching the end of its useful life and was unable to provide adequate cooling during the summer at 10 MW. It has been replaced by a larger tower with three cells instead of two, which provides greater flexibility in choice of operating mode. The three-cell tower is 21 ft high, 27 ft wide, and 65 ft long. The capacity of the basin is 75,000 gal. At 10 MW, the average daily water makeup is about 50,000 gal. The blowdown is routed to the sanitary sewer. The average blowdown rate is about 11 gpm. The blowdown and water makeup rates should be approximately twice these numbers at 20 MW.

The secondary water treatment services for corrosion and microbiological control are provided by the Nalco Chemical Company, Chicago, Illinois. An important consideration in the selection of the treatment is that the chemicals used are environmentally acceptable. In this regard, some time ago, NBS changed from a chromate treatment, which is a superior corrosion inhibitor, to organic treatment strictly to obtain improved environmental results. The types, amounts, and functions of the chemicals used in the treatment are outlined below.

3.5.2.1 Corrosion Control

Product Identification: Nalco 8376 contains zinc for corrosion, polyester (organic phosphate) for scale inhibitor and dispersant, and liquid sulfonate

as a dispersant. This product is added at a rate of 1 to 2 gpd to maintain 30 to 50 ppm in the system with zinc concentrations maintained at 1 to 2 ppm.

3.5.2.2 Microbiological Control

(1) Product Identification: Nalco 7328 is a mixture of quarternary and organotin compounds of n-alky dimethyl benzyl, ammonium chloride (12.5%) and Bis (tris-n-butyltin oxide) (2.5%). This product is added at a rate of 5 to 6 gal per week to maintain 40 to 200 ppm in the system.

(2) Product Identification: Nalco 7320 is an organo-bromine labeled 2, 2 dibromo 3 nitrilo propionamide. This product is added at a rate of 2 liters per week to maintain 1 to 12 ppm in the system.

3.5.2.3 pH Control

Concentrated sulfuric acid is added to the system at a rate of 1 to 2 liters per day to maintain the pH in the range of 7.8 to 8.3.

3.5.2.4 Blowdown

As stated previously, the blowdown is released to the sanitary sewer at a rate of about 11 gpm and contains the chemical concentrations given above. These are further diluted by more than a factor of 20 by dilution in the NBS sanitary sewer system before leaving the NBS site.

Doubling the reactor power to 20 MW will not require any change in the chemical concentrations in the secondary cooling water and, at most, will require a doubling of the blowdown rate to 22 gpm.

3.5.3 Emergency Core Cooling

It is highly unlikely that the NBS reactor would suffer a loss-of-coolant accident (LOCA). The only mechanism for a LOCA is a rupture of the reactor vessel itself or of one of the main cooling pipes. Because the reactor primary cooling system is a low-temperature, unpressurized system, the possibility of a major rupture is extremely remote. The possibility of damaging the piping in the primary system by external forces also is extremely remote because the portion of the basement containing the piping is locked during reactor operation and heavy equipment access to that area is not possible. Nevertheless, as an extra precaution, the NBS reactor was originally designed to include both passive and active emergency core cooling systems. The completely passive system operates automatically with no electronic or moving parts and provides emergency cooling by gravity flow for the first half-hour after shutdown. The other system requires operator action and provides a longer period of cooling.

As shown in Figure 3.5, there is a large volume of water in the reactor vessel above the core. This space was provided to facilitate fuel element transfer, but it also provides the opportunity for a unique emergency core cooling capability. The passive emergency cooling system consists of an annular water tank, open at the top, which is placed in the reactor vessel above the core. The operating water level in the reactor core vessel is above the top of the annular tank so the tank is always filled with water and available for emergency cooling. The tank has two openings at the bottom leading to a distribution pan that directs

water to the top of each individual fuel element. If water is drained from the reactor vessel for any reason, the water from the annular tank automatically drains by gravity into the distribution pan and cools the fuel. No valves have to be opened, no operator action is required, and there are no signals required. The tank takes about one-half hour to drain providing time for the initiation of additional action.

A second active system provides additional cooling to the one described above and provides cooling for an indefinitely longer period of time if necessary. Because the D_2O moderator is very expensive light water, which would degrade the heavy water, it is not used initially for emergency cooling. The second system consists of a 3000-gal tank filled with D_2O located near the top of the reactor building. The operator can open one of two redundant valves, which permits water from the tank to be forced up through the core by gravity. The plenum piping is designed to minimize the possibility of a rupture in the plenum region. If, however, the plenum is ruptured in such a way that emergency cooling water would be lost and not forced up through the fuel, then the operator can simply change the valving and shunt the water from the emergency cooling tank to replenish the water drained out of the annular tank in the reactor vessel. The water spilled into the basement is collected in sumps and can be used to refill the 3000-gal emergency tank to provide continued emergency core cooling if necessary. Another potential active emergency core cooling system in the form of H_2O from the NBS fire or potable water supply system is available to back up the D_2O emergency cooling.

3.6 Confinement and Emergency Ventilation Features

3.6.1 Confinement Features

The building housing the reactor is designed to confine any radioactive material released in an accident so that it may be exhausted in a controlled manner through an emergency exhaust system that filters out the particulate and reactive radioactive materials before the air is exhausted to the environment. The NBS reactor confinement building is designed to be as tight as possible with a conservative upper limit on the allowed leak rate of 24 cfm for a pressure differential across the exterior walls of 1 in. of water.

Any release of radioactive material into the confinement system is detected by redundant detectors in the normal ventilation system and initiates a building closure. The sliding steel doors, previously mentioned, are closed automatically and sealed by inflatable gaskets, and all normal ventilation ducts are sealed shut to isolate the confinement building. The emergency exhaust fan is automatically started and maintains the building at a negative internal pressure differential across the exterior walls of about 0.25 in. of water so that any leakage is into the building. At the same time, a large internal cleanup system of 5000-cfm capacity can be activated to circulate air within the building through filters to clean it up and minimize the release of radioactive iodine to the environment.

3.6.2 Emergency Exhaust System

The emergency exhaust system consists of two redundant subsystems, A and B, each of which contains an exhaust fan and identical filters and controls. Either

exhaust system can drain air from the normal exhaust system ductwork at a rate up to 100 cfm. The fan operates to maintain a negative differential pressure of about 0.25 in. of water between the inside and the outside of the building. Each of the redundant emergency exhaust fans is provided with an ac and dc motor to ensure continuous availability even when ac power fails. The exhaust fans were sized to ensure that a negative pressure could not be generated inside the building that would impair the building's structural integrity.

3.6.3 Internal Cleanup System

During emergency operation, the air in the reactor building can be recirculated and filtered by a separate system. Air in this system is drawn from all areas of the reactor building and circulated at a rate of 5000 cfm through an absolute filter and an activated charcoal filter bank similar to the filter system for the emergency exhaust system. This system is designed to remove particulate and gaseous activity such as iodine with an approximate time constant (time required to reduce concentration by 1/e) of 2 hours.

3.7 Radwaste Treatment

3.7.1 Argon-41

Radioactive ^{41}Ar is produced by neutron capture in the stable argon normally present in air when the air is introduced into voids mainly around the reactor vessel. An extensive CO_2 system has been installed at the NBS reactor to reduce ^{41}Ar production by maintaining a positive pressure of CO_2 in the void area between the reactor vessel and the thermal shield, thereby significantly reducing the amount of air present in that area. CO_2 also is used in all pneumatic facilities, further reducing ^{41}Ar production.

3.7.2 Gaseous Tritium

Radioactive tritium is produced by neutron capture in the D_2O moderator. Gaseous tritium, in the form of water vapor, is the result of evaporation of tritiated D_2O. The primary D_2O system, including the cooling and moderating water, is a closed system to prevent the escape of the water or its vapor. All the free D_2O water surfaces are swept by helium gas that passes through a recombiner and returns the condensed vapor to the primary system storage tank. Thus, the tritiated water vapor is contained and only very small amounts escape to the environment.

3.7.3 D_2O Spills and Leakage

Tritiated D_2O may be transferred, spilled, or leaked from the closed primary system as a result of fuel transfer operations, routine maintenance activities, failure of components, and leakage from the primary side of the heat exchanger to the secondary side. The means for preventing or minimizing the D_2O leakage from each of these sources are listed below.

3.7.3.1 Fuel Transfer Operations

The release of tritium during fuel transfer operations is minimized by drying and purging the fuel transfer lock with helium before opening the isolation

valve between the reactor vessel and the storage pool. Because of their decay heat, the transferred elements are essentially dry, and only a few drops may be transferred to the storage pool.

3.7.3.2 Routine Maintenance Activities

Release of tritiated D_2O during routine maintenance activities is prevented or reduced by appropriate operational procedures and practices, temporary containment of the material, and radiological safety controls and monitoring. Most of the small amount of D_2O involved is retrieved and stored for future reprocessing.

3.7.3.3 Failure of Components

Leak detectors, located throughout the D_2O system, are employed to indicate component failure that could release tritiated D_2O. A tritium monitor constantly monitors the air within the process room and the air exhausted from the confinement building. Other reactor instrumentation, such as the reactor vessel level indicator, provides a prompt indication of D_2O losses from the system as a result of component failure.

Even if such failures were to occur, the spilled D_2O is contained within the process room and is retrieved by an installed sump pump, with little or no loss of D_2O.

3.7.3.4 Heat Exchanger Leaks

The secondary system is monitored continuously by an online ^{16}N monitor which would detect very small D_2O leaks to that system during reactor operation. This monitor is augmented by level instrumentation and by periodic sampling and evaluation of the secondary system water for the presence of tritium, using an independent counting system. Since the installation of the new stainless steel heat exchangers, there have been no failures. The amount of leakage permissible is strictly limited by NBS reactor Technical Specifications and would result in average concentrations substantially below allowable limits.

3.7.4 Solid Waste

Solid radioactive waste is produced at the facility from three principal sources: experiments, routine maintenance, and fuel cutting operations.

The amount of solid waste produced by each of the first two sources is minimized by careful planning before initiation of the activity, procedural controls, and adherence to accepted radiological safety practices. In general these are very small amounts.

The solid waste generated by fuel cutting operations is from spent fuel resulting from reactor operations. This waste has been significantly reduced by improved fuel element design and increased fuel loading.

3.8 Radwaste Disposal

3.8.1 Storage

All low-level solid waste and limited amounts of high specific-activity solid waste (such as drums of used resin) are stored and prepared for shipment in a

separate waste storage building located at the rear of the reactor facility. Spent fuel and high specific-activity waste resulting from fuel cutting operations are stored in the reactor spent-fuel storage pool. When a shipment of this material is necessary, it is loaded under water into a shipping cask and transferred directly to the carrier.

Liquid effluent from the reactor is stored in a 1000-gal retention tank or a 5000-gal holdup tank. Small amounts of liquid waste that contain activity other than tritium and cannot be released to the sanitary sewer system are collected by health physics personnel, absorbed in a suitable material, and then treated as solid waste for disposal.

3.8.2 Control

All radiological control of solid and liquid waste is the general responsibility of the NBS health physics section, after the material has been collected by health physics personnel.

Radiation monitoring of solid waste material is performed by health physics personnel. All gaseous and liquid effluents released from the facility are monitored by inline monitors, health physics personnel, or both.

Monitoring of the gaseous effluent released from reactor stack is accomplished by a radiation monitor mounted in the stack itself and sampling of the effluent air by health physics personnel.

Liquid effluent released from the facility is sampled and evaluated for activity content by health physics personnel before it is released.

3.8.3 Disposal

Solid waste is transferred to an offsite land disposal facility approved by the state or NRC for land burial. (Small amounts of liquid waste are absorbed in a suitable material and treated as solid waste.) All packaging and shipping of solid wastes and spent fuel are conducted in conformance with Department of Transportation and NRC regulations. Spent fuel is reprocessed at the Department of Energy Savannah River facility. Low-level liquid effluent is diluted and released to the sanitary sewer system in accordance with 10 CFR 20 requirements.

Gaseous activity is diluted by a high-volume stack discharge rate before its release and subsequent dispersion in the atmosphere. Concentrations at various locations within the contiguous community are detailed in Section 4 of this report.

4 ENVIRONMENTAL IMPACTS OF PROPOSED ACTION

4.1 Radiological Impacts

The primary sources of radiation exposure to plant personnel are tritium in the heavy water, direct radiation from various components of the primary process system, and the small amounts of radiation from the vicinity of the beam-tube experimental facilities.

As new beam-tube facilities are built and old ones modified, the radiation shielding is constantly being upgraded. This upgrading will ensure that the radiation dose to the users of the experimental facilities at 20-MW operation will not increase significantly above the very small exposures they currently receive.

Operating personnel do not receive significant exposures in the performance of their routine duties. When they are working in radiation areas or on radioactive or contaminated equipment, their exposures are carefully limited by the health physics section monitoring procedures. These procedures will, of course, continue to ensure that personnel exposures will remain within NRC regulations and be as low as reasonably achievable.

The increased reactor power will increase radiation levels in some maintenance situations, but in no case by more than a factor of 2 above that which would exist at the present power level. The effects of the increased levels will be mitigated by the use of increased shielding, careful decontamination, remote handling, and shorter exposure times. Furthermore, improvements and modifications to the primary system will make its maintenance (currently the major cause of operating personnel exposure) easier, which will further compensate for the somewhat higher source strengths resulting from 20-MW operation. The reactor biological shield was designed originally for 20-MW operation and is adequate to reduce radiation levels around the reactor to acceptable values.

Fuel transfer operations and the storage of spent fuel will not be significantly affected by the power increase. The fuel element being transferred is always either within the reactor shielding itself or in transit through the heavily shielded subpile room until it enters the canal leading to the storage pool. The handling within the pool is through many feet of water so no personnel exposure results from the fuel transfer procedure. This will not change at the higher power. The fuel element cutting, cask loading, and packaging of non-fueled components all take place under water so no significant dose is received now; this situation will not change at 20-MW operation. (The waste handling has been described in Sections 3.7 and 3.8 and will not result in any dose increases.)

The source strengths for operation at 10 MW and 20 MW are described below.

4.1.1 Argon-41

Over the last 3 years, an average of approximately 700 Ci of ^{41}Ar was released annually from the reactor stack. This is equivalent to an average annual

concentration of 1.6×10^{-6} μCi/mL at the stack. Using neutral atmospheric dispersion conditions (Class D, μ = 3m/s), this is reduced to $<8 \times 10^{-11}$ μCi/mL at the site boundary averaged over the year.

Even if the fact that the release takes place at the top of a 100-ft stack is ignored and the very conservative assumption is made that the release is at ground level, the concentration at the site boundary is still only 3×10^{-10} μCi/mL. If this is doubled at 20 MW (to 1400 Ci of ^{41}Ar), the concentration at the boundary will be about 1-1/2% of the maximum permissible concentration (MPC).

4.1.2 Gaseous Tritium

The average release of gaseous tritium for the last 3 years was less than 300 Ci per year. This corresponds to an average annual concentration of 7.6×10^{-7} μCi/mL at the stack, corresponding to about 3×10^{-11} μCi/mL at the boundary if neutral atmospheric dispersion conditions are assumed, and 1.2×10^{-10} μCi/mL if a ground release is assumed. The amount of release of tritiated water vapor is not expected to increase appreciably at 20 MW. However, tritium concentration in the moderator increases with reactor operations. Even if the present concentrations were to triple, the actual concentration of tritium at the boundary would be less than 1% of the allowable limits of 2×10^{-7} μCi/mL.

4.1.3 D_2O Spills and Leakage

The tritium concentration in the D_2O at the end of 1981 was approximately 1000 μCi/mL. The expecte maximum activity in the future is 2000 μCi/mL. The typical average annual releases to the sanitary sewer are less than 2 Ci of tritium and less than 2 mCi of other β-γ activity. The average annual concentration of tritium in the liquid effluent released from the reactor, when diluted by the NBS total annual discharge to the sanitary sewer system, is less than 2×10^{-6} μCi/mL, which is typically 0.004% of MPC for tritium. β-γ concentrations are similarly insignificantly small. Therefore, even if the concentrations at 20-MW operation were to be triple that at 10-MW operation, they would still be several orders of magnitude below allowable limits.

D_2O spills and leakage have been minimal and easily controlled in the past and should not be affected by an increase in the power level. Similarly, the probability of leakage through the heat exchanger should not be affected in view of the installation of reliable stainless steel heat exchangers and the quick detection of such leakage.

4.1.4 Solid Waste

The estimated average activity of the low-level radioactive waste generated by experiments and the major part of reactor maintenance activities is about 350 mCi per year.

The estimated average activity of the high specific-activity radioactive waste generated by reactor maintenance activities such as replacement of filters and resins is about 2.5 Ci per year.

The amounts of low-level solid wastes from routine experimental and reactor maintenance operations are expected to increase only slightly as a result of operation at 20 MW.

As of the end of 1981, approximately 220 spent-fuel elements had been shipped for reprocessing. These covered 8 years of 10-MW operation. Improved fuel element design and increased loading over the years have reduced the number of elements used annually by more than a factor of 2 from that originally planned. Further improvements are planned so that the number of spent-fuel elements used per year at 20-MW operation will be less than that originally planned for 10-MW operation. The same holds true for the number of nonfuel sections of spent-fuel element shipped as solid waste. The amount of radioactive solid waste from fuel cutting operations (that is, nonfuel sections) shipped from the facility was 84 Ci, in a total of four shipments. This is the total quantity of low-level waste produced from fuel cutting operations since the reactor became operational in 1967. It is anticipated that one shipment containing 18 Ci of fuel cutting operations will be shipped each year as a result of reactor operation at 20 MW. A maximum of 48 spent-fuel elements will be shipped each year in either two or four shipments, depending on whether the MH1A Army cask (which holds 24 elements) or the GE cask (which holds 12 elements) is used.

4.1.5 Conclusions

In summary, the staff has concluded that the radiological effects of increasing the reactor power on plant operations and personnel will be small and will remain a fraction of that allowed by 10 CFR 20. Routine operating procedures will not have to be changed, and maintenance procedures will have to be changed only to accommodate the 20-MW modifications of the process system and to minimize exposures. Inplant personnel radiological doses will not increase significantly. The typical releases that may be anticipated at 20-MW operation are shown in Table 4.1.

Table 4.1 Typical releases at 20-MW operation

Releases	Amount
Gaseous	
Argon-41	1400 Ci/yr
Tritium	900 Ci/yr
Liquid to sanitary sewer	
Tritium	3 Ci/yr
Other β-γ	<4 mCi/yr

These releases are spread almost uniformly through the year and so the concentrations in the atmosphere resulting from these releases can be averaged over the year. This also indicates the use of neutral dispersion coefficients in

the atmospheric dispersion calculations. The concentrations have been calculated for the point on the site boundary where they would be the largest, based on proximity to the source (reactor stack) and prevailing winds. Although the radioactive gases are released from the top of a 100-ft stack, a ground level release has been assumed. In the case of liquid effluents, the concentrations are diluted by the total NBS sanitary sewer effluent and are given for the point where the Bureau system joins the county system. The results for 20-MW operation are shown in Table 4.2, which also includes the concentration at the stack and the MPC levels as determined from 10 CFR 20, Appendix B, Table II.

Table 4.2 Concentrations of radionuclides for a ground level release at site boundary at 20-MW operation (μCi/mL)

Releases	Stack	Boundary	10 CFR 20 MPC
Gaseous at southeast boundary			
Argon-41	3.2×10^{-6}	5.7×10^{-10}	4×10^{-8}
Tritium	2.3×10^{-6}	3.6×10^{-11}	2×10^{-7}
Liquid to sanitary sewer			
Tritium	6×10^{-6}		1×10^{-1}
Other β-γ	1×10^{-8}		9×10^{-5}

The amount of radionuclides released as a result of the operation of the NBS reactor is small and is carefully monitored and controlled. Environmental monitoring and sampling of areas around the reactor covering a period of many years, before and after reactor startup in 1967, showed no significant deviation from background levels. Furthermore, more than 5 years of reactor stack effluent sampling has revealed no evidence of the release of ^{131}I (instrumental sensitivity better than 10 μCi/mL). As shown in Table 4.1, the anticipated radiological releases at 20-MW operation will be no more than two or three times the 10-MW levels and will result in radioisotope concentrations no greater than 2% of MPC at the site boundary.

4.1.6 Public Radiation Exposure

In estimating the health effects resulting from offsite radiation exposures as a result of normal operation of this facility, the NRC staff used somatic (cancer) and genetic risk estimators that are based on widely accepted scientific information. Specifically, the staff's estimates are based on information compiled by the National Academy of Sciences Committee on the Biological Effects of Ionizing Radiation (BEIR I, 1972). The estimates of the risks to workers and the general public are based on conservative assumptions (that is, the estimates are probably higher than the actual number). The following risk estimators were used to estimate health effects: 135 potential deaths from cancer per

million person-rems and 258 potential cases of all forms of genetic disorders per million person-rems. The cancer-mortality risk estimates are based on the "absolute risk" model described in BEIR I. Higher estimates can be developed by use of the "relative risk" model along with the assumption that risk prevails for the duration of life. Use of the relative risk model would produce risk values up to about four times greater than those used in this report. The staff regards the use of the relative risk model values as a reasonable upper limit of the range of uncertainty. The lower limit of the range would be zero because health effects have not been detected at doses in this dose-rate range. The number of potential nonfatal cancers would be approximately 1.5 to 2 times the number of potential fatal cancers, according to the 1980 report of the National Academy of Sciences Committee on the Biological Effects of Ionizing Radiation (BEIR III, 1980).

Values for genetic risk estimators range from 60 to 1500 potential cases of all forms of genetic disorders per million person-rems (BEIR I). The value of 258 potential cases of all forms of genetic disorders is equal to the sum of the geometric means of the risk of specific genetic defects and the risk of defects with complex etiology.

The preceding values for risk estimators are consistent with the recommendations of a number of recognized radiation-protection organizations, such as the International Commission on Radiological Protection (ICRP, 1977), the National Council on Radiation Protection and Measurements (NCRP, 1975), the National Academy of Sciences (BEIR III, 1980), and the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR, 1977).

The risk of potential fatal cancers in the exposed population residing within 10 mi of the NBS facility and the risk of potential genetic disorders in all future generations of this population is estimated as follows: multiplying the annual population dose summarized in Table 4.3 (about 6.9 person-rems, 5.7 of which are the result of ^{41}Ar) by the risk estimators, the staff estimates that about 10^{-3} cancer deaths may occur in the total exposed population and about 2×10^{-3} genetic disorders may occur in all future generations of the same exposed population. The value of 10^{-3} cancer deaths means that the probability of 1 cancer death over the lifetime of this population as a result of 1 year of facility operation is about 1 chance in 1000. The value of 2×10^{-3} genetic disorders means that the probability of 1 genetic disorder in all future generations of this population as a result of 1 year of facility operation is about 2 chances in 1000.

4.2 Nonradiological Impacts

The nonradiological impacts as a result of the operation of the NBS reactor are discussed below.

4.2.1 Aquatic

Impacts to the aquatic ecosystem are expected to be minimal.

Blowdown of 22 gpm (32,000 gpd) from the cooling tower basin with a concentration of 1 to 2 ppm of zinc (for corrosion control) and 600 ppm of dissolved

Table 4.3 Calculated dose commitments for the population within 10 mi of operation of the NBS research reactor

Source	Total Body Dose* (person-rem/year)
Natural background radiation**	39,400
Liquid effluents***	0
Argon-41	5.7
Tritium	1.2

*Dose calculations are based on models discussed in Regulatory Guide 1.109, Revision 1, using the atmospheric concentrations in Table 4.2 of this statement.

**Natural Radiation Exposure in the United States," U.S. Environmental Protection Agency, ORP-SID-72-1, June 1972; using the average background for Maryland of 94 mrems/year, and year 2000 projected population of 419,100.

***No liquids are released to potential drinking water sources.

solids* will result in an annual discharge of about 100 lb of zinc and about 20 tons of dissolved solids to the sanitary sewer system. These discharges will not cause detectable changes in the composition of the Washington Suburban Sanitary Commission (WSSC) sanitary sewer system waste, where the average daily capacity is 180 million gpd.

No impacts on aquatic biota as a result of impingement, entrainment, heat, or chlorination are expected to occur because the closed-cycle secondary cooling system has its intake from the treated Montgomery County water supply, and blowdown is discharged to the sanitary sewer system.

The loss of 100,000 gpd of water as a result of evaporation and cooling tower drift is small compared to the total water supply available. It represents less than 0.1% of the 140 million gpd average capacity of the WSSC water supply system.

4.2.2 Terrestrial

Impacts to the terrestrial ecosystem also are expected to be minimal.

*Most of the salts and dissolved solids are already present in the makeup water provided by the WSSC, and do not originate on the NBS site.

Cooling tower drift, which amounts to less than 0.2% of the circulating water flow in the secondary system, will deposit an estimated 23 tons of salts and dissolved solids on the NBS grounds and immediate vicinity each year. Impact on surrounding terrestrial vegetation will be minimal because the 41 in. or so of annual precipitation, which is distributed more or less evenly throughout the year, will wash the deposited drift from vegetative surfaces and prevent accumulation of high salt levels in the soil.

Bird impactions are not expected to occur on either the cooling towers or confinement building, and neither of these buildings is expected to affect other terrestrial fauna in other ways.

Fogging and icing as a result of cooling tower drift and evaporation are not expected except in the immediate vicinity of the cooling towers.

Some noise will occur as a result of operation of the mechanical-draft cooling towers, but it should not be objectionable at a distance of 100 yd from the towers or inside nearby buildings.

The presence of the mechanical-draft cooling towers and the confinement building may constitute what might be considered by some to be a visual impact. However, they are not aesthetically out of place on a suburban site that has other similar cooling towers and various other large modern structures scattered throughout.

4.2.3 Endangered Species

No impacts to endangered species are expected to occur as a result of reactor operation because there will be no significant impacts to either aquatic or terrestrial biota.

4.2.4 Cultural, Historical, and Archeological Impacts

No adverse cultural, historical, or archaeological impacts will occur as a result of reactor operation or maintenance.

4.2.5 Socioeconomics

Operation of the reactor should have a minimal impact on the socioeconomics of the surrounding areas. The proposed action will not change the current staff size or ratio of funding for operation. Two physical changes will be required: mechanical modification of the primary system and instrument rearrangement and modification. Both actions can be performed by the existing staff. The first costs $16,000 and takes a total of 800 hours of work. The second costs $8000 and requires 480 hours. A 2-to-3 month period is anticipated for these changes and related testing. The budget for operation at 20 MWt will be $2,400,000, or just over 1% of the NBS budget. The NBS reactor staff represents less than 1% of the NBS staff. NBS provides 75 to 85% of the operation budget and 65 to 75% of the users. The remainder of the operation budget and use will come from others. The 2-to-3 month backlogs for use of many of the facilities are expected to continue.

Thus, the proposed action is expected to have minimal socioeconomic impact because no changes in employment are forecast, the operation and modification

costs are a minor part of the NBS budget, and the costs are an even smaller factor in the local economy.

4.3 Impacts of Plant Accidents

4.3.1 Small Release Outside Containment

Regulatory Guide 4.2, Revision 2, addresses possible releases of small amounts of radioactive materials resulting from steam relief valves and other systems handling radioactive material external to the reactor containment. The NBS reactor has no such systems external to the confinement building; thus, no significant mechanism for this type of release exists. In 10 years of operation of the NBS reactor, there have been no releases of this type.

4.3.2 Radwaste System Failure

4.3.2.1 System Description

All liquid effluent from the controlled areas of the reactor building is drained into a 1000-gal retention tank. When a predetermined level is reached, liquid in the 1000-gal tank is pumped into a 5000-gal holdup tank. This tank is monitored and trips an alarm if the liquid effluent from the reactor to the tank contains significant amounts of radioactivity. The monitoring process consists of drawing a sample from the 5000-gal tank, taking it to a monitoring location in the reactor building, and analyzing it there for gross β-γ activity and tritium. The contents of the 5000-gal tank are discharged to the sewer only after the analysis has shown the activity to be at an acceptable level. The two tanks are located in concrete vaults, approximately 20 ft underground, outside the building, which will contain any leakage from the 5000-gal holdup tank. These vaults are accessible, and the tanks can be inspected visually.

4.3.2.2 Maximum Concentrations Expected

Typical annual liquid effluent releases expected at 20-MW operation from the NBS reactor are 3 Ci of tritium and less than 4 mCi of gross β-γ activity. During a year, the 5000-gal holdup tank is filled and discharged more than 20 times (20 daily discharges a year). The average activity contained in each discharge, therefore, is an order of magnitude lower than the total for the 6-month period. An upper limit on the concentration of the liquid discharge in any 1 day averaged over the day (that is, the maximum daily concentration) can be determined as follows: assuming that one-half the annual total activity was contained in one 5000-gal tank of water, the maximum tritium concentration in the tank would then be about 0.15 μCi/mL while that for β-γ activity would be 2×10^{-4} μCi/mL. When further diluted by the Bureau's daily water discharge, the upper limit on the daily maximum concentrations is about 1×10^{-3} μCi/mL for tritium and 1.5×10^{-6} μCi/mL for β-γ. These concentrations are at least a factor of 10 below allowable daily limits. The use of the activities accumulated over a 6-month period in a single release allows for any possible fluctuations.

4.3.2.3 Accidental Release

Because of the small activity and the low concentration described above, the consequences of any spill or accidental release would be insignificant even if

the entire 5000-gal in the holdup tank were involved. In case of damage to the holdup tank, the spilled effluent would be contained in the underground vault and could be disposed of by either pumping into the sanitary sewer or into suitable containers. Further, the concentrations are so small that they represent no hazard. The accidental release of the entire contents of the holdup tank into the sanitary sewer would result in a concentration substantially below allowable daily limits because it would be extensively diluted by the Bureau's daily effluent discharge.

4.3.3 Fission Product Release to Primary System

Normally, the fission product inventory of the primary system is completely negligible. The only way for fission products to get into the primary system is from a leaky fuel element. The NBS reactor is not normally operated with faulty fuel elements. If a fission product release is detected in the primary system, normal operation is terminated and a procedure to detect the faulty fuel element and subsequently remove it from the reactor begins. This happened only once during the 14 years of operation of the NBS reactor; and the resulting release of fission products to the primary system was so small that it was difficult to be sure that a faulty fuel element existed. The small release into the primary system did not result in any measurable release to the environment. The normal water treatment system for the primary system readily took care of the fission products, and once the fuel element had been removed, the primary water was easily cleaned up. Because the fission product monitor in the primary system can readily detect fission products leaking from a fuel element and because the reactor can be quickly shut down, large fission product releases to the primary system can be prevented. As small amounts of fission products may be introduced to the primary system from faulty fuel elements, and as these would not escape from the primary system to the environment, fission product releases of this type are not expected to have any significant impact on the environment or in the work area.

4.3.4 Primary to Secondary Leak

Precautions are taken to prevent the heavy water in the primary system from mixing with the light water in the secondary. Any leak is quickly detected by a detector located in the secondary system that senses the ^{16}N activity in any heavy water that might leak into the secondary system. If this detector alarms, the secondary water is sampled for tritium. D_2O levels in the primary system also are checked. If these checks confirm that a leak has developed, the reactor is shut down and steps are taken to isolate the heat exchanger. If the ^{16}N monitor indicates a ^{16}N level very much higher than the alarm set point, the reactor is shut down immediately without a delay for additional confirmation.

As stated above, the NBS reactor does not routinely operate with faulty fuel elements. Consequently, the fission products in the primary water during normal operations are not significant. If an element does develop a leak, it is quickly detected and appropriate action taken. Other than the very short-lived ^{16}N activity, the only significant radioactivity in the primary system is tritium. In addition to the ^{16}N monitor, a leak into the primary system can be detected by a change in the level of the D_2O storage tank and by periodic sampling of the secondary water for tritium. The sensitivity of these methods are such that a leak of about 36 gal in 1 day or 50 gal in 1 week can be detected.

Under either of these conditions, the reactor would be shut down and the leak corrected.

Assuming that the tritium concentration at 20 MW has reached a level of 5 mCi/mL (a very conservative number because the operations are designed to maintain the concentration below 2 mCi/mL), the 36-gal release in 1 day would result in a maximum concentration of 0.1 μCi/mL (1 MPC) in the sanitary sewer. Based on a 100% release into the atmosphere, the maximum concentration at the site boundary would be about 6×10^{-7} μCi/mL, or about 3 MPC for that day. When averaged over a year, the concentrations would be less than 1/500 MPC, including changing wind conditions.

The 50-gal leak in 1 week would, of course, give lower daily concentrations than those above, but it would give a slightly higher concentration of about 1/350 MPC at the site boundary when averaged over a year.

These leaks are of such a magnitude that they can be easily detected and the faulty tube in the heat exchanger located and repaired. It is conceivable, however, that a leak rate on the order of 0.5 gpd might be so small that it could not be located in the heat exchanger. A leak of this size could still be detected through the tritium sampling of the secondary water. It is unlikely that any such leak would remain small for a long period of time. If, however, it were not possible to locate and repair it for a whole year, 180 gal of D_2O would be released to the secondary system during the year. This would give an average airborne concentration at the site boundary of no more than 1.7×10^{-9} μCi/mL, or less than 1/100 MPC.

As stated initially, these calculations are based on an arbitrarily high tritium level of 5 mCi/mL occurring in the primary system. It is anticipated that the tritium concentration will be maintained at a much lower level, which would result in much smaller releases in the event of a leak. The only releases directly from the primary system to the environment result from the unusual occurrence of a leak in the heat exchanger and even then releases would be only a very small fracton of MPC.

4.3.5 Refueling Accidents

The top shielding plug in the reactor is never removed while there is fuel in the core. Thus, it is not credible to assume that a heavy object can fall on the core. The fuel is moved within the core or removed from the core by hand operated pickup tools built into the top plug.

The elements, weighing about 25 lb each in air and 16 lb in water, are moved one at a time. Even if one should fall from the transfer mechanism, it is not heavy enough to damage the core, which is located under a heavy top grid plate. Because the element would only drop a few feet at most, it would not be very seriously damaged. Inasmuch as the fueled plates are completely enclosed by unfueled side plates, there would most likely be no damage to the fueled plates and no release of fission products. The most severe consequence would be some damage to the end fittings or unfueled side plates that would render the element unfit for future use. This prediction is supported by observations of fuel elements that have been dropped previously at NBS during refueling without serious damage.

4.3.6 Spent Fuel Handling Accident

4.3.6.1 Fuel Element Drop-In Pool

The NBS reactor fuel element weighs only 25 lb in air and about 16 lb in water. Thus, if it were dropped in the pool during handling, it would not be subjected to significant stress and at most would dent the end fittings or side plates (unfueled). There would be no release of fission products and no personnel exposure.

4.3.6.2 Heavy Object Drop Onto Fuel Rack

Following the guidelines set forth in Regulatory Guide 4.2, the following assumptions are made:

- The void reactivity (1% of total noble gas and halogen activity) of a typical fuel element is released.
- The decay time is 30 days.
- Iodine decontamination factor in water is 500.
- Charcoal filter efficiency for iodine is 99%.

The whole-body dose for a person standing at the nearest site boundary (1300 ft) for 30 days or more was calculated assuming the fission products released by the accident were uniformly distributed throughout the building. Because the walls around the lower part of the building are thicker than those around the upper floor, the only significant contribution comes from the fraction in the top portion of the building. This is 49% of the total volume. The dose consists of two components--the sky shine from radiation penetrating the 4-in.-thick concrete roof and the direct radiation penetrating the 16-in. concrete walls. The results are given below. The whole-body dose from the cloud and the thyroid dose were estimated under the assumption of neutral dispersion coefficients (Pasquill Type D, windspeed 3 m/s) and the wind blowing into a 22.5° sector towards the nearest boundary one-third of the time. Furthermore, the very conservative assumption was made that the release was at ground level.

The calculated doses that would result from dropping a heavy object onto the fuel rack are

Whole body dose	
Direct and sky shine	4.2×10^{-6} rads
From cloud	1.7×10^{-6} rads
Thyroid dose	1.4×10^{-6} rems

These extremely small doses clearly present no threat to the environment or to the general public.

4.3.6.3 Fuel Cask Drop

It is highly unlikely that the fuel cask for shipping the NBS reactor elements could be dropped in such a way as to damage the contained elements when the cask is not within the sealed confinement building. When the truck door is open (no containment), the transfer cask is only lifted above the floor to place it on the truck. A drop from this elevation would not injure the contents of the cask. Nevertheless, for the purpose of these calculations it is assumed that such an accident does occur releasing some fission product gases. The assumptions are

- Ground level puff release is external but adjacent to the reactor building.
- One percent of total noble gas activity in fully loaded cask is released.*
- Fuel cooling period is 120 days.*
- Procedures for calculating dose are those given in ANSI/ANS 15.7-1977, "Research Reactor Site Evaluation," and Regulatory Guide 1.4, Revision 2, 1974.

From ANSI/ANS 15.7, the concentration for a puff release at a distance, x, from the puff release point and at time, t, after the release is

$$\frac{\chi}{Q} = \frac{\exp\left\{-\left(\frac{x-ut}{2\sigma_x^2} \quad \frac{y^2}{2\sigma_y^2} \quad \frac{h^2}{2\sigma_z^2}\right)\right\}}{\sqrt{2}\,\pi^{3/2}\,(\sigma_x \cdot \sigma_y \cdot \sigma_z)} \quad m^{-3}.s$$

where y is the lateral displacement from dead downwind, Q is the total curies released, h is the elevation of the release, u is the wind speed, and σ_x, σ_y, and σ_z have their usual meaning in the dispersion formulation. For a position dead downwind, y = 0. For a ground level release such as is postulated here, h = 0 and the expression for χ/Q reduces to

$$\frac{\chi}{Q} = 2\,\frac{\exp\left(-\frac{x-ut}{2\sigma_x^2}\right)}{\sqrt{2}\,\pi^{3/2}\,(\sigma_x \cdot \sigma_y \cdot \sigma_z)}$$

*Based on Regulatory Guide 4.2, Revision 1 (January 1975).

The factor, 2, arises because 100% reflection from the ground is assumed. The dose as a function of time (Regulatory Guide 1.4, Rev. 2, 1974) for ^{85}Kr (the only significant contributor) is

$$D = (0.23)\,\bar{E}_{\beta}\chi \text{ rad/s}$$

Therefore, the total dose is

$$(0.23)\,\bar{E}_{\beta}\int_{0}^{\infty}\chi dt = (0.23)(\bar{E}_{\beta})(Q)\,\frac{1}{\omega\pi\sigma_y\sigma_z}$$

Assuming Pasquill Type F dispersion conditions and a windseed of 1 m/s, this yields a dose of 0.023 rad at the site boundary.

4.3.7 Loss of Coolant

A rapid loss of coolant from the NBS reactor is extremely unlikely. It would require a major rupture in the primary system. A major rupture, such as a double-ended rupture of the 18-in. primary coolant return line, is extremely remote. All the piping is within the process room, which is locked at all times during operation, so there is no possibility of accidental damage to the piping by the heavy equipment operating in the area. The reactor operates at low pressure (80 psig maximum at the primary pump discharge) and low temperature (150°F), greatly reducing the likelihood of a major rupture.

4.3.7.1 Fission Product Release

Should a rupture occur, it is conceivable that the reactor vessel could be drained in about 30 seconds. In this event, cooling is immediately provided by water from the inner emergency cooling tank, which simply drains through the fuel elements as the water drains from the vessel. No valve action of any kind is involved. About 30 minutes is required to drain this tank, thus allowing ample time to initiate additional action. Additional emergency cooling is provided by a 3000-gal D_2O emergency cooling tank located about 30 ft above the reactor. The emergency D_2O cooling is backed up by light water from the NBS water supply.

Thus, even if a major rupture occurred anywhere in the primary piping, the elements are still adequately protected by the emergency cooling systems.

In summary, a major rupture of the primary system resulting in a rapid draining of the reactor vessel is extremely remote. If such an event should occur, the emergency cooling system will prevent excessive fuel element temperature and prevent the release of fission products. Because the inner emergency cooling tank is a passive standby system that provides emergency cooling, whenever needed, without the need for any signals, valve openings, or reactor operator action and because the loss of water shuts down the reactor, the core is protected in this type of accident for the first half hour regardless of any actions the operator might or might not take. During that time, the operator

has ample time to assess the situation and initiate the additional emergency cooling.

Because of the simple, reliable design of the emergency cooling system, no accident scenario can be conceived that would project the failure of this system at the same time that a highly unlikely major rupture occurs in the primary system. Therefore, a loss-of-coolant accident is not expected to result in any release of fission products.

4.3.7.2 Tritium Release

Although a loss of coolant will not result in the release of fission products from the fuel, the radioactivity within the water itself must be considered. The major sources of radiation in the water are tritium from thermal neutron capture in deuterium, small amounts of dissolved radioactive argon (2-hour half-life), small amounts of ^{24}Na activity from (n,α) reaction on aluminum, and traces of fission products from surface contamination of fuel elements originating during fabrication. All but the tritium activity are very small. Except for tritium, the major activity is ^{41}Ar, which is kept very low by minimizing the air content of the helium sweep system. The low concentration, combined with its 2-hour half-life, makes it insignificant. The sodium activity is kept small by maintaining very pure water so very little aluminum dissolves in the water. It presents a small, direct radiation hazard, but would remain confined to the water and decay with a 15-hour half-life. It might present a small short-lived radioactive contamination problem after the accident. The trace of fission products is very small because the fuel elements are thoroughly cleaned before installation and, of course, the NBS reactor is not operated with faulty fuel elements. Therefore, the only significant hazard is that from the tritium, which can reach levels of the order of 2000 μCi/g of D_2O in the primary system after prolonged reactor operation. Spilled D_2O will, of course, evaporate and introduce a tritium concentration in the air over the water. The results of a major spill of D_2O are analyzed below.

The accident assumes that all the water in the reactor vessel (11 m^3 or about 3000 gal) is discharged into the process room through a major pipe rupture. The building is sealed and emergency cooling maintains the integrity of the core. The area below the primary system is surrounded by a curb, which will contain the water and direct it to a sump from which it can be pumped either to the main storage tank or to the emergency cooling tank. Normally the water would be pumped out of the curbed area into one of the closed tanks, thereby reducing the area for evaporation. For the present analysis, however, it is assumed that the whole curbed area with an area of 320 ft remains flooded. The activity of the D_2O water is assumed to be 2000 μCi/g, its initial temperature 125°F, and the air temperature 80°F, with an initial relative humidity of 50%.

Based on the work of Budyko, the following expression for the rate of evaporation was used to evaluate the tritium problems:

$$M = 2.3\ (1 + 0.75s)(q_s - q)\quad g \cdot s^{-1} \cdot m^{-2}$$

where

M = mass of D_2O evaporated per second from a 1 m^2 area of water
s = air velocity across water in m/s
q_s = saturation specific humidity at temperature of water
q = specific humidity (H_2O + D_2O) of air

The two cases that have been evaluated are described below.

- Tritium release under normal ventilation conditions. No interzone mixing. The D_2O has come into thermal equilibrium with its surroundings after 2 hours.

  Assumptions:

  (1) Ventilation air flow is 7.8 m^3/s (16,500 cfm).
  (2) Air intake temperature is 27°C.
  (3) Air relative humidity is 50%.
  (4) D_2O water temperature is 42°C.
  (5) Air velocity across water surface is 0.5 m/s.
  (6) Area of water surface is 100 m^2.
  (7) Tritium concentration in D_2O = 2000 μCi/g.

The assumption subject to the greatest uncertainty is the air velocity across the D_2O. This can be estimated by asking what velocity would result from an air flow of 7.8 m^3/s through the process room, with approximate dimensions of 6 m high, 16 m wide, and 20 m long. A flow of 7.8 m^3/s flowing through an area 6 m x 16 m would give an average velocity of .08 m/s. Convection currents will, of course, increase this so a velocity of 0.5 m/s (~1 mi/hr) has been assumed to be conservative. Under these assumptions, the D_2O exhaust rate is 11.7 g/s which corresponds to 2.3 x 10^{-2} Ci/s if 2000 μCi/mL is assumed for the concentration of the liquid. Under neutral atmospheric conditions and uniform wind direction, the concentration downwind of the stack at the site boundary would only be 9 x 10^{-7} μCi/mL, corresponding to about 4 MPC for unrestricted areas.

- Tritium release under emergency conditions. No mixing with remainder of building.

  The pumpout rate is assumed to be 3% of the room volume in 2 hours based on inleakage, a falling barometer, and vapor pressure buildup. This yields an exhaust rate of 8.3 x 10^{-3} m^3/s resulting in a tritium exhaust rate of 6.9 x 10^{-4} Ci/s. Under neutral atmospheric conditions and uniform wind direction, this yields a concentration at the downwind site boundary of only 3 x 10^{-8} μCi/mL or less than 0.2 MPC for unrestricted areas.

  If mixing were allowed with the remainder of the building, the exhaust rate would go up and the concentration at the downwind site boundary would increase to 2 x 10^{-7} μCi/mL or 1 MPC for unrestricted areas.

This calculation assumed that the tritium is released from the top of the stack. If the very conservative assumption is made that the building wake causes the plume center line to be at ground level giving the equivalent of a ground release, then the concentration at the site boundary under neutral

atmospheric conditions (Pasquill Type D, windspeed 3 m/s) and wind variable within a 22.5° sector would be 7.3×10^{-8} μCi/mL for the no building mixing case and 6.2×10^{-7} μCi/mL if mixing takes place with the remainder of the building. These concentrations correspond to 0.4 mpc and 3 mpc, respectively.

4.3.8 Accident Initiation Events Considered in the Design-Basis Accident

The design-basis accident (DBA) assumed the complete blockage of one fuel element. This would be detected almost immediately by fluctuations in the nuclear instrumentation, and the reactor would be quickly shut down. Consequently, there would be very little fission product release. It is assumed, however, that the reactor is not shut down and that all the fuel cladding on the blocked element melts releasing fission products into the water. This is the accident which initiates the DBA. The DBA goes one step further and assumes that all the fission products released by the element escape from the primary system into the confinement building although there is no credible mechanism for this.

This section addresses those events which could initiate the DBA. The DBA itself is analyzed in the staff's SER. The flow blockage, which could initiate the DBA, could release fission products into the primary system. Because the primary cooling loop is a closed system, this event would not lead to any release of radioactivity outside the primary system. Other accidents considered involve the rupture of primary system components. Because all primary system components are located completely within the confinement building, no rupture would release primary water directly to the exterior of the building with the exception of leaks in the heat exchangers (see Section 4.3.4). The loss of coolant accident resulting in the loss of tritiated water to the basement of the confinement building is treated in Section 4.3.7.

4.3.9 Summary

The impact of various plant accidents has been evaluated following the guidance provided by Regulatory Guide 4.2, Revision 2. Excluding the DBA, which is analyzed in the staff's SER, the most severe accident was the loss of coolant with the resultant release of tritium into the confinement building basement. Under the conservative assumption of a ground level release of tritiated water vapor from the sealed confinement building, this accident resulted in a tritium concentration at the nearest site boundary of 6.2×10^{-7} μCi/mL during the first 2 hours. If the release were assumed to continue for 24 hours at the initial rate, the concentration, averaged over one year, would be 1.7×10^{-9} μCi/mL or less than 1% of MPC.

4.4 Impacts of Decommissioning and Decontamination

The annualized cost of decommissioning and decontaminating (D&D) the NBS reactor is essentially the same as that of continued operation. Therefore, little or no additional Congressional appropriation will be required. Nevertheless, NBS plans, if necessary, to request through the Department of Commerce the funds needed for D&D once the reactor is no longer in operation.

The following estimates are based upon the July 31, 1980, estimated costs identified by Argonne National Laboratory (ANL) for the D&D of the CP-5 Research

Reactor Facility. The CP-5 facility was operated for 19 years at a power level of 5 MW. It is heavy-water moderated and cooled reactor operating with aluminum clad, highly enriched uranium fuel in the aluminum tank surrounded by a heavy biological shield containing many beamports. This similarity to the NBS facility adds greatly to the confidence to be placed in using the cost estimates as a basis for estimating NBS reactor D&D costs.

The following assumptions were used in applying the ANL estimates to generate the cost estimates shown in Table 4.4:

- All costs are in FY 1980 dollars.
- D&D is expected to take 3 years and would start about 2 years after final fuel removal.
- The D&D, when completed, would make available for unrestricted use the existing high-bay structure and 20-ton overhead crane. Costs for remodeling are not included.
- Deconstruction costs in the Washington, D.C. area are assumed to be 20% higher than the Chicago, Illinois area.

Table 4.4 Cost estimates for decontamination and decommissioning the NBS reactor (in $1000)

Task	Cost
Preliminary engineering & design	$ 400
Fuel disposal	120
Heavy water reprocessing	400
Site preparation	290
Ancillary structures	40
Structure components	3,200
Control blades, experimental facilities, shield plug, thermal shield, tank, thermal column, biological shield, etc.	
Process systems	280
Electrical, instruments, cooling systems, rabbit system, ventilation, fuel pool, etc.	
Repair and refurbishment	320
Final decontamination	160
Subtotal	$ 5,210
Engineering design & inspection (24% of construction)	900
Contingency (25% of construction & engineering)	1,180
Staff personnel and overhead	
1st year	1,600
4 subsequent years	2,940
Total	$ 11,830

As part of the decommissioning process, major systems and components will be dismantled and disposed of as waste. Detailed plans will be formulated at the time of decommissioning; however, is is assumed that most of the work in high-radiation areas will be done remotely or behind shielding, that adequate wait time will be provided for short-lived activities to decay, and that wherever possible the waste will be compacted. It is also assumed that 50 workers will be involved ove: a 5-year period. On this basis it is estimated that the average annual exposure per worker will be less than 0.5 rem. This estimate is consistent with that estimated for the CP-5 reactor at ANL after allowing for power level differences.

Table 4.5 lists the estimated volumes of waste generated from the dismantling of major systems.

Table 4.5 Estimated volumes of waste from dismantling major systems

Material	Volume (ft^3)	Activity
Concrete, including rod and coil	15,000	LSA*
Lead	100	LSA
Steel, including stainless	100	LSA
Two stainless steel heat exchangers	1,500	LSA
Aluminum	330	LSA

*Low specific activity

It also is estimated that an additional 1100 ft^3 of low-level waste will be generated from auxiliary systems and from general cleanup. These include other materials such as graphite, bismuth, cadmium, and Boral, and some organic materials.

4.5 Impacts From the Uranium Fuel Cycle

The Uranium Fuel Cycle rule, 10 CFR 51.20 (44 FR 45362), reflects the latest information relative to the reprocessing of spent fuel and to radioactive waste management as discussed in NUREG-0116, "Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," and NUREG-0216, which presents staff responses to comments on NUREG-0116. The rule also considers other environmental factors of the uranium fuel cycle, including aspects of mining and milling, isotopic enrichment, fuel fabrication, and management of low- and high-level wastes. These are described in the Atomic Energy Commission (AEC) report WASH-1248, "Environmental Survey of the Uranium Fuel Cycle." The NRC staff also was directed to develop an explanatory narrative that would convey in understandable terms the significance of releases in the table. The narrative also was to address such important fuel cycle impacts as environmental

dose commitments and health effects, socioeconomic impacts and cumulative impacts, where these are appropriate for generic treatment. This explanatory narrative was published in the Federal Register on March 4, 1981 (46 FR 15154-15175). Appendix A to this report addresses those impacts of the light-water-reactor (LWR)-supporting fuel cycle that seem to be significant for individual reactor licensing and to warrant attention for the purposes of the National Environmental Policy Act.

Table S-3 of the final rule is reproduced in its entirety as Table 4.6 herein. Specific categories of natural resource use included in the table relate to land use, water consumption and thermal effluents, radioactive releases, burial of transuranic and high- and low-level wastes, and radiation doses from transportation and occupational exposures. The contributions in the table for reprocessing, waste management, and transportation of wastes are maximized for either of the two fuel cycles (uranium only and no recycle); that is, the cycle that results in the greater impact is used.

Appendix A to this report contains a description of the environmental impact assessment of the uranium fuel cycle as related to the operation of the 1000 MW (electric) reference reactor. The environmental impacts are based on the values given in Table S-3, and on an analysis of the radiological impact from ^{222}Rn and ^{99}Tc releases. The NRC staff has determined that the environmental impact of the reference facility on the U.S. population from radioactive gaseous and liquid releases (including radon and technetium) resulting from the uranium fuel cycle is very small when compared with the impact of natural background radiation. In addition, the nonradiological impacts of the uranium fuel cycle have been found to be acceptable. The NBS research reactor, operating at 20 MWt, is more than 100 times smaller than the reference reactor. Therefore, its radiological impact on the U.S. population through the fuel cycle is less than 1% of that found acceptable in Appendix A. To put this into perspective, the radiological impact through the fuel cycle resulting from 20 years of operation of the NBS reactor is equivalent to operating the reference reactor for a few months.

4.6 Environmental Monitoring Program

4.6.1 Monitoring Methods

The environmental measurement and monitoring program at the NBS reactor includes a variety of sampling and analysis activities to detect any changes in environmental radioactivity levels and the radiation background as a result of reactor operation. Samples of soil, grass, and water are collected and analyzed for activity. Measurements of the ambient radiation level at the site perimeter also are made.

4.6.2 Soil and Grass Sampling

Samples of soil from five designated areas on the NBS site are collected monthly, except during the growing season. During the growing season, grass samples are collected monthly. Both soil and grass samples are radiochemically analyzed for ^{90}Sr content following collection. The minimum detectable level for ^{90}Sr in soil and grass samples is better than 10^{-8} $\mu Ci/g$, which is below background levels. The sites from which soil and grass samples are collected are shown in Figure 4.1.

Table 4.6 Summary Table S-3, uranium-fuel-cycle environmental data

(Normalized to model LWR annual fuel requirement [WASH-1248] or reference reactor year [NUREG-0116])

Environmental considerations	Total	Maximum effect per annual fuel requirement or reference reactor year of model 1,000 MWe LWR
NATURAL RESOURCES USE		
Land (acres):		
Temporarily committed [1]	100	
Undisturbed area	79	
Disturbed area	22	Equivalent to a 110 MWe coal-fired power plant
Permanently committed	13	
Overburden moved (millions of MT)	2.8	Equivalent to 95 MWe coal-fired power plant
Water (millions of gallons):		
Discharged to air	160	≈2 percent of model 1,000 MWe LWR with cooling tower
Discharged to water bodies	11,090	
Discharged to ground	127	
Total	11,377	<4 percent of model 1,000 MWe LWR with once-through cooling
Fossil fuel:		
Electrical energy (thousands of MW-hour)	323	<5 percent of model 1,000 MWe LWR output
Equivalent coal (thousands of MT)	118	Equivalent to the consumption of a 45 MWe coal-fired power plant
Natural gas (millions of scf)	135	<0.4 percent of model 1,000 MWe energy output
EFFLUENTS—CHEMICAL (MT)		
Gases (including entrainment): [2]		
SO_x	4,400	
NO_x [3]	1,190	Equivalent to emissions from 45 MWe coal-fired plant for a year
Hydrocarbons	14	
CO	29.6	
Particulates	1,154	
Other gases:		
F	67	Principally from UF_6 production, enrichment, and reprocessing. Concentration within range of state standards—below level that has effects on human health
HCl	014	
Liquids:		
$SO_4^{=}$	9.9	From enrichment, fuel fabrication, and reprocessing steps. Components that constitute a potential for adverse environmental effect are present in dilute concentrations and receive additional dilution by receiving bodies of water to levels below permissible standards. The constituents that require dilution and the flow of dilution water are: NH_3—600 cfs. NO_3—20 cfs. Fluoride—70 cfs.
NO_3^-	25.8	
Fluoride	12.9	
Ca^{++}	5.4	
Cl	8.5	
Na	12.1	
NH_3	10.0	
Fe	4	
Tailings solutions (thousands of MT)	240	From mills only—no significant effluents to environment
Solids	91,000	Principally from mills—no significant effluents to environment

Table 4.6 (Continued)

[Normalized to model LWR annual fuel requirement [WASH-1248] or reference reactor year [NUREG-0116]]

Environmental considerations	Total	Maximum effect per annual fuel requirement or reference reactor year of model 1,000 MWe LWR
EFFLUENTS—RADIOLOGICAL (CURIES)		
Gases (including entrainment)		
Rn-222		Presently under reconsideration by the Commission.
Ra-226	02	
Th-230	02	
Uranium	034	
Tritium (thousands)	18.1	
C-14	24	
Kr-85 (thousands)	400	
Ru-106	14	Principally from fuel reprocessing plants.
I-129	1.3	
I-131	83	
Tc-99		Presently under consideration by the Commission.
Fission products and transuranics	203	
Liquids		
Uranium and daughters	2.1	Principally from milling—included tailings liquor and returned to ground—no effluents, therefore, no effect on environment.
Ra-226	0034	From UF_6 production.
Th-230	0015	
Th-234	01	From fuel fabrication plants—concentration 10 percent of 10 CFR 20 for total processing 26 annual fuel requirements for model LWR.
Fission and activation products	5.9×10^{-6}	
Solids (buried on site)		
Other than high level (shallow)	11,300	9,100 Ci comes from low level reactor wastes and 1,500 Ci comes from reactor decontamination and decommissioning—buried at land burial facilities. 600 Ci comes from mills—included in tailings returned to ground. Approximately 60 Ci comes from conversion and spent fuel storage. No significant effluent to the environment.
TRU and HLW (deep)	1.1×10^{3}	Buried at Federal Repository.
Effluents—thermal (billions of British thermal units)	4,063	[3] 5 percent of model 1,000 MWe LWR.
Transportation (person-rem)		
Exposure of workers and general public	2.5	
Occupational exposure (person-rem)	22.6	From reprocessing and waste management.

[1] In some cases where no entry appears it is clear from the background documents that the matter was addressed and that, in effect, the Table should be read as if a specific zero entry had been made. However, there are other areas that are not addressed at all in the Table. Table S-3 does not include health effects from the effluents described in the Table, or estimates of releases of Radon-222 from the uranium fuel cycle or estimates of Technetium-99 released from waste management or reprocessing activities. These issues may be the subject of litigation in the individual licensing proceedings.

Data supporting this table are given in the "Environmental Survey of the Uranium Fuel Cycle," WASH-1248, April 1974; the Environmental Survey of the Reprocessing and Waste Management Portion of the LWR Fuel Cycle," NUREG-0116 (Supp. 1 to WASH-1248); the "Public Comments and Task Force Responses Regarding the Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," NUREG-0216 (Supp. 2 to WASH-1248); and in the record of the final rulemaking pertaining to Uranium Fuel Cycle Impacts from Spent Fuel Reprocessing and Radioactive Waste Management, Docket RM-50-3. The contributions from reprocessing, waste management and transportation of wastes are maximized for either of the two fuel cycles (uranium only and no recycle). The contribution from transportation excludes transportation of cold fuel to a reactor and of irradiated fuel and radioactive wastes from a reactor which are considered in Table S-4 of § 51.20(g). The contributions from the other steps of the fuel cycle are given in columns A-E of Table S-3A of WASH-1248.

[2] The contributions to temporarily committed land from reprocessing are not prorated over 30 years, since the complete temporary impact accrues regardless of whether the plant services one reactor for one year or 57 reactors for 30 years.

[3] Estimated effluents based upon combustion of equivalent coal for power generation.

[4] 1.2 percent from natural gas use and process.

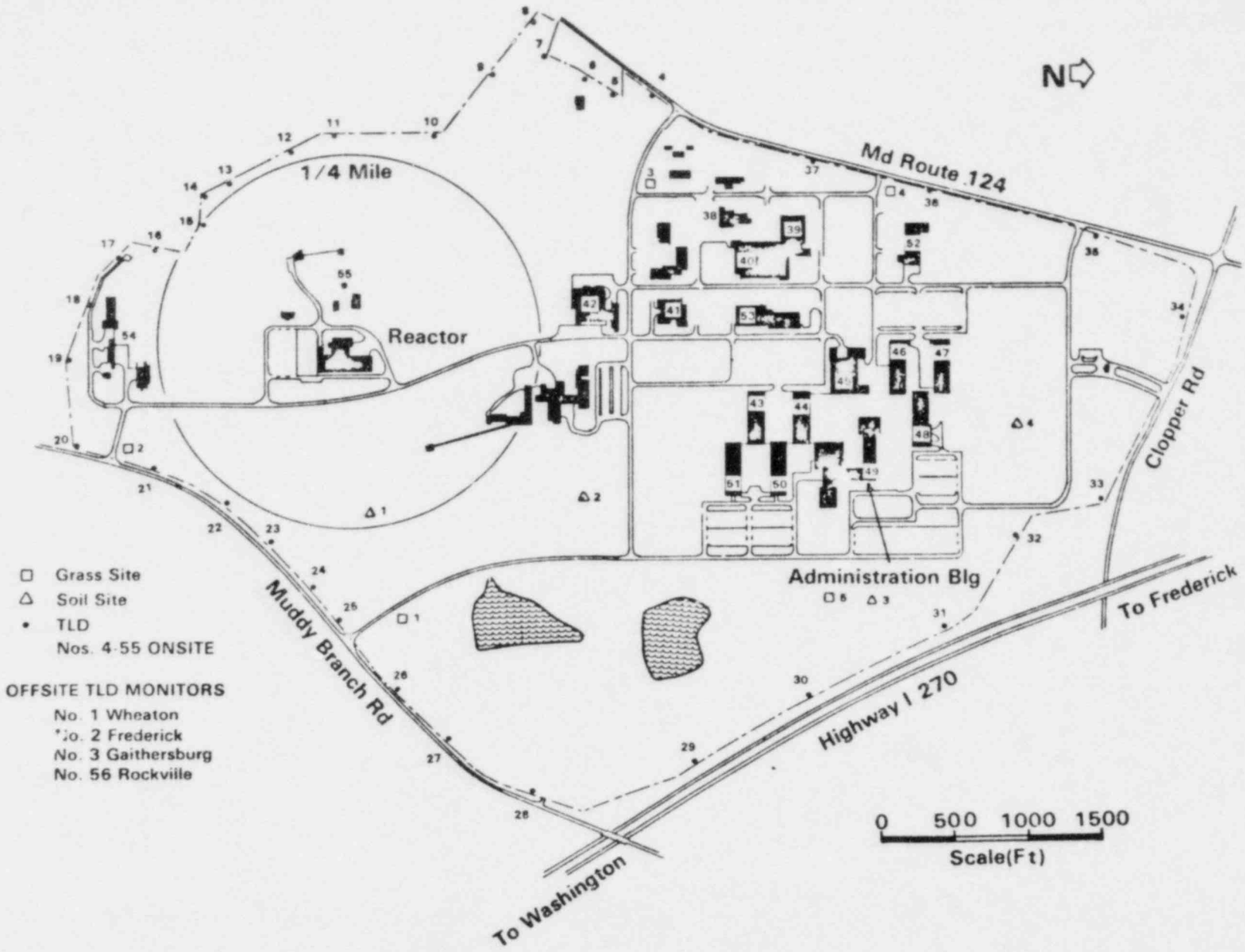


Figure 4.1 Environmental monitoring stations

4.6.2.2 Water Sampling

Samples of water in the vicinity of the NBS reactor facility are collected monthly from five surface streams and from groundwater in one residential well. These samples are analyzed for gross gamma activity and tritium content. The minimum detectable level for tritium in water samples is 2×10^{-7} μCi/mL, with the MPC for tritium at 3×10^{-3} μCi/mL. The minimum detectable level for other prominent radioisotopes in water is better than 10^{-7} μCi/mL, which is 1 percent of mpc. The locations of these sampling sites are shown in Figure 4.2.

4.6.2.3 External, Background Monitoring

The ambient-background radiation level at the NBS site is measured by more than 50 thermoluminescent dosimeters (TLDs) placed around the site perimeter fence and on NBS buildings, as shown in Figure 4.1. The minimum detectable level for external, background monitoring is 4 mrems, or less, per TLD. Four control monitors are kept at locations 3 to 20 mi from NBS.

4.6.3 Summary of Results

The environmental measurement and monitoring program at the NBS site was begun in 1963 when the monthly collection of water samples from the surrounding area was initiated. Soil and grass sampling analysis at the site was started in 1965. Measurements of the ambient background were initiated in 1966.

In December 1967, the NBS reactor went critical, and in February 1969, the reactor achieved 10-MWt operation.

No significant changes in the activity levels present in the soil, grass, and water samples collected nor the external radiation background at the site have been observed since the start of the environmental monitoring program. Minor fluctuations in levels were noted both before and after commencement of NBS reactor operation. These varied from month to month and year to year, but none of the variations could be correlated with reactor operation.

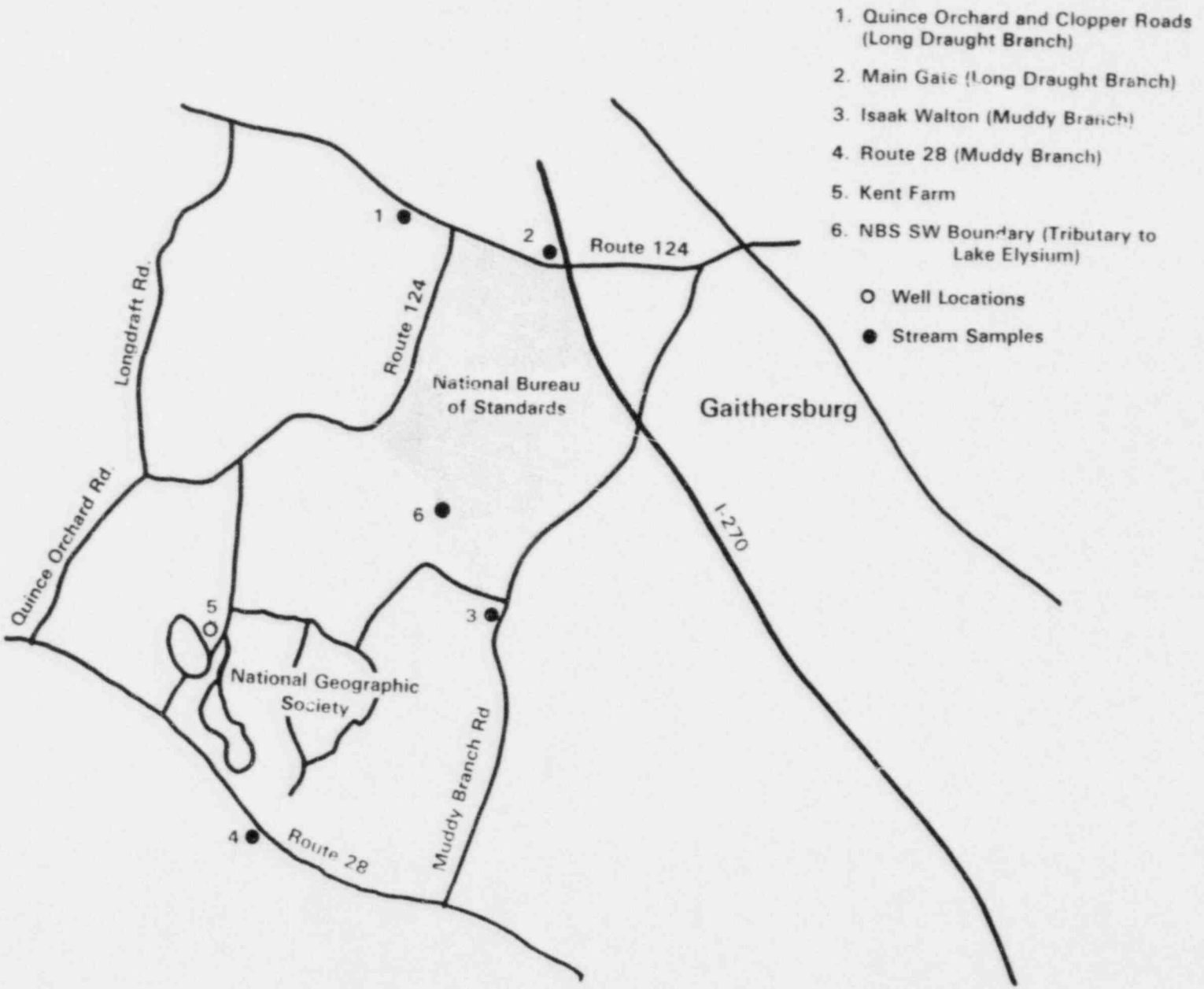


Figure 4.2 Location of water sampling sites

5 ALTERNATIVES TO PROPOSED ACTIONS

The two alternatives to the proposed action considered by the staff are (1) denial of the application and (2) license renewal at 10 MW (that is, denial of the power increase). These alternatives are discussed in the following sections.

5.1 Denial of the Application

This alternative, simply, is to allow the current license to expire June 30, 1985, by denying the application for renewal at increased power. The small environmental impacts of operation at 20 MW would not occur. Impacts of decommissioning and decontamination would occur immediately, rather than 20 years from now. The consequences of refusing the license renewal would be the elimination of the present basic and applied research, establishment of standards, and nondestructive material evaluation performed by NBS for many users. No other reactors could be used to perform the specific functions of that at NBS. Thus, a significant loss of unique services in great demand would result from immediate decommissioning. In addition, there would be a reduction in personnel after the decommissioning. The direct socioeconomic impact of the employment loss, nevertheless, would be minimal because of the small number of staff members involved and the high demand for these skilled technical personnel.

Because the environmental impacts of operation are insignificant, no benefit would accrue from shutting the reactor down. However, the cost of this alternative would be significant because the unique standards development provided by the NBS reactor would go unperformed and the ability of the NBS to perform its statutory mission would be diminished.

5.2 License Renewal at 10 MW

This alternative would renew the operating license for a period of 20 years, but would require that the maximum authorized power level be limited to 10 MW, rather than increasing it to 20 MW, as requested by NBS. The operational impacts would be roughly one half of those which would be present at a power level of 20 MW. However, because the environmental impacts of 20-MW operation are insignificant, the difference between environmental impacts of 10-MW operation compared to 20-MW operation also is insignificant. Thus, no significant benefit would be derived by requiring continued low-power operation. The overall number of procedures expected to be completed at 20 MW is 50% more than at 10 MW. Those experiments requiring the 20-MW flux density would not be possible at 10 MW. The user population and number of procedures completed is less than at 20 MW. The cost of this alternative is significant because the current work backlog would continue and the special experiments, which are possible only at 20 MW, could not be performed.

6 EVALUATION OF PROPOSED ACTION

6.1 Adverse Effects Which Cannot Be Avoided

The staff has assessed the physical, social, and economic impacts that can be attributed to the operation of the NBS reactor for 20 years at 20 MW, and has identified adverse impacts in the form of low-level releases of radionuclides and a small increase in consumption of water and electricity. The staff has determined that the unavoidable effects of facility operation are small and will have no significant adverse effects on the environment.

6.2 Irreversible and Irretrievable Commitments of Resources

Uranium is the principal natural resource irretrievably consumed in facility operation. Other materials consumed, for practical purposes, are fuel-cladding materials, reactor-control elements, other replaceable reactor core components, chemicals used in processes such as water treatment and ion-exchanger regeneration, ion-exchange resins, and minor quantities of materials used in maintenance and operation. Except for the uranium isotopes ^{235}U and ^{238}U, the consumed resource materials have widespread usage; therefore, their use in the proposed operation is reasonable with respect to needs in other industries. The major use of the natural isotopes of uranium is for production of useful energy by generation of electricity in commercial power reactors.

The reactor will be fueled with uranium enriched in the isotope ^{235}U. After use in the plant, the fuel elements will still contain ^{235}U well above the natural fraction. This highly enriched uranium, after separation from plutonium and other radioactive materials (separation takes place in a chemical reprocessing plant), is available for recycling through the gaseous diffusion plant. Scrap material containing valuable quantities of uranium also is recycled through appropriate steps in the fuel production process.

In view of the quantities of materials in natural reserves, resources, and stockpiles, and the quantities produced yearly, the expenditure of such material for the facility is justified by the benefits from the research and services performed.

6.3 Short-Term Use and Long-Term Productivity

The staff's evaluation of the use of land for the site of the NBS reactor has not changed since the construction review. The presence of this site in Montgomery County will not influence the future use of other land in its immediate environs.

6.4 Benefits of Proposed Action

Benefits of the proposed action are difficult to quantify in dollar estimates because basic scientific research can have varying consequences on a broad range of practical problems. The 20-MW operation offers (1) a more efficient

use that better meets the needs of the facility, and (2) new, more complex and sophisticated experiments that expand the basic and applied research capabilities available through 10-MW operation. Details of the many current programs are summarized in a 200-page NBS Technical Note (NBS TN-1117). Several existing and potential programs are discussed to provide examples of qualitative benefits of doubling the NBS reactor power.

The NBS reactor is a highly automated, around-the-clock, 7-day-week operation that has a 2-to-3 month backlog of users. The doubling of power will double the number of possible procedures that can be completed. The actual increase, however, is less because new, time-consuming procedures also will be added. At least a 50 percent increase in procedures is expected. Table 6.1 illustrates the current types of measurements and procedures by requested use.

Table 6.1 Percentage of types of programs requested by NBS and users

Program Type	Percentage
Material characterization	50
Trace analysis	22
Nondestructive evaluation	17
Radiation sources	5
Others	6

Material characterization, the most requested procedure, is fundamental research to determine the atomic structure of material and to study the forces holding the material together. This program seeks to answer such practical questions as what causes steel and other metals to become brittle and fail, why catalysts work to accelerate chemical reactions, and how the digestive process is accomplished.

Trace analysis identifies small amounts of impurities in many materials. For example, the procedure is used to calibrate standard reference materials used in medical laboratories, steel production, agriculture, environmental pollution studies of water and air, and tests of food and drug impurities.

Although these two general procedures represent more than 70% of the current requests by users, 20-MW operation would offer new, more sophisticated programs. For example, the high flux density can examine fundamental biological processes of complex digestive enzymes. Currently research is possible only on simple enzymes. The proposed research would test several different hypotheses of the digestive process that have direct health applications. Trace analysis would have increased sensitivity at 20 MW. The additional power will increase the understanding of why metals have different properties under different formation conditions. In addition, very high resolution diffraction would permit more definitive micro-impurity experiments, such as hydrogen embrittlement, and

provide a more efficient solar cell, through the use of very pure or controlled materials.

The old programs are available at only a few highly utilized facilities, and many of the new services are not available anywhere in the free world. The places where similar programs are available also have long backlogs (6 months or more). The programs described are only a small sample of those currently in process or planned.

6.5 Cost-Benefit Summary

The staff has evaluated the economic benefits and costs of the continued plant operation at 20 MW and has concluded that the scientific and societal benefits significantly outweigh the environmental costs.

6.5.1 Benefits

The primary benefits of the NBS reactor operation will be the continuation of current research and standards development and an increase in use and capacity for new types of research (Section 6.4).

6.5.2 Societal Costs

No major economic or societal costs are expected from either facility operation or the presence of station personnel and their families living in the area.

6.5.3 Economic Costs

The economic costs associated with the station operation at 20 MW are $24,000 for modifications and testing and $2.4 million for operation and maintenance (1981 dollars). Decommissioning costs for complete restoration of the site are expected to be $11.8 million at 1980 cost levels (Section 4.4).

6.5.4 Environmental Costs

Current analysis of environmental costs associated with the operation of the NBS reactor remain basically unchanged from the analysis performed in connection with construction and licensing permits. No change in expected impact is anticipated.

6.5.5 Environmental Costs of Uranium Fuel Cycle

The contribution of environmental effects associated with the uranium fuel cycle of a power reactor is sufficiently small that they would not alter the overall benefit-cost balance. The NBS reactor fuel cycle costs, which include expenditures for transportation of fuel and waste, are much smaller than for a power reactor.

6.5.6 Summary of Cost-Benefit

As a result of this review of potential environmental, economic, and social impacts, the staff has concluded that environmental effects of the NBS reactor's

continued operation will be insignificant, even at the higher power level. The staff finds that the benefits of continued and increased research and standards development greatly outweigh the small environmental and economic costs associated with continued operation of the reactor at 20 MW.

7 STAFF RESPONSE TO COMMENTS

7.1 Background

The NRC invited comments on the Draft Environmental Statement from interested persons by a notice published in the Federal Register on February 25, 1982 (47 FR 8273).

In response to the notice referred to above, comments were received from

- Department of Agriculture (DOA)
- Department of Health and Human Services, Food and Drug Administration (HHS)
- Department of Housing and Urban Development (HUD)
- Department of the Interior (DOI)
- Environmental Protection Agency (EPA)
- Lochstet, William A. (WAL)
- Maryland Historical Trust (MHT)

The comments are reproduced according to the date of the letter in this statement as Appendix B. The staff's consideration of the comments received and its disposition of the issues involved are reflected in part by changes in the text in the pertinent sections of this Final Environmental Statement and in part by the responses in this section. The comments from DOA, HUD, EPA, and MHT did not require a staff response; therefore, no changes were made to the statement because of those comments. The comments from HHS, DOI and WAL did require a staff response. The staff responses to those comments and the pages in Appendix B on which copies of the respective comments appear follow.

7.2 Department of Health and Human Services

HHS Comment (B-6)

> The impact of plant accidents is presented in Section 4.3 and summarized in Section 4.3.9. In the summary, it is stated that the most severe accident was a loss of coolant with resultant release of tritium into the confinement building basement. Using conservative assumptions such an accident would result in a tritium concentration at the nearest site boundary of 6.2×10^{-7} µCi/ml during the first two hours. This concentration is about three times the $(MPC)_a$ of Appendix B, Table II, Column 1. In our view, Section 4.3 should be expanded by adding a discussion on emergency preparedness which would include the facility's emergency radiation plan as well as actions that have been taken to coordinate the plan with State and local officials.

Staff Response

10 CFR 20.106(a) requires that releases of radioactive materials to unrestricted areas must not exceed the concentrations specified in Appendix B, Table II, of 10 CFR 20. This paragraph also provides that effluent concentrations may be

averaged over a 1-year period. The levels cited in this comment are for a 2-hour duration during an accident situation. When this short duration value is averaged over 1 year and combined with the normal site boundary value of 0.002 MPC, the regulatory limits for normal operation will not be exceeded. Consequently, while procedures exist to deal with situations of this nature, no offsite emergency action by the licensee is required. It shall be noted that short-term releases of up to 10 MPC are permitted. Furthermore, ANSI/ANS 15.16, as endorsed by Regulatory Guide 2.6, states that for the lowest emergency class emergency action only is required if releases exceed 10 MPC at the site boundary when averaged over 24 hours.

HHS Comment (B-6)

> Section 4.1.4 describes the typical gaseous and liquid releases at 20-MW operations. It states that the liquid releases would meet the 10 CFR 20 regulations (1977). The regulations for disposal by release into a sanitary sewerage system have been updated and are set forth in 10 CFR 20.303, dated March 27, 1981. This regulations states that no licensee shall discharge licensed material into a sanitary sewerage system unless it meets the provisions of 10 CFR 20.303(a), (b), (c), and (d). It would be helpful if this section and Section 3.8.2 (Control) could be modified to clearly state that they meet current 10 CFR 20 regulations.

Staff Response

This comment refers to a reference given in Table 4.2 of DES Section 4.1.4. The latter portion of this section was reorganized, and Table 4.2 appears in Section 4.1.5 of this report. The staff agrees that the reference to a 1977 version of 10 CFR 20 in Table 4.2 is incorrect and has been removed. The licensee is always required to meet the most current requirements of the regulations.

7.3 U.S. Department of the Interior

DOI Comment (B-4)

> Water Sampling
>
> The streams sampled in the environmental monitoring program should be named and shown on figure 4.2. Consideration should be given to including Lake Elysium, downstream from the reactor on the Muddy Branch tributary that drains the reactor site, in the sampling program.

Staff Response

Consideration has been given to including Lake Elysium in the environmental monitoring program. However, the large volume of water in lakes makes them less sensitive to change, and lakes tend to have a higher radiation background than streams. Therefore, instead of sampling Lake Elysium itself, the water sampling program has been extended to include sampling upstream from Lake Elysium at the point where the stream leaves the NBS grounds in the vicinity of the reactor. Figure 4.2, "Location of Water Sampling Sites," has been revised to show the additional sampling location and the names of the streams being sampled.

DOI Comment (B-4)

Seismicity

While the discussion of the seismicity of the area around this site is reasonable, there is no indication as to the manner in which this information was used to influence the design of either the reactor building or the equipment within that building. The final statement should indicate how this data was used in the design of the facilities.

Staff Response

The last paragraph of Section 2.4 of the FES addresses this concern.

7.4 William A. Lochstet

WAL Comment (B-9)

The evaluation of fuel supply and disposal is inadequate. The amount of fuel necessary is not stated. The total fuel, and the total amount of ore necessary to be mined for it should be presented. In addition the emmissions of radon from these ores must be considered for the full amount of time that these materials are radioactive. Since the major constituent of the ore is Uranium-238, the times necessary are in the billions of years. The curie content of the spent fuel is not explicitly stated in the Draft. The environmental impact of this reprocessing and disposal should be evaluated numerically. The impact does not stop when the spent fuel leaves the site boundary. The implied reprocessing and disposal must be considered. It is important to notice the impact of long lived products in the waste, particularly Iodine-129. This isotope should be evaluated over the entire world population for millions of years.

Staff Response

Section 4.5 of this Final Environmental Statement addresses the impacts of the uranium fuel cycle. In regard to the impacts of radon emissions and other long-lived radionuclides over the full life of the isotope, the staff has not estimated health effects from ^{222}Ra emissions beyond 1000 years for the following reasons. Predictions over time periods greater than even 100 years are subject to great uncertainties. These uncertainties result from, but are not limited to, political and social considerations, population size, and health characteristics, and, for time periods on the order of thousands of years, geologic and climatologic effects. In contrast to Dr. Lochstet's conclusion, some authors estimate that the long-term (thousands of years) impacts from the uranium used in reactors will be less than the long-term impacts from an equivalent amount of uranium left undisturbed in the ground. For example, see B. L. Cohen (1979). Consequently, the NRC staff limits its time periods of consideration to 1000 years or less for decision-making and impact-calculational purposes.

WAL Comment (B-9)

It is suggested in Table 4.1 that 1400 Ci/yr of argon-41 are expected to be released. It is shown that this will produce concentrations below maximum permissible concentrations (MPC) and therefore dismissed. Concentrations below mpc do have an environmental impact and must not be dismissed as such. One important result of the Lewis APS study of reactor safety (Reviews of Modern Physics, Summer 1975) is that if a large population is exposed to a small dose, the impact can be quite large and significant.

The situation with the projected release of 900 Ci/yr of tritium (Table 4.1) is much the same. The health impacts should be evaluated numerically as for Argon-41. In addition the possibility of getting deposited onto the ground and getting into surface or groundwater should be discussed. In particular, what fraction will end up in the DC water supply described in Section 2.2.

Staff Response

Section 4.1.6 of this statement has been added to discuss public radiation exposure, including the effects of tritium and ^{41}Ar. In regard to tritium deposition, tritium will most likely be deposited through precipitation scavenging (washout), although tritium deposits may occur when it comes in contact with soil, vegetation, and surface water. Once deposited, however, tritium can reevaporate from soil, vegetation, and water surfaces and subsequently be transported and deposited downwind, continuing the cycle of deposition, reevaporation, and transport. Tritium can reach surface waters through several different pathways, such as directly through precipitation scavenging, through molecular exchange between the atmosphere and water surface, and indirectly through runoff or influx of groundwater. Washout of tritium by precipiatation probably occurs at a rate similar to that for radioiodines. Although washout is a much more efficient mechanism for removal of maaterial from the atmosphere than dry deposition, precipitation only occurs a small fraction of the time over an annual cycle resulting in deposited amounts comparable to those estimated for dry deposition. Assuming a representative deposition rate for the vicinity of the NBS reactor, the staff expects that the amount of tritium deposited in all water reservoirs will be less than 1% of the gaseous tritium released. The amount of trititum that reaches surface water through runoff or influx of groundwater is expected to be even less than that deposited directly from the atmosphere because of retention by soil and vegetation and subsequent reevaporation and atmospheric transport.

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APPENDIX A

IMPACTS OF THE URANIUM FUEL CYCLE

The following assessment of the radiological environmental impacts of the light-water-reactor (LWR)-supporting fuel cycle is based on the values given in Table S-3 (see Section 4.4 of the main body of this report) and the NRC staff's analysis of the radiological impact from radon and technetium releases. The following analysis of fuel-cycle impacts has been cast in terms of a model 1000-MWe light-water-cooled reactor operating at an annual capacity factor of 80%. However, the staff's conclusions would not be altered if the analysis were to be based on the operation of the 20 MWt National Bureau of Standards (NBS) research reactor after applying appropriate scaling factors for the decrease in output of 3000 MWt (1000 MWe) to 20 MWt. Infact, the impact through the fuel cycle resulting from operation of the NBS reactor would be less than 1% of that resulting from the reference reactor described below.

1 RADIOACTIVE EFFLUENTS

Radioactive effluents estimated to be released to the environment from reprocessing and waste-management activities and certain other phases of the fuel-cycle process are set forth in Table S-3. Using these data the staff has calculated, for 1 year of operation of the model 1000 MWe LWR, the 100-year involuntary environmental dose commitment* to the population of the United States from the LWR-supporting fuel cycle.

It is estimated from these calculations that the overall involuntary total-body gaseous dose commitment to the U.S. population from the fuel cycle (excluding reactor releases and the dose commitment resulting from radon-222 and technetium-99) would be approximately 400 person-rems for each year of operation of the model 1000-MWe LWR (reference reactor year, or, RRY). Based on Table S-3 values, the additional involuntary total-body dose commitments to the U.S. population from radioactive liquid effluents (excluding technetium-99) as a result of all fuel-cycle operations other than reactor operation would be about 100 person-rems per year of operation. Thus, the estimated involuntary 100-year environmental dose commitment to the U.S. population from radioactive gaseous and liquid releases resulting from these portions of the fuel cycle is about 500 person-rems (whole-body) per RRY.

At this time the radiological impacts associated with radon-222 and technetium-99 releases are not addressed in Table S-3. Principal radon releases occur during mining and milling operations and as emissions from mill tailings; whereas principal technetium-99 releases occur from gaseous diffusion enrichment facilities. The staff has determined that radon-222 releases per RRY from these operations are as given in Table A.1. The staff has calculated population-dose commitments for these sources of radon-222 using the RABGAD computer code described in Volume 3 of NUREG-0002, Appendix A, Chapter IV, Section J. The results of these calculations for mining and milling activities before tailings stabilization are listed in Table A.2.

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*The 100-year environmental dose commitment is the integrated population dose for the 100 years; that is, it represents the sum of the annual population doses for a total of 100 years.

When added to the 500 person-rem total-body dose commitment for the balance of the fuel cycle, the overall estimated total-body involuntary 100-year environment dose commitment to the U.S. population from the fuel cycle for the model 1000-MWe LWR is approximately 640 person-rems. Over this period of time, this dose is equivalent to 0.00002% of the natural-background total-body dose of about 3 billion person-rems to the U.S. population.*

The staff has considered the health effects associated with the releases of radon-222, including both the short-term effects of mining and milling, and active tailings, and the potential long-term effects from unreclaimed open-pit mines and stabilized tailings. The staff has assumed that after completion of active mining, underground mines will be sealed, returning releases of radon-222 to background levels. For the purpose of providing an upper bound impact assessment, the staff has assumed that open-pit mines will be unreclaimed and it has calculated that if all ore were produced from open-pit mines, releases from them would be 110 Ci per RRY. However, because the distribution of uranium-ore reserves available by conventional mining methods is 66% underground and 34% open-pit (Department of Energy, 1978), the staff has further assumed that the uranium used to fuel LWRs will be produced by conventional mining methods in these proportions. This means that long-term releases from unreclaimed open-pit mines will be 0.34 x 110 or 37 Ci per year per RRY.

Based on the above, the radon released from unreclaimed open-pit mines over 100- and 1000-year periods would be about 3700 Ci and 37,000 Ci per RRY, respectively. The total dose commitments for a 100- to 1000-year period would be as shown in Table A.3.

These commitments represent a worst-case situation because no mitigating circumstances are assumed. However, state and Federal laws currently require reclamation of strip and open-pit coal mines, and its is very probable that similar reclamation will be required for open-pit uranium mines. If so, long-term releases from such mines should approach background levels.

For long-term radon releases from stabilized tailings piles, the staff has assumed that these tailings would emit, per RRY, 1 Ci per year for 100 years, 10 Ci per year for the next 400 years, and 100 Ci per year for periods beyond 500 years. With these assumptions, the cumulative radon-222 release from stabilized-tailings piles per RRY would be 100 Ci in 100 years, 4090 Ci in 500 years, and 53,800 Ci in 1000 years (Gotchy, 1978). The total-body, bone, and bronchial epithelium dose commitments for these periods are as shown in Table A.4.

Using risk estimators of 135, 6.9, and 22 cancer deaths per million person-rems for total-body, bone, and lung exposures, respectively, the estimated risk of cancer mortality resulting from mining, milling, and active-tailings emissions of radon-222 is about 0.11 cancer fatality per RRY. When the risk from radon-222 emissions from stabilized tailings over a 100-year release period is added, the estimated risk of cancer mortality over a 100-year period is unchanged. Similarly, a risk of about 1.2 cancer fatalities per RRY is estimated over a 1000-year release period. When potential radon releases from reclaimed and unreclaimed

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*Based on an annual average natural-background individual dose commitment of 100 millirems and a stabilized U.S. population of 300 million.

open-pit mines are included, the overall risks of radon-induced cancer fatalities per RRY range as follows:

0.11 to 0.19 fatality for a 100-year period
0.19 to 0.57 fatality for a 500-year period
1.2 to 2.0 fatalities for a 1000-year period

To illustrate: A single model 1000-MWe LWR operating at an 80% capacity factor for 30 years would be predicted to induce between 3.3 and 5.7 cancer fatalities in 100 years, 5.7 and 17 in 500 years, and 36 and 60 in 1000 years as a result of releases of radon-222.

These doses and predicted health effects have been compared with those that can be expected from natural-background emissions of radon-222. Using data from the National Council on Radiation Protection (NCRP 1975), the staff calculates the average radon-222 concentration in air in the contiguous United States to be about 150 pCi/m^3, which the NCRP estimates will result in an annual dose to the bronchial epithelium of 450 millirems. For a stabilized future U.S. population of 300 million, this represents a total lung-dose commitment of 135 million person-rems per year. Using the same risk estimator of 22 lung-cancer fatalities per million person-lung-rems used to predict cancer fatalities for the model 1000 MWe LWR, the staff estimates that lung-cancer fatalities alone from background radon-222 in the air can be calculated to be about 3000 per year, or 300,000 to 3,000,000 lung-cancer deaths over periods of 100 to 1000 years, respectively.

The staff is currently in the process of formulating a specific model for analyzing the potential impact and health effects from the release of technetium-99 during the fuel cycle. However, for the interim period until the model is completed, the staff has calculated that the potential 100-year environmental dose commitment to the U.S. population from the release of technetium-99 should not exceed 100 persons-rems per RRY. These calculations are based on the gaseous and the hydrological pathway model systems described in Volume 3 of NUREG-0002, Chapter IV, Section J, Appendix A. When these figures are added to the 640 person-rem total-body dose commitment for the balance of the fuel cycle, including radon-222, the overall estimated total-body involuntary 100-year environment dose commitment to the U.S. population from the fuel cycle for the model 1000-MWe LWR is about 740 person-rems. Over this period of time, this dose is equivalent to 0.00002% of the natural-background total-body dose of about three billion person-rems to the U.S. population.*

The staff also considered the potential health effects associated with this release of technetium-99. Using the modeling systems described in NUREG-0002, the major risks from technetium-99 are from exposure of the GI tract and kidney, although there is a small risk from total-body exposure. Using organ-specific risk estimators, these individual organ risks can be converted to total-body risk equivalent doses. Then, by using the total-body risk estimator of 135 cancer deaths per million person-rems, the estimated risk of cancer mortality resulting from technetium-99 releases from the nuclear fuel cycle is about 0.01 cancer fatality per RRY over the subsequent 100 to 1000 years.

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*Base on an annual average natural-background individual dose commitment of 100 mrems and a stabilized U.S. population of 300 million.

In addition to the radon- and technetium-related potential health effects from the fuel cycle, other nuclides produced in the cycle, such as carbon-14, will contribute to population exposures. It is estimated that an additional 0.08 to 0.12 cancer death may occur per RRY (assuming that no cure for or prevention of cancer is ever developed) over the next 100 to 1000 years, respectively, from exposures to these other nuclides.

The latter exposures also can be compared with those from naturally occurring terrestrial and cosmic-ray sources. These average about 100 millirems. Therefore, for a stable future population of 300 million persons, the whole-body dose commitment would be about 30 million person-rems per year, or 3 billion person-rems and 30 billion person-rems for periods of 100 and 1000 years, respectively. These natural-background dose commitments could produce about 400,000 and 4,000,000 cancer deaths during the same time periods. From the above analysis, the staff concludes that both the dose commitments and health effects of the LWR-supporting uranium fuel cycle are very small when compared with dose commitments and potential health effects to the U.S. population resulting from all natural-background sources.

2 RADIOACTIVE WASTES

The quantitites of buried radioactive waste material (low-level, high-level, and transuranic wastes) associated with the uranium fuel cycle are specified in Table S-3. For low-level waste disposal at land-burial facilities, the Commission notes in Table S-3 that there will be no significant radioactive releases to the environment. The Commission notes that high-level and transuranic wastes are to be buried at a Federal repository and that no release to the environment is associated with such disposal. NUREG-0116, which provides background and context for the high-level and transuranic Table S-3 values established by the Commission, indicates that these high-level and transuranic wastes will be buried and will not be released to the biosphere. No radiological environmental impact is anticipated from such disposal.

3 OCCUPATIONAL DOSE

The annual occupational dose attributable to all phases of the fuel cycle for the model 1000-MWe LWR is about 200 person-rems. The staff concludes that this occupational dose will have a small environmental impact.

4 TRANSPORTATION

The transportation dose to workers and the public is specified in Table S-3. This dose is small in comparison with the natural-background dose.

5 FUEL CYCLE

The staff's analysis of the uranium fuel cycle did not depend on the selected fuel cycle (no recycle or uranium-only recycle), because the data provided in Table S-3 include maximum recycle-option impact for each element of the fuel cycle. Thus, the staff's conclusions as to acceptability of the environmental impacts of the fuel cycle are not affected by the specific fuel cycle selected.

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Table A.1 Radon releases from mining and milling operations and mill tailings for each year of operation of the model 1000-MWe LWR*

Radon-222 source	Quantity released (Ci)
Mining**	4060
Milling and tailings*** (during active mining)	780
Inactive tailings*** (before stabilization)	350
Stabilized tailings*** (several hundred years)	1 to 10 per year
Stabilized tailings*** (after several hundred years)	110 per year

*After three days of hearings before the Atomic Safety and Licensing Appeal Board (ASLAB) using the Perkins record in a "lead case" approach, the ASLAB issued a decision on May 13, 1981 (ALAB-640) on the radon-222 release source term for the uranium fuel cycle. The decision, among other matters, produced new source term numbers based on the record developed at the hearings. These new numbers did not differ significantly from those in the Perkins record, which are the values set forth in this table. Any health effects relative to radon-222 are still under consideration before the ASLAB. Because the source term numbers in ALAB-640 do not differ significantly from those in the Perkins record, the staff continues to conclude that both the dose commitments and health effects of the uranium fuel cycle are insignificant when compared to dose commitments and potential health effects to the U.S. population resulting from all natural background sources. Subsequent to ALAB-640, a second ASLAB decision (ALAB-654, issued September 11, 1981) permits intervenors a 60-day to challenge the Perkins record on the potential health effects of radon-222 emissions.

**R. Wilde, NRC transcript of direct testimony given "In the Matter of Duke Power Company (Perkins Nuclear Station)," Docket No. 50-488, April 17, 1978.

***P. Magno, NRC transcript of direct testimony given "In the Matter of Duke Power Company (Perkins Nuclear Station)" Docket No. 50-488, April 17, 1978.

Table A.2 Estimated 100-year environmental dose commitment per year of operation of the model 1000-MWe LWR

Radon Source	Radon-222 Releases (Ci)	Dosage (person-rems)		
		Total Body	Bone	Lung (bronchial epithelium)
Mining	4100	110	2800	2300
Milling and active tailings	1100	29	750	620
Totals	5200	140	3600	2900

Table A.3 Population-dose commitments from unreclaimed open-pit mines for each year of operation of the model 1000-MWe LWR

Time span (years)	Radon-222 Releases (Ci)	Population dose commitments (person-rems)		
		Total Body	Bone	Lung (bronchial epithelim)
100	3,700	96	2,500	2,000
500	19,000	480	13,000	11,000
1,000	37,000	960	25,000	20,000

Table A.4 Population-dose commitments from unreclaimed open-pit mines for each year of operation of the model 1000-MWe LWR

Time span (years)	Radon-222 Releases (Ci)	Population dose commitments (person-rems)		
		Total Body	Bone	Lung (bronchial epithelium)
100	100	2.6	68	56
500	4,090	110	2,800	2,300
1,000	53,800	1,400	37,000	30,000

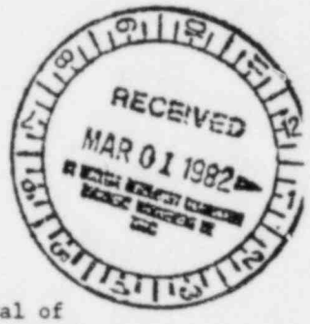
APPENDIX B
COMMENTS ON THE DRAFT ENVIRONMENTAL STATEMENT

United States
Department of
Agriculture

Economics
and Statistics
Service

Washington, D.C.
20250

February 25, 1982



Mr. James R. Miller, Chief
Standardization and Special Projects Branch
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Miller:

Thank you for sending the material concerning the renewal of Facility License No. TR-5 for operation of the National Bureau of Standards (NBS) Reactor located in Gaithersburg, Maryland.

We have reviewed Docket No. 50-184 and have no comments.

Sincerely,

VELMAR W. DAVIS
Associate Director
Natural Resource
Economics Division

DEPARTMENT OF HOUSING AND URBAN DEVELOPMENT
PHILADELPHIA REGIONAL OFFICE
CURTIS BUILDING, SIXTH AND WALNUT STREETS
PHILADELPHIA, PENNSYLVANIA 19106

REGION III

MAR 3 1982

IN REPLY REFER TO:

U.S. Nuclear Regulatory Commission
Attn.: Director, Division of Licensing
Washington, D.C. 20555

Dear Sir:

We have reviewed the Draft Environmental Statement related to the license renewal and power increase for the National Bureau of Standards Reactor and have no comment to offer.

Thank you for the opportunity to comment.

Sincerely,

Thomas J. Gola

Thomas J. Gola
Regional Administrator

Maryland Historical Trust

March 26, 1982

Mr. James R. Miller, Chief
Standardization and Special Projects Branch
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: License renewal and power increase for the
National Bureau of Standards
Reactor (NUREG-0877)
Docket No. 50-184

Dear Mr. Miller:

Thank you for the opportunity to comment on the project listed above. The license renewal and the power increase for the NBS Reactor will have no effect on historic resources.

Sincerely,

J. Rodney Little
Director/State Historic
Preservation Officer

JRL/GJA/mf

cc: Ms. Eileen McGuckian
Ms. Anita Hall

Shaw House, 21 State Circle, Annapolis, Maryland 21401 (301)269-2212, 269-2438
Department of Economic and Community Development

United States Department of the Interior

OFFICE OF THE SECRETARY
WASHINGTON, D.C. 20240

ER 82/374

APR 2 1982

James R. Miller, Chief
Standardization and Special
Projects Branch
Division of Licensing
Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Miller:

Thank you for your letter of February 18, 1982, which transmitted copies of the draft environmental impact statement related to the License Renewal and Power Increase for the National Bureau of Standards Reactor, Montgomery County, Maryland. We have reviewed the draft statement and have the following comments.

Water Sampling

The streams sampled in the environmental monitoring program should be named and shown on figure 4.2. Consideration should be given to including Lake Elysium, downstream from the reactor on the Muddy Branch tributary that drains the reactor site, in the sampling program.

Seismicity

While the discussion of the seismicity of the area around this site is reasonable, there is no indication as to the manner in which this information was used to influence the design of either the reactor building or the equipment within that building. The final statement should indicate how this data was used in the design of the facilities.

We hope these comments will be helpful to you in the preparation of a final statement.

Sincerely,

Bruce Blanchard, Director
Environmental Project Review

UNITED STATES ENVIRONMENTAL PROTECTION AGENCY
REGION III
6TH AND WALNUT STREETS
PHILADELPHIA, PENNSYLVANIA 19106

APR 15 1982

Mr. James R. Miller, Chief
Standardization and Special Projects Branch
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Re: Docket No. 50-184

Dear Mr. Miller:

We have completed our review of the Draft Environmental Impact Statement concerning the license renewal and power increase for the National Bureau of Standards Research Reactor (NUREG-0877).

We believe the proposal has been very well described in the subject document and we have no objections to the reissuing of the license and the power increase of the National Bureau of Standards Reactor.

Routine releases from the reactor (page 4-4) appear small. The primary air release is Argon-41, which has a short (1.8 hr.) half-life, which has nonreactive chemical properties, and which will not concentrate in food chains. The monitoring system is adequate (pages 4-17 through 4-20).

In view of the low impact of this action we have rated the proposal LO-1. This means we lack objections to the project and believe that the Impact Statement adequately covers all phases of the proposal.

We thank you for the opportunity to review the document.

Sincerely yours,

Peter N. Bibko
Regional Administrator

DEPARTMENT OF HEALTH & HUMAN SERVICES

Public Health Service

Food and Drug Administration
Rockville MD 20857

APR 12 1982

50-184

Mr. James H. Wilson
Licensing Project Manager
Office of Nuclear Reactor Regulations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Wilson:

The Bureau of Radiological Health staff has reviewed the Draft Environmental Statement (DES) related to license renewal and power increase for the National Bureau of Standards Reactor, NUREG-0877, dated January 1982. We have the following comments to offer:

It is recognized that the NBS reactor has been operating at 10-MW since June 1970, and during this time has maintained the individual radiation doses as low as reasonably achievable (ALARA). Increasing the power level to 20-MW operation would be within the original design of this facility. At this power level, it is anticipated that there would be some increase in radiation levels for maintenance personnel, but any radiation doses would remain within the current radiation protection standards.

The impact of plant accidents is presented in Section 4.3 and summarized in Section 4.3.9. In the summary, it is stated that the most severe accident was a loss of coolant with resultant release of tritium into the confinement building basement. Using conservative assumptions such an accident would result in a tritium concentration at the nearest site boundary of 6.2×10^{-7} uCi/ml during the first two hours. This concentration is about three times the $(MPC)_a$ of Appendix B, Table II, Column 1. In our view, Section 4.3 should be expanded by adding a discussion on emergency preparedness which would include the facility's emergency radiation plan as well as actions that have been taken to coordinate the plan with State and local officials.

Section 4.1.4 describes the typical gaseous and liquid releases at 20-MW operations. It states that the liquid releases would meet the 10 CFR 20 regulations (1977). The regulations for disposal by release into a sanitary sewerage system have been updated and are set forth in 10 CFR 20.303, dated March 27, 1981. This regulation states that no licensee shall discharge licensed material into a sanitary sewerage system unless it meets the provisions of 10 CFR 20.303(a), (b), (c) and (d). It would be helpful if this section and Section 3.8.2 (Control) could be modified to clearly state that they meet current 10 CFR 20 regulations.

The radiological monitoring program presented in Section 4.5 appears to provide adequate sampling frequency in expected exposure pathways. The analyses for specific radionuclides are considered sufficiently inclusive to verify that the liquid and gaseous emissions meet applicable radiation protection standards.

Thank you for the opportunity to review and comment on this Draft Environmental Statement.

Sincerely yours,

John C. Villforth
Director
Bureau of Radiological Health

THE PENNSYLVANIA STATE UNIVERSITY
104 DAVEY LABORATORY
UNIVERSITY PARK, PENNSYLVANIA 16802

College of Science
Department of Physics

Area Code 814

12 April 1982

50-5

Director, Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C.
20555

Dear Director:

Enclosed are my comments on the "Draft Environmental Statement related to license renewal and power increase for the National Bureau of Standards Reactor", NUREG-0877. Please note that the evaluation presented here does not necessarily reflect the opinion of the Pennsylvania State University.

I hope that these comments are used in developing the Final Environmental Statement. Would you also please send me a copy of that Final EIS when it is available.

Sincerely,

William A. Lochstet

Wm. A. Lochstet, Ph.D.

The Long Term Consequences
of Operation of
NBS Reactor
by

William A. Lochstet, Ph.D.
The Pennsylvania State University*
April 1982

The Nuclear Regulatory Commission (NRC) has attempted to meet its responsibility under section 102 of the NEPA by the Draft Environmental Statement NUREG-0877. Unfortunately, the Draft does not meet the requirement in at least three aspects. These are the impacts of releases of Argon-41, Tritium, and the uranium fuel cycle used.

It is suggested in table 4.1 that 1400 Ci/yr of argon-41 are expected to be released. It is shown that this will produce concentrations below maximum permissible concentrations (MPC) and therefore dismissed. Concentrations below mpc do have an environmental impact and must not be dismissed as such. One important result of the Lewis APS study of reactor safety (Reviews of Modern Physics, Summer 1975) is that if a large population is exposed to a small dose, the impact can be quite large and significant.

The situation with the projected release of 900 Ci/yr of tritium (Table 4.1) is much the same. The health impacts should be evaluated numerically as for Argon-41. In addition the possibility of getting deposited onto the ground and getting into surface or groundwater should be discussed. In particular, what fraction will end up in the DC water supply described in section 2.2.

The evaluation of fuel supply and disposal is inadequate. The amount of fuel necessary is not stated. The total fuel, and the total amount of ore necessary to be mined for it should be presented. In addition the emissions of radon from these ores

* For identification purposes only.

must be considered for the full amount of time that these materials are radioactive. Since the major constituent of the ore is Uranium-238, the times necessary are in the billions of years. The curie content of the spent fuel is not explicitly stated in the Draft. The environmental impact of this reprocessing and disposal should be evaluated numerically. The impact does not stop when the spent fuel leaves the site boundry. The implied reprocessing and disposal must be considered. It is important to notice the impact of long lived products in the waste, particularly Iodine-129. This isotope should be evaluated over the entire world population for millions of years.

I trust that these comments will be useful in preparing an adequate Final Statement.

NRC FORM 335 (7.77) U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC)
NUREG-0877

4. TITLE AND SUBTITLE (Add Volume No., if appropriate)
Final Environemental Statement related to license renewal and power increase for the National Bureau of Standards Reactor

2. (Leave blank)

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S)

5. DATE REPORT COMPLETED
MONTH July YEAR 1982

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
US Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Washington, D.C. 20555

DATE REPORT ISSUED
MONTH August YEAR 1982

6. (Leave blank)

8. (Leave blank)

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Same as 9 above

10. PROJECT/TASK/WORK UNIT NO.

11. CONTRACT NO.

13. TYPE OF REPORT
Final Environmental Statement

PERIOD COVERED (Inclusive dates)

15. SUPPLEMENTARY NOTES
Docket No. 50-184

14. (Leave blank)

16. ABSTRACT (200 words or less)

This Final Environmental Statement contains an assessment of the environmental impact associated with renewal of Operating License No. TR-5 for the National Bureau of Standards (NBS) reactor for a period of 20 years at a power level of 20 MW. This reactor is located on the 576-acre NBS site near Gaithersburg in Montgomery County, Maryland, about 20 mi northwest of the center of Washington, D.C. The reactor is a high-flux heavy-water-moderated, cooled and reflected test reactor, which first went critical on December 7, 1967. Though the reactor was originally designed for 20-MW operation, it has been operating for 14 years at a maximum authorized power level of 10 MW. Program demand is now great enough to warrant operation at a power level of 20 MW. No additional major changes to the physical plant are required to operate at 20 MW.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
Unlimited

19. SECURITY CLASS (This report)
Unclassified

21. NO. OF PAGES

20. SECURITY CLASS (This page)
Unclassified

22. PRICE
$

NRC FORM 335 (7.77)

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

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