NUREG-1059

Safety Evaluation Report

related to the renewal of the operating license for the Union Carbide Subsidiary B, Inc. research reactor

Docket No. 50-54

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

June 1984

NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W. Washington, DC 20555
2. The NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the NRC/GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations,* and *Nuclear Regulatory Commission Issuances.*

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

GPO Printed copy price: $4.75

NUREG-1059

Safety Evaluation Report

related to the renewal of the operating license for the Union Carbide Subsidiary B, Inc. research reactor

Docket No. 50-54

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

June 1984

ABSTRACT

This Safety Evaluation Report, for the application filed by the Union Carbide Subsidiary B, Inc. (UCS), for renewal of operating license number R-81 to continue to operate the research reactor, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is operated by UCS and is located in Sterling Forest, in the city of Tuxedo, Orange County, New York. The staff concludes that the reactor facility can continue to be operated by UCS without endangering the health and safety of the public.

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

LIST OF FIGURES

LIST OF TABLES

LIST OF TABLES (Continued)

1 INTRODUCTION AND HISTORY

By letter dated May 23, 1980, the Union Carbide Subsidiary B, Inc. (UCS/licensee) submitted an application for renewal of the Class 104 operating license (R-81) for their reactor for a period of 20 years and at the current authorized maximum power level of 5 megawatts thermal. The renewal application is supported by information provided in the Safety Analysis Report dated May 1980, Technical Specifications dated May 1980, the Environmental Impact Appraisal, the Physical Security and Emergency Plans, and the Reactor Operator Requalification Program. The UCS reactor is a pool-type, high-flux, light-water moderated and cooled, and water- and graphite-reflected research reactor. The principal function of the UCS reactor is to produce radiochemicals, radiopharmaceuticals and to provide research and service irradiations. The reactor is located on the UCS complex in an area known as Sterling Forest, in the city of Tuxedo, Orange County, New York.

The renewal application contains information regarding the original design of the facility and modifications to the facility that have been made since initial licensing. The Physical Security Plan is protected from public disclosure under 10 CFR 73.21 and 10 CFR 9.5(a)(4).

The purpose of this Safety Evaluation Report (SER) is to summarize the results of the safety review of the UCS reactor and to delineate the scope of the technical details considered in evaluating the radiological safety aspects of continued operation. This SER will serve as the basis for renewal of the license for operation of the UCS reactor facility and for continued operation at steady-state thermal power levels up to and including 5 MWt. The facility was reviewed against the requirements of 10 CFR 20, 30, 50, 51, 55, 70, and 73; applicable regulatory guides (RGs) (Division 2, Research and Test Reactors); and appropriate accepted industry standards (American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 Series). Because there are no accident-related regulations for nonpower reactors, the staff has compared calculated dose values with related standards in 10 CFR 20, the standards for protection against radiation both for employees and the public for normal reactor operations.

The staff technical safety review with respect to issuing a renewal operating license for operation of the UCS reactor has been based on the information contained in the renewal application and supporting supplements, generic studies performed by national laboratories, site visits, responses to requests for additional information, and a report prepared by the NRC staff.

Major contributors to the technical review include the NRC project manager and J. Hyder, C. Thomas, D. Whitaker, and C. Linder of the staff of the Los Alamos National Laboratory (LANL) under contract to the NRC. This material is available for review at the NRC Public Document Room at 1717 H Street, N.W., Washington, D.C. This Safety Evaluation Report was prepared by Harold Bernard, Project Manager, Standardization and Special Projects Branch, Division of Licensing, Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission.

1.1 Summary and Conclusions of Principal Safety Considerations

The staff evaluation considered the information submitted by the licensee, past operating history recorded in annual reports submitted to the Commission by the licensee, reports by the Commission's Office of Inspection and Enforcement, and onsite observations. The principal safety matters reviewed for the UCS reactor and the conclusions reached follow.

(1) The design, testing, and performance of the reactor structures, systems, and components important to safety during normal operation are inherently safe, and safe operation can reasonably be expected to continue.

(2) The expected consequences of a broad spectrum of postulated credible accidents have been considered, emphasizing those likely to cause loss of integrity of fuel-element cladding. The staff performed conservative analyses of the most serious credible accidents and determined that the calculated potential radiation doses outside of the reactor building are small fractions of 10 CFR 20 allowable doses in unrestricted areas.

(3) The licensee's management organization, operator training, conduct of operations, and security measures are adequate to ensure safe operation of the facility and protection of special nuclear material.

(4) The systems provided for control of radiological effluents can be operated to ensure that releases of radioactive wastes from the facility are within the limits of 10 CFR 20 and are as low as is reasonably achievable (ALARA).

(5) The licensee's Technical Specifications, which provide limiting conditions for the operation of the facility, are such that there is a high degree of assurance that the facility will be operated safely and reliably.

(6) The UCS facility is funded within the annual budget of the Union Carbide Corporation, a multi-billion dollar corporation. The staff concludes that sufficient funds will always be available for the safe operation of the reactor facility.

(7) The licensee's program for providing for the physical protection of the facility and its special nuclear material comply with the applicable requirements in 10 CFR 73.

(8) The licensee's procedures for training its reactor operators and the plan for operator requalification are adequate; they give reasonable assurance that the reactor facility will be operated competently.

(9) The UCS reactor facility's Emergency Plan was submitted separately by letter and report dated September 3, 1982, as part of the current NRC requirements that were published in the Federal Register (47 FR 88) May 6, 1982. The staff's evaluation of the plan will be reported in a supplement to this report (see Section 13.6).

1.2 History

Construction of the UCS reactor started in 1957 when a construction permit was issued by the Atomic Energy Commission (AEC). The reactor first went critical on September 1, 1961, and began full-power operation at 5 MWt on March 3, 1962. The USC reactor has been operating routinely at 5 MWt since that time. The reactor is operated 24 hours per day, with routine shutdowns for refueling, fuel rearranging in the core, and core maintenance. Operating experience indicates the reactor is on-line approximately 90% of the time. Through December 1983, the reactor had accumulated approximately 662,000 MW-hours operation. By letter dated May 23, 1980, UCS requested a 20-year extension of the existing operating license.

1.3 Reactor Facility Description

The UCS 5-MW pool-type research reactor is a light-water moderated, heterogeneous reactor, which uses Materials Testing Reactor (MTR)-type fuel. Demineralized water is used for cooling and shielding and moderation, and demineralized water and graphite are used as a reflector. The reactor core is immersed in either section of a two-section concrete pool. One section of the pool contains an experimental stall into which beam tubes and other experimental facilities converge. The other section is an open area permitting bulk irradiation. A 12-ft deep canal connects the pool with a hot laboratory located in an adjacent building.

A manually operated bridge spans the pool. The bridge supports a suspended aluminum tower, which contains the reactor core, control rods, and control rod drives. Heat, created by the nuclear reaction in the pool, is dissipated by convection at power levels of 250 kW and less and by a forced circulation cooling system for higher power levels of operation.

The primary coolant system consists of demineralized water that is recirculated in a predominantly aluminum system. The water is pumped through stainless steel heat exchangers where it is cooled by water before re-entering the core. The secondary side water transfers its heat to the atmosphere by means of evaporation from a cooling tower located outside the building.

The reactor operates at 5 MWt for 24 hours a day in 14-day cycles with routine shutdown to replenish and rearrange fuel and to perform maintenance.

1.4 Design and Facility Modifications

The only major modification is the replacement of the boron control rods with silver, indium, cadmium control rods; the current control rods also have a thicker surface plating of tin-nickel and a strengthened rod-to-piston connection. Other modifications were in the course of normal maintenance and/or replacement.

1.5 Operation

From initial startup through 1983, the reactor had accumulated in excess of 662,000 MW-hours of operation. The requirements for its radiochemicals and radiopharmaceutical products dictate operating cycles of 10 24-hour days. Routine shutdowns are scheduled between cycles for partial refueling. The reactor is normally on line at least 90% of the time.

The reactor is used for a wide range of programs by members of the resident staff and research groups of the other companies in the Union Carbide Corporation. A list of the general type of work done with the reactor includes

(1) testing of reactor materials
(2) preparation of radioisotopes
(3) activation for wear studies and other nondestructive tests
(4) neutron activation analysis
(5) chemical research

However, the principal use of the reactor is to produce radiochemicals and radiopharmaceuticals. As indicated in Table 1.1, the UCS reactor produces approximately 50% of those products used in the listed procedures.

Table 1.1 Union Carbide, nuclear products production and utilization

Type of test	Total 1980 test procedures	UCS contribution(%)
Nuclear Medicine		
Lung inhalation	320,000	30
Liver	1,425,000	50
Bone HMDP	1,615,000	50
Blood pool	100,000	50
Kidney	456,000	50
Lung perfusion	775,000	50
Brain thyroid	500,000	50
Total	5,181,000	2,500,000
Radioimmunoassay		
Thyroid hormones	48,000,000	50
Digoxin	14,000,000	50
Human chornionic gonadotropin	11,000,000	50
B-12/folate	6,000,000	50
I GE immune test	6,000,000	50
Cortisol	4,000,000	50
Estriol	4,000,000	50
Angiotensin	1,500,000	50
Digitoxin	1,400,000	50
Methotrexate	120,000	50
Others	25,000,000	50
Total (approximate)	120,000,000	60,000,000

1.6 Shared Facilities and Equipment

The reactor building is constructed of reinforced concrete and situated partially below grade. It is attached to a hot laboratory complex designed primarily to permit disassembly, inspection, testing, and analysis of highly radioactive material. Irradiated samples are transferred to the laboratory by way of a canal that runs through the reactor building into the hot laboratory. The hot laboratory has five hot cells to perform the various steps involved in handling and processing of the highly radioactive material. Both the reactor and hot laboratory facilities share a common main exhaust system. A single exhaust stack receives discharge from both facilities.

In addition, two emergency power diesel generators are shared, albeit each one is principally assigned for either the reactor or hot laboratory emergency requirements. In the event of a power failure, both generators automatically start up. Utilities such as municipal water and sewage, natural gas, and electricity are provided to the reactor hot laboratory complex by the local utilities.

1.7 Comparison With Similar Facilities

The UCS reactor is typical of the MTR-fuel, pool-type reactors that are based on the Oak Ridge National Laboratory Bulk-Shielding-Reactor (BSR) design, and is one of many pool-type research reactors that have been in use in the United States for about 30 years. Though the reactor has higher power than most other reactors of the same type, it is similar in design and operation to other NRC-licensed, pool-type reactors in the United States that utilize highly enriched MTR-type fuel. Several of these BSR-type reactors have been shut down and/or dismantled; however, their total accumulated operating experience is about 340 years. If Department of Energy research reactors based on the BSR design are also included, the total operating experience would be about 1,000 reactor-years.

1.8 Nuclear Waste Policy Act of 1982

Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982 provides that the NRC may require, as a precondition to the issuance or renewal of an operating license for a research or test reactor, that the licensee shall have entered into an agreement with the Department of Energy (DOE) for the disposal of high-level radioactive waste and spent nuclear fuel. UCS has an agreement with DOE to ship spent fuel to DOE facilities or designated receivers of the fuel, and therefore, is in conformance with the Nuclear Waste Policy Act.

2 SITE CHARACTERISTICS

2.1 Site Description

The Union Carbide Subsidiary B Inc., Medical Products Division, nuclear reactor facility is located within the city of Tuxedo, in Orange County, New York. Orange County, in southeastern New York State, is bordered on the south by New Jersey and is approximately 40 mi northwest of New York City. Tuxedo, in the extreme southeastern corner of Orange County, is approximately 4 mi north of the New Jersey state line. The plant site is located on 100 acres of land, owned by Union Carbide. It is in an industrial park area known as Sterling Forest and is about 3-1/4 mi northwest of the village of Tuxedo Park. Figure 2.1 is a map of the area surrounding the site. The plant itself, constructed along Long Meadow Road on the eastern slope of Hogback Mountain, is at an average elevation of 800 ft above mean sea level (MSL). A layout of the UCS complex is shown in Figure 2.2.

The five principal buildings at the plant site are

Building 1	Reactor
Building 2	Hot Laboratory (structurally joined to the reactor building)
Building 3	Maintenance
Building 4	Administration
Building 5	Heating Plant

There is also a small concrete block structure at the north end of the plant site used for temporary storage of drummed, miscellaneous low-level radioactive wastes.

2.2 Geography

The UCS reactor site is within a 22,000-acre woodland area called Sterling Forest, which is owned by a private development company. Sterling Forest contains three residential areas, several small research centers, the UCS facility, and a conference center. These developed areas make up a total of less than 1,500 acres. The remainder of the land is undeveloped. Adjoining Sterling Forest to the east is another large undeveloped area that is a part of the Palisades Interstate Park System. This 75,000-acre woodland contains approximately 31 summer camps, but essentially no year-round residents.

The approximately 20,500-acre undeveloped portion of Sterling Forest is managed ecologically by the Sterling Forest Development Corporation. This organization permits regulated (license and bag-limit) hunting in designated, marked portions of the area.

Regulated fishing also is permitted in designated lakes on the property. The Sterling Forest Development Corporation employs regulated lumbering, the main objective of which is to remove dead and disease-infested trees and promote maintenance of healthy understory.

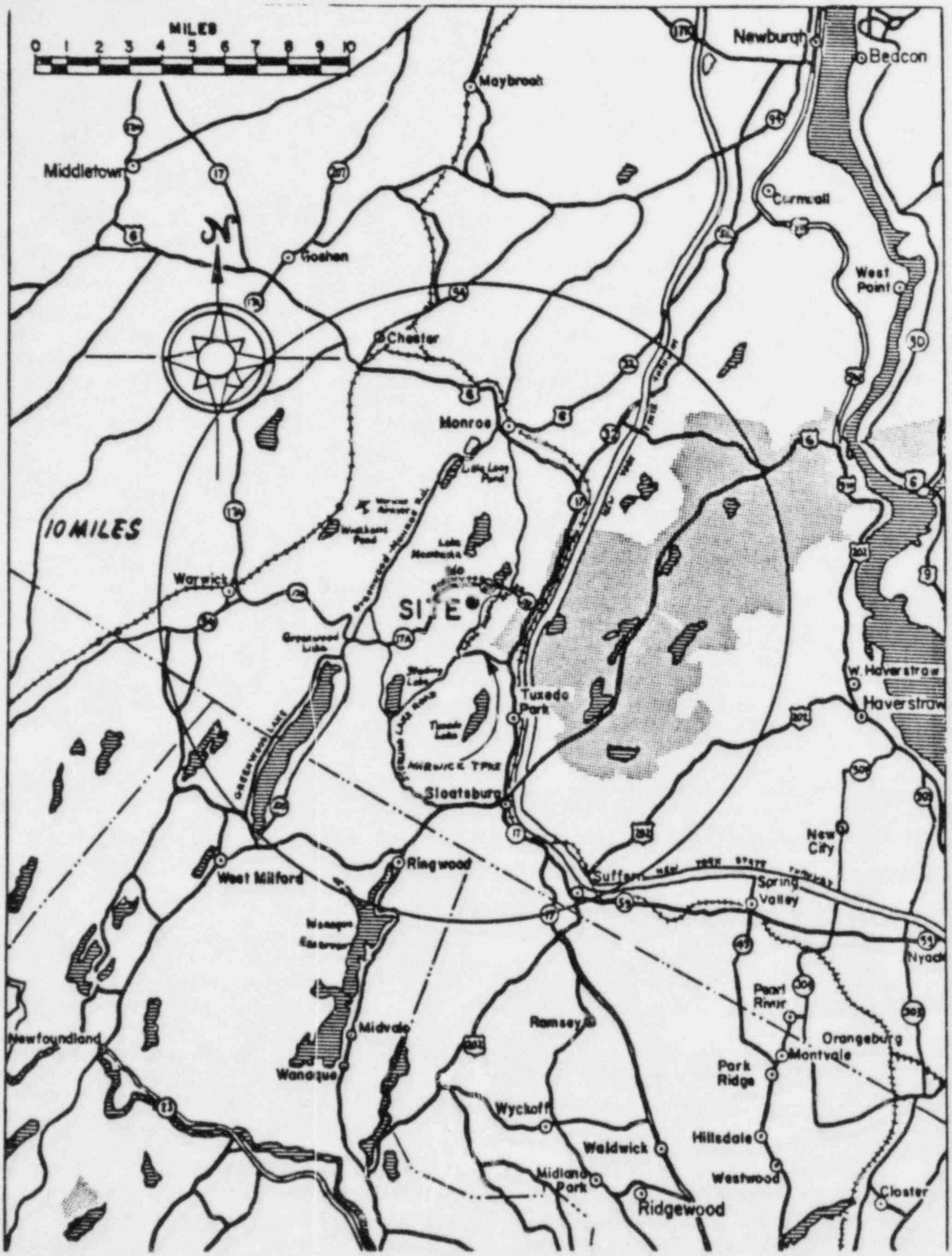

Figure 2.1 10-mile radius map of site

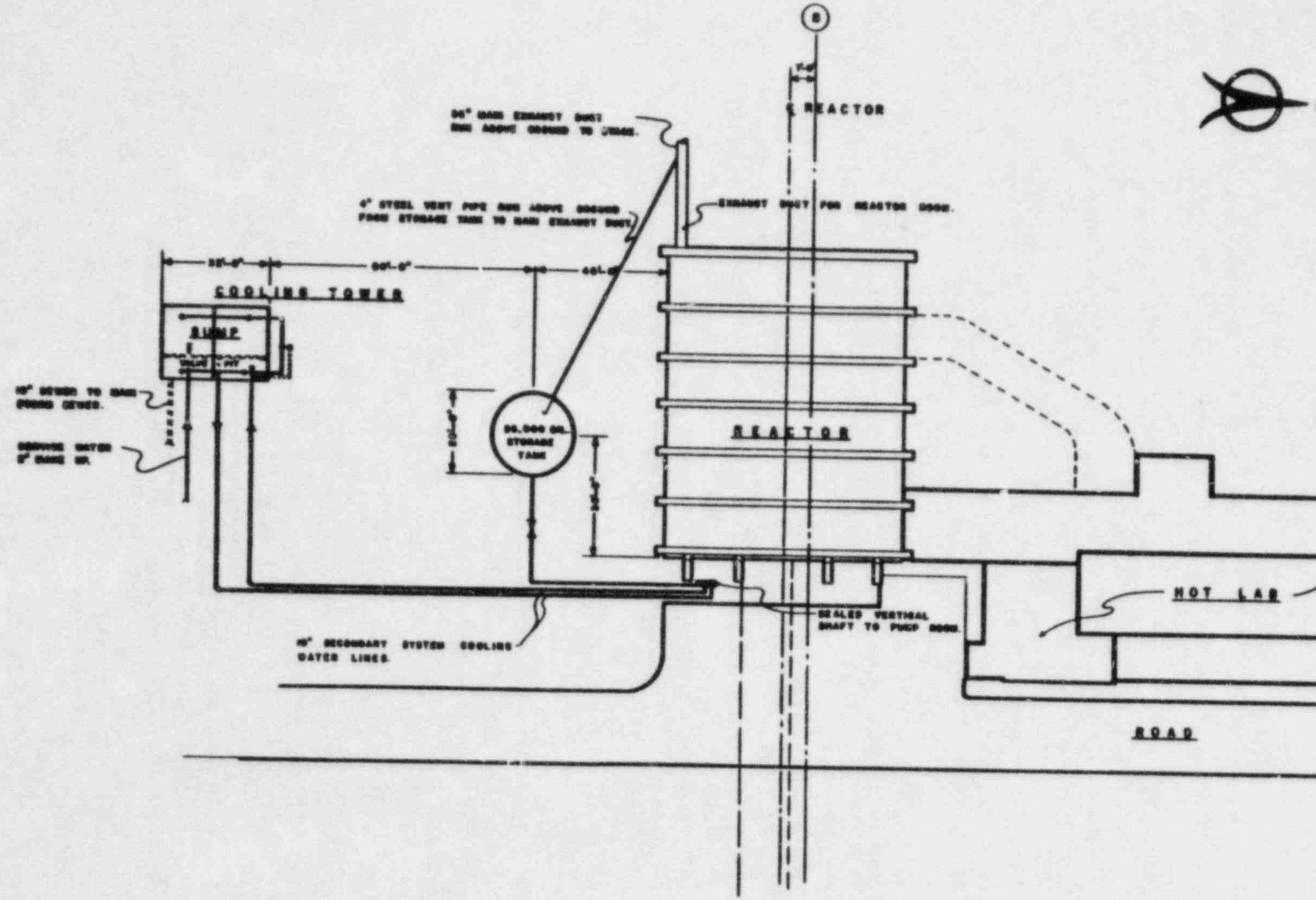

Figure 2.2 Reactor building-site plan

2.3 Topography and Surface Drainage

The reactor site is about 1,500 ft southwest of Indian Kill Brook, a small stream flowing southeast for a mile and a half to the Ramapo River. The plant borders Long Meadow Road at an elevation of approximately 800 ft.

There is a very low north-south topographic divide between Indian Kill Brook drainage and drainage of Warwick Brook to the south, which also flows east to the Ramapo River. These two small streams (Indian Kill Brook and Warwick Brook) drain into the Ramapo River from the vicinity of the site and thus dominate the surface drainage pattern away from the site.

Although the relief in the area is only 400 to 700 ft from valley floors to ridge tops, the hillsides are considered to be steep and rugged. From a past era of glaciation, the area features clogged drainage systems such as swamps, ponds, and lakes along stream channels. Fill, clay, sand, gravel, and boulders of every size also strew the hillsides. The reactor building is located at the eastern toe of the north trending spur of Hogback Mountain, which slopes from an elevation over 1,500 ft down to the level of Indian Kill Lake at an elevation of 700 ft.

As stated above, surface drainage from the site is exclusively by way of Indian Kill Brook. Indian Kill Brook enters the Ramapo River 1-1/2 mi east of the plant at el 463 ft. Tuxedo Lake stands at el 560 ft. Wee Wah, the adjoining lake to the north, stands lower than Tuxedo Lake to which it is joined by a small stream of high gradient. Wee Wah Lake consists of two segments. The southern, higher segment is separated from the northern, lower segment by a stream of steep gradient. This northern segment, in turn, discharges over an earth dam and masonry spillway to a small stréam that discharges into Ramapo River. Thus, if Indian Kill Brook were contaminated as a result of some incident, contamination of this chain of three lakes by surface flow from the plant would not be possible.

2.4 Demography

The UCS plant is located in a thinly populated area. The closest occupied offsite area is the Laurel Ridge housing development which contains 132 houses at a minimum distance of 1,100 ft east of the reactor building. A second development, consisting of 27 houses in an area called Clinton Woods, is located 3,200 ft to the north. There are no other housing developments within 1.5 mi of the reactor.

Table 2.1 shows the population distribution in 22.5° compass sectors out to 50 mi. The north sector is centered on true north but includes 11°15' on either side of true north, a total of 22.5°. Likewise, all other sectors embrace an arc of 22.5°. The table indicates the most heavily populated areas to be to the southeast, south-southeast, and south of the site within the 20 to 50 mi radii. The high population density of these sectors is a result of the large metropolitan centers, e.g., New York City, Newark, and Bayonne.

Table 2.1 Population density around the Union Carbide reactor by 16 compass sectors out to 50 miles*

	Miles					
Sector	0 - 5	5 - 10	10 - 20	20 - 30	30 - 40	40 - 50
N	614	16,851	36,264	29,237	23,692	19,458
NNE	346	1,903	12,958	53,101	104,808	48,873
NE	108	1,081	14,135	26,503	31,023	26,850
ENE	187	10,489	47,176	43,204	50,442	183,112
E	107	17,330	61,749	30,032	76,475	84,156
ESE	330	7,481	15,506	49,112	275,267	127,105
SE	132	18,868	72,622	476,642	132,749	1,753,651
SSE	3,878	11,829	142,070	454,282	2,296,153	2,334,641
S	91	10,892	96,848	499,486	971,105	2,611,407
SSW	43	10,393	42,750	124,960	133,923	133,281
SW	0	2,951	18,238	60,235	97,985	45,567
WSW	125	2,834	27,381	20,741	30,893	35,991
W	68	23,192	10,754	19,855	13,179	7,900
WNW	878	2,195	8,022	21,651	8,775	7,443
NW	192	2,196	28,510	8,858	6,016	17,724
NNW	190	2,108	18,339	8,964	18,853	7,908

*Population estimates based on 1980 Census of Population and Housing.

2.5 Nearby Industrial, Transportation, and Military Facilities

As stated earlier, UCS is located about 3 1/4 mi from Tuxedo Park, New York, in an industrial park. Other tenants in the industrial park are light industry. The closest major highway or railroad is about 1.5 miles from the plant.

The closest military installation is West Point, about 10 mi from UCS. There are no commercial airports closer than 40 mi from the plant and the nearest private airport is 5.2 mi from the UCS plant. Stewart Airport, both a commercial and military airport, is located approximately 16 mi away on the outskirts of Newburgh, New York.

As none of the above industrial transportation, or military activities occur close to the reactor, the staff concludes that these activities pose no threat to the safe operation of the UCS reactor.

2.6 Meteorology and Climatology

The climate of the Sterling Forest area is predominantly influenced by air mass movements and prevailing winds from an inland direction. Cold frontal weather moves across the area from west to east at average velocities of 30 to 35 mph in winter and considerably more slowly in summer. This is a part of the normal cyclonic circulation in which high- and low-pressure systems follow paths toward the northeastern United States. About 40% of the low centers pass over or close to southeastern New York so that there is regular change in weather patterns without any consistent periods of stagnation.

Centers of high pressure alternate more or less regularly with the lows. In the wintertime, their movement is variable, depending on the strength of cold air outthrusts from the arctic area to the northwest. This movement is slowest during summer and early fall so that, with the prevailing westerlies aloft reaching their most northerly movement at the same time, high-pressure centers can become stationary for a few days during these seasons. During a 2-year period of inversion observation, only 21 inversions persisted for more than 12 hours and only 6 persisted longer than 24 hours.

Mean ambient air temperatures vary from 28°F (minus 1°C) in January, to 75°F (24°C) in July, with extremes of -19°F (-30°C) and 105°F (41°C). Precipitation is fairly uniform throughout the year with an average rainfall of about 44 in.

Dispersion values, χ/Q, from field sampling and measurements were obtained at various atmospheric monitoring points and from the reported annual releases of I-125 and I-131. These were used to compare against calculated concentrations of the particular isotopes at the corresponding points. In all cases, the predicted value exceeded the measured value. Additional calculations were performed to investigate the local topographical effects on the χ/Q values.

2.7 Geology

The UCS plant site is located within a seismogenic zone trending along the Ramapo Fault System. The Ramapo Fault System and other fault systems within the zone encompass the Manhattan Prong, Newark Basin, and the Hudson Highlands. They are made up of semiductile thrust and strike-slip faults of Precambrian and early Paleozoic age, and brittle faults with dip-slip and oblique sense of motion, generally of Mesozoic and younger age. The younger faults show evidence of long periods of recurrent movement. The seismogenic zone is approximately 30 km (18 mi) wide and is centered on the northeasterly striking, southeasterly dipping Ramapo Fault System. The hypocentral depths are from 0 to 10 km (6 mi) and the focal mechanisms indicate either reverse or right lateral strike-slip motion on the northeast striking faults.

On the basis of the available data from recent investigations and mapping of the Ramapo Fault System, there is no evidence of recent movement. Many of the investigations by the U.S. Geological Survey and by consultants for the Indian Point Nuclear Power Plant applicant were conducted for the specific purpose of determining the age of deformation in the vicinity of the Ramapo Fault System and whether or not there is evidence of geologically old fault activity. The staff concluded that the Ramapo Fault System should not be considered capable within the meaning of 10 CFR 100, Appendix A.

The plant is situated on a northerly trending spur of Hogback Mountain along Long Meadow Road. Bedrock is highly metamorphosed and consists of very dense, hard Precambrian granite gneiss that is fractured near the surface. Drill holes in the area produced nearly complete core recoveries and ground water was measured at a depth of 85 ft below the surface.

2.8 Hydrology

As stated in Section 2.3, the surface water features of significance in connection with the operation of the UCS reactor are the Indian Kill Reservoir, Indian Kill Brook, Warwick Brook, and the Ramapo River.

Surface drainage from the site is exclusively by way of Indian Kill Brook, but because of the unique surface hydrology, even if the Indian Kill Brook were contaminated, it is not remotely possible to carry such contamination by surface flow to any of this chain of three lakes (Section 2.3).

Indian Kill Brook presents the only obvious path for contamination by underground flow, that is, through alluvial sand, silts, and gravels that lie beneath the stream channel, resting on the gneissoid bedrock of the region. Water passes downstream easily but slowly through these alluvial deposits. Such waters could not possibly ascend into the chain of Tuxedo Lakes. Furthermore, it does not seem possible that water could pass underground beneath the mountainous ridges, through the fractures in the hard rocks. The mountainous tract, which is bounded by Indian Kill Brook, Long Meadow Road, Warwick Brook, and Ramapo River, naturally contains some ground water within fractures in the rocks. But this water drains outward to the nearest and most accessible exists, namely either Indian Kill Brook, Warwick Brook, or Ramapo River. Water cannot pass against this outward flow, across this mountainous tract, and even assuming it could, it could not pass the boundary of Warwick Brook, which flows east to Ramapo River. Therefore, the possibility of contamination of the Tuxedo Lakes chain by water from the vicinity of the plant may be dismissed. The only reasonable route for contaminated liquid effluents that might come from the plant site would be via Indian Kill Brook to the Ramapo River and thence to the Passaic River in New Jersey.

2.9 Seismology

Recent studies in New York State and adjacent areas have brought to light some of the geotectonic features that account for the seismicity in the region. Aggarwal and Sykes (1978) conclude that earthquakes occur predominantly along northeast trending faults of which the Ramapo fault is one such fault. However, in a more recent study (1983), Kafka concludes that earthquakes recorded from 1970 to 1982 by the microearthquake networks in the New York City metropolitan area do not corroborate evidence that northeast trending faults lying to the northwest of the Newark Basin (such as the Ramapo Fault) are any more active than those lying to the north and east of the basin. Woodward-Clyde Consultants (1982) report that earthquakes instrumentally located in the region do not lie preferentially along either the Ramapo Fault or along other northeast trending structures. No spatial correlation is observed between the distribution of epicenters and geologic structures or terranes that are mapped at the surface. The two largest events since the local seismic network has been operating in the New York metropolitan area were not associated with the Ramapo Fault but rather were located in the Coastal Plain east of the Newark Basin (Cheesequake, New Jersey) and in the Valley and Ridge Province north of the Newark Basin (Wappingers Falls, New York). Epicenter locations of magnitude 2 and larger earthquakes appear to be in the region surrounding the Newark Basin. Although the Ramapo Fault shows a spatial correlation with some of the earthquakes, fault plane solutions for many of these events indicate primarily thrust-type faulting on north-to-west striking planes, which is inconsistent with movement on the Ramapo Fault (Woodward-Clyde, 1982).

A relevant observation made by Kafka (1983) is that similarities between the distribution of seismicity of the recently (instrumentally) recorded earthquakes and the distribution of more than 200 years of historic earthquakes suggest that the seismic activity in this area has been relatively stationary over the last few hundred years.

The highest intensity reported for historic earthquakes in the area is Modified Mercalli Intensity (MMI) VII (1737, 1884). Kafka (1983) estimated the 1884 earthquake epicentral magnitude to be 4.9 (m_n). However, both the 1737 and 1884 earthquakes occured east of the Newark Basin and, as Kafka points out, the 1884 earthquake may have occurred offshore. There are, however, several earthquakes that occurred within the Ramapo Fault zone that have magnitudes estimated to be in the 4.7-4.8 range (Aggarwal and Sykes, 1978).

2.10 Conclusion

The staff has reviewed and evaluated the UCS reactor site and contiguous regions for natural and manmade hazards and concludes that there are no risks associated with the site that make it unacceptable for the continued operation of the reactor at the power level of 5 MW.

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Reactor Building

The reactor building is a 70x92x57-ft-high (from the beam hole floor) reinforced-concrete structure set into an excavation in the side of the adjacent rock mountain (see Figure 3.1). Shielding and containment are provided on three sides of the building: by the native mountain rock against the west wall, and a combination of rock and fill on the north and south sides. The exposed portions of the walls and roof are reinforced concrete with a minimum thickness of 12 in. and 8 in., respectively. The volume of the reactor building is about 285,000 ft^3. The building is designed to withstand an internal negative pressure of 3/4 psig.

The experimental area around the reactor is serviced by a 10-ton bridge crane traveling the length of the building. The reactor control room, several offices, and laboratories for low activity work are provided inside of the reactor building. All personnel entrances to the building are of the double-airlock type. When required, equipment can be brought into the reactor building through a motor-operated, air-tight sliding door which is locked during reactor operation.

3.2 Wind Damage

As indicated in Section 3.1, the reactor building is constructed into a hillside on three sides, with the fourth side constructed of 12-in. concrete walls and the roof constructed of 8-in.-thick (minimum) concrete. Occurrences of tornadoes and hurricanes are rare in the general area because of the ruggedness of the topography. Tornadoes in the immediate vicinity of the complex could be considered to be extremely rare, if not impossible.

On the basis of the above, wind damage to the reactor building is considered to be remote.

3.3 Water Damage

Because of the topography, the only type of water damage that could be postulated is that generated by a flash flood. The building has been designed to withstand flooding without damage. Although flood waters may enter the reactor pool, no release of radiation would result. Release of the flood/coolant waters would be monitored or controlled as indicated in Section 11.1.

3.4 Seismic-Induced Reactor Damage

As described in Section 2, the reactor is classified as being in an area of low seismic intensity. No earthquake with destructive intensity has been recorded within 60 mi of the reactor site. This, plus the fact that the reactor is constructed on hard rock and is outfitted with redundant scram systems, makes it extremely unlikely that any seismic activity could cause damage to the reactor or components sufficient to release any fission products to the environment.

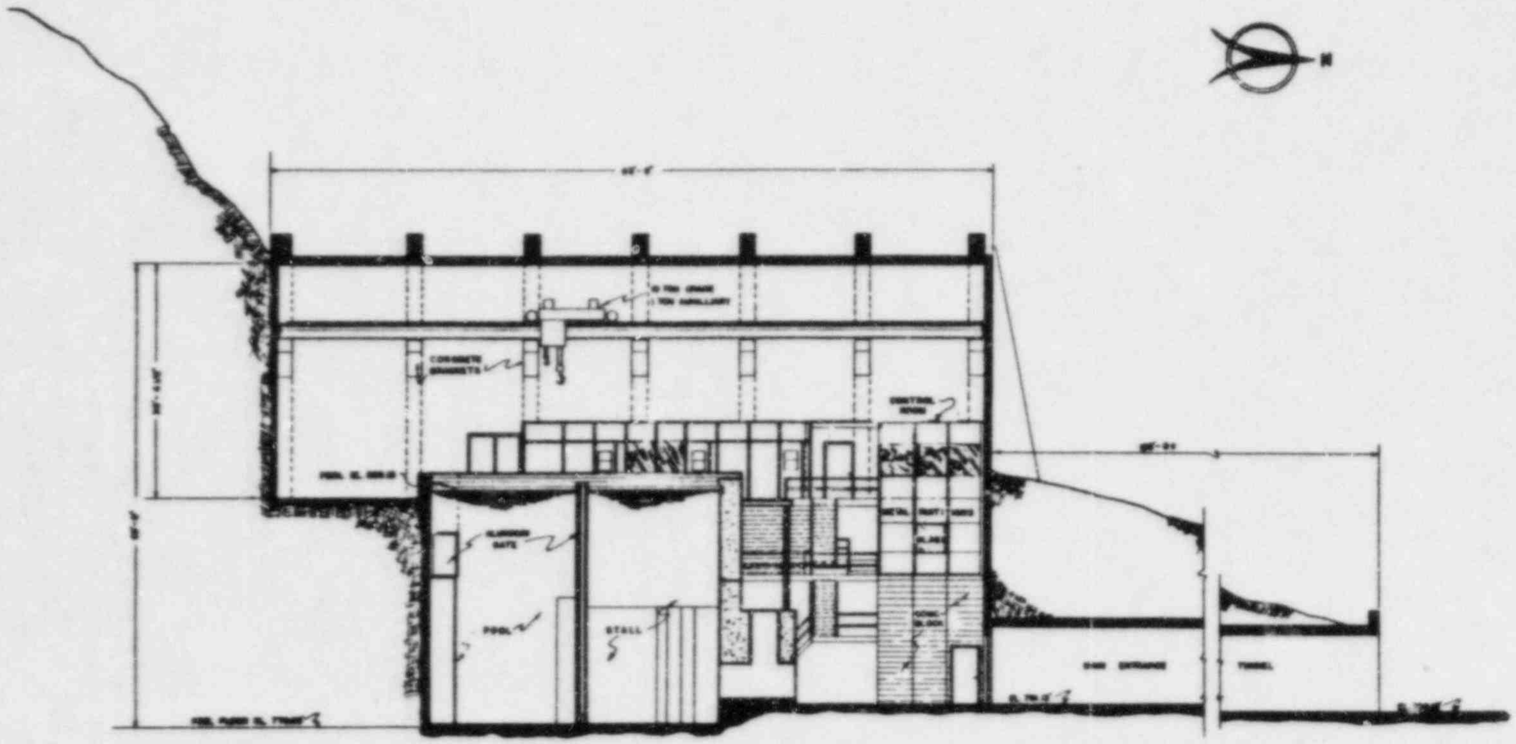

Figure 3.1 Reactor building elevation

3.5 Mechanical Systems and Components

The mechanical systems and components were designed and installed using applicable industry codes and standards. These systems and components have been operating since 1957 with a minimum of problems. By adhering to maintenance schedules and the performance requirements of the Technical Specifications, the mechanical systems and components have been kept in an excellent state of operation.

The staff believes that the same attention and the performance requirements of the Technical Specifications will ensure the mechanical components and systems being maintained at a continuing acceptable level of performance.

3.6 Conclusion

From the above design considerations of the reactor facility, the staff concludes that the UCS reactor facility is adequate to withstand potential wind damage, floods, and potential minor earthquake activity without any significant damage, and, therefore, will not present a significant risk to the public.

4 REACTOR

4.1 General

The UCS nuclear reactor is a pool-type research reactor located in Sterling Forest near Tuxedo, New York, and is licensed to operate at thermal power levels up to and including 5 MW. The reactor is a light-water-moderated, -cooled, and -shielded, water-and-graphite-reflected, solid-fuel reactor. It is typical of a number of NRC-licensed reactors based on the Oak Ridge National Laboratory Bulk Shielding Reactor (BSR) design. The reactor has been in operation since 1961, and is one of 17 pool-type research reactors still in operation. As stated in Section 1, the total accumulated operating experience of the BSR-type reactor is about 340 years at various power levels. Its principal use is for the production of radiochemicals and radiopharmaceuticals for use in medical therapy, research, and commerce.

The reactor has a number of experimental facilities, including six beam tubes, a thermal column, three tangential pneumatic tubes, at least (depending on configuration) nine in-core irradiation positions, and a number of out-of-core irradiation positions. In addition, bulk irradiations can be performed in the pool section of the reactor configuration (see Figures 4.1 through 4.4).

4.2 Reactor Pool and Reactor Core

4.2.1 Reactor Pool

The reactor core, suspended from a movable bridge, is immersed in a 49x23x32-ft-high pool of demineralized water.

The pool is divided into two sections separated by a 4-ft-wide opening that can be closed by a removable watertight gate. The narrower stall section contains the fixed experimental facilities such as the beam tubes and thermal column. The open end of the pool permits bulk irradiations and provides storage space for irradiated fuel and experiments. A 12-ft-deep canal connects the open pool with the hot cells to permit the transfer of irradiated material between the two facilities (see Figures 4.1 through 4.4).

Shielding in the stall area consists of a 5.8-ft-thick magnetite concrete wall extending to a height of 15 ft above the pool floor. The wall thickness is then reduced to 3 ft to the top of the stall. The first 4 ft of the wall above the step is of magnetite concrete and the remainder of regular concrete.

All sections of the pool and stall areas that are in contact with the reactor water are coated for ease of decontamination and to prevent interaction of the reactor water with the concrete. Areas normally exposed to high radiation are coated with glazed ceramic tile. Such areas include the floor of the pool and stall, the sides of stall up to and including the 15-ft step, and the sides of the pool behind the fuel storage racks.

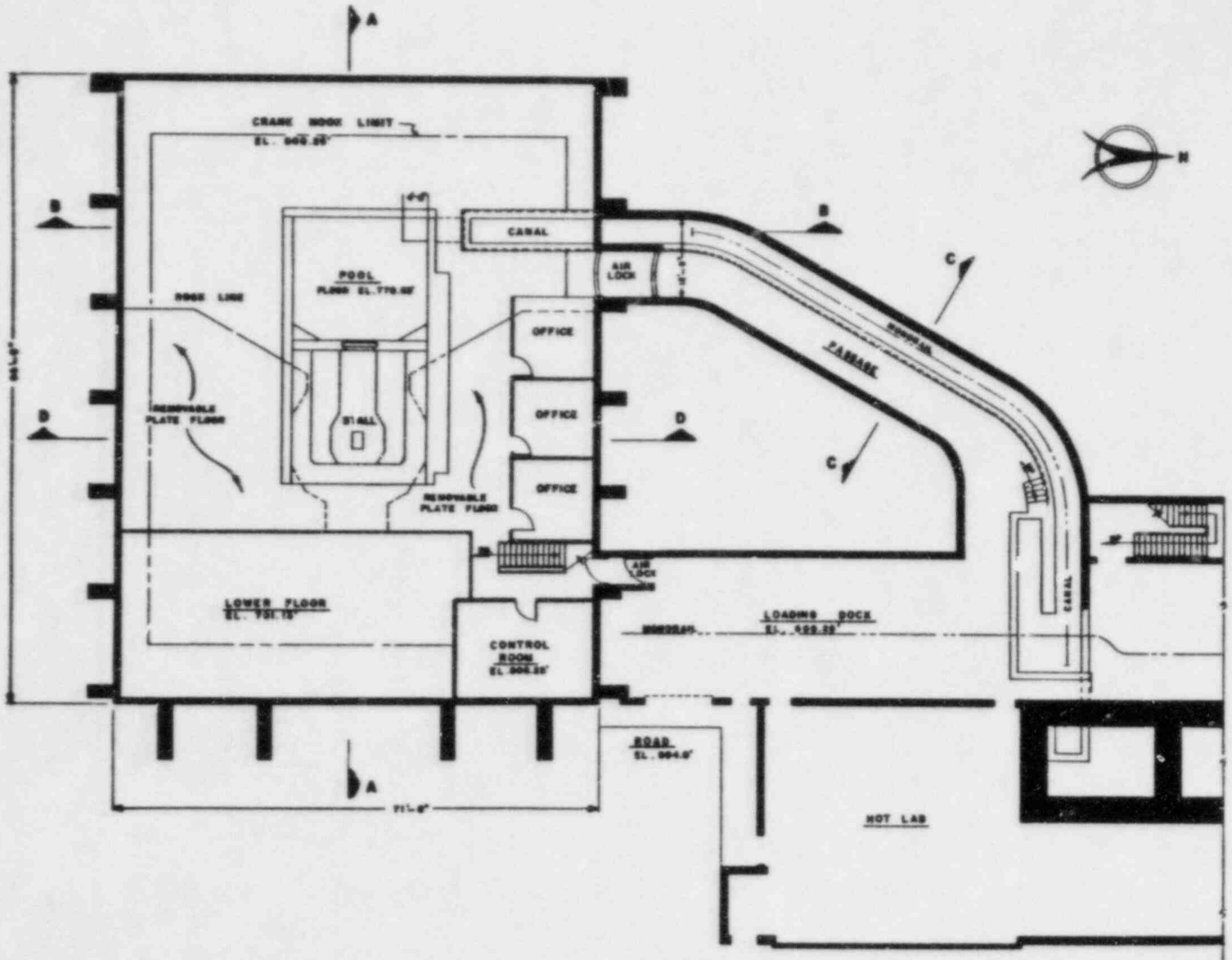

Figure 4.1 Reactor building at elevation 808.25 feet

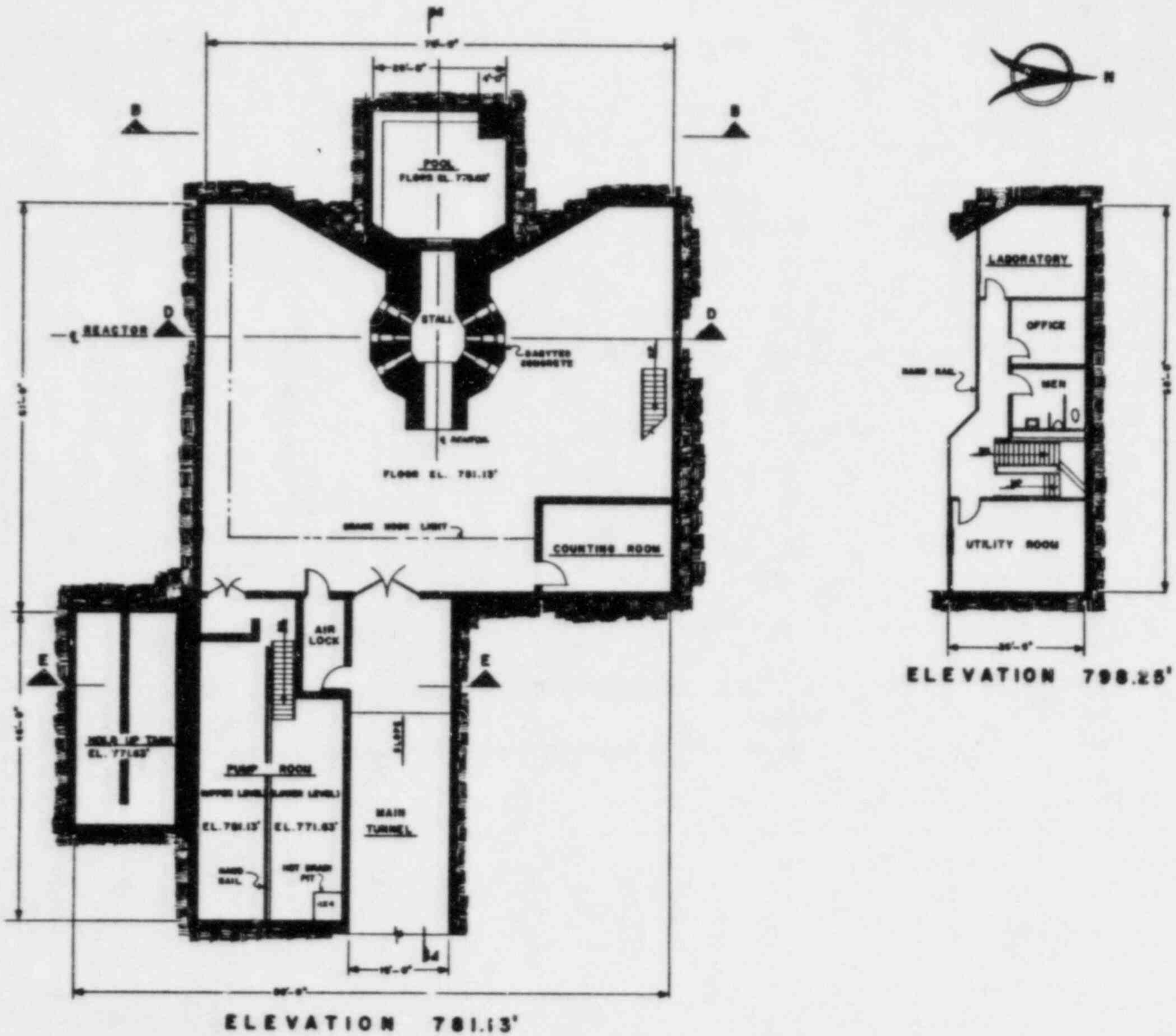

Figure 4.2 Reactor building plan at elevations 781.13 and 798.25 feet

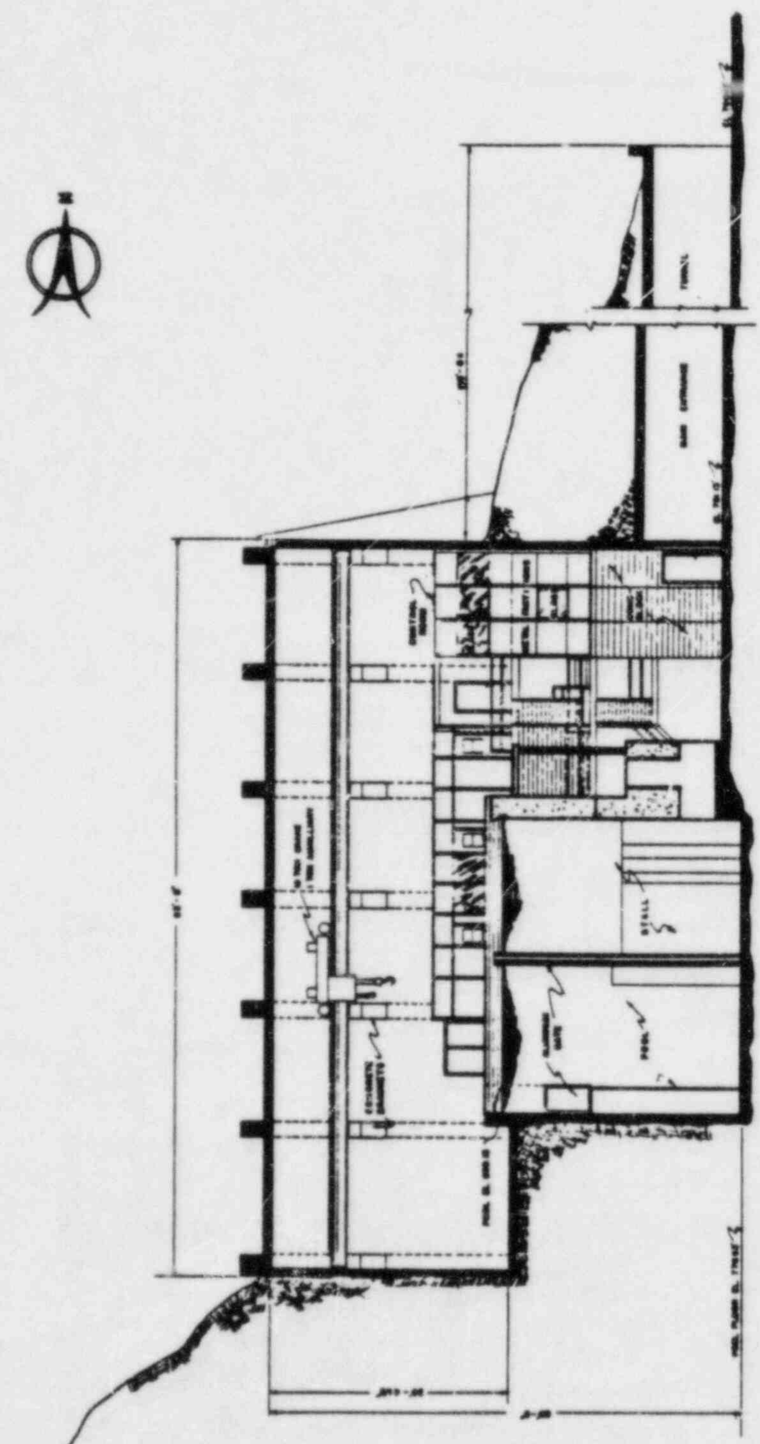

Figure 4.3 Reactor building.

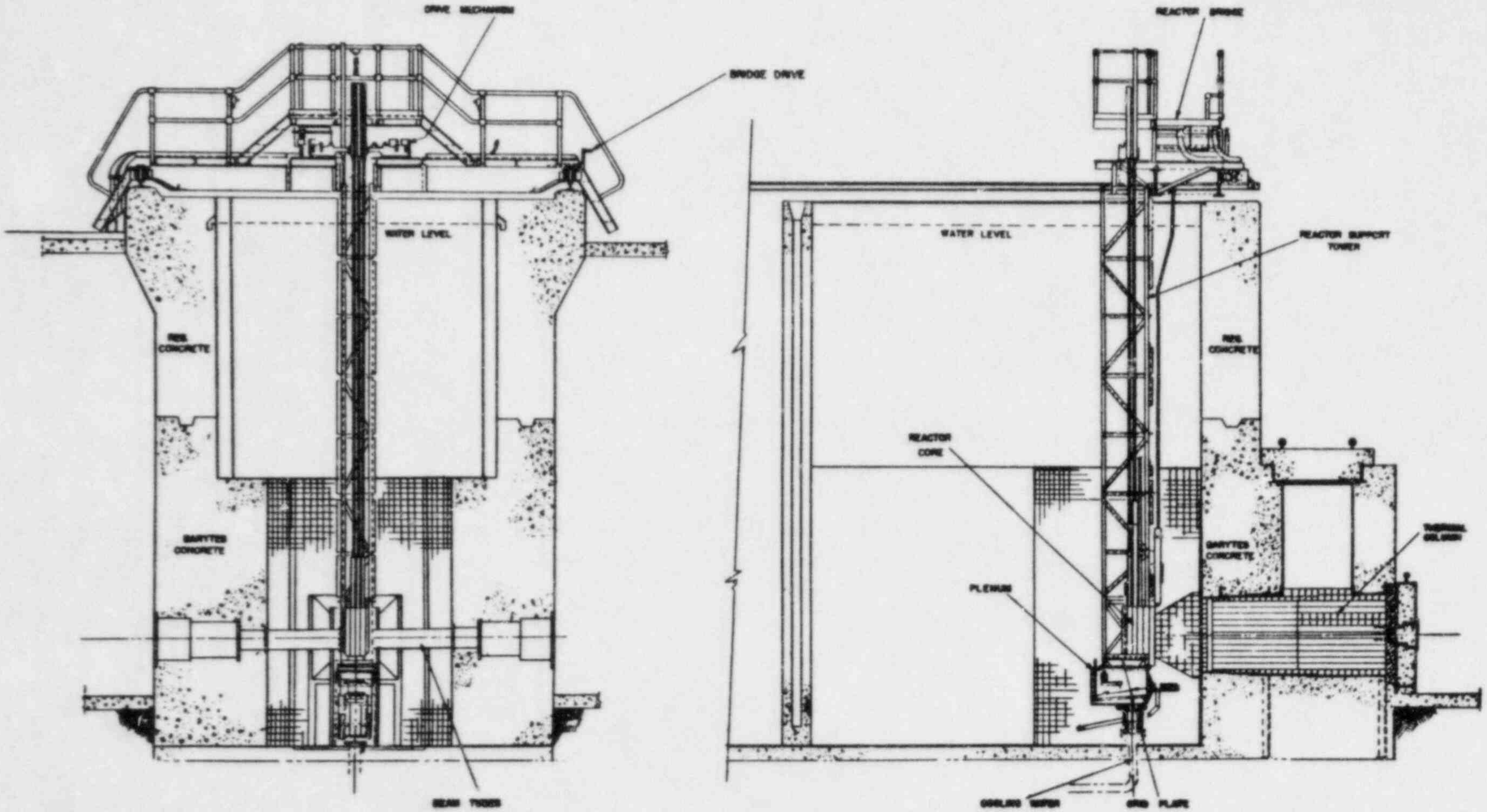

Figure 4.4 Vertical sections through reactor pool

The core support bridge, which is movable on rails mounted on top of the pool walls, is constructed of structural steel. The central section incorporates a superstructure to allow for the mounting of reactor control mechanisms and electrical equipment. The bridge is moved by manual rotation of a crank handle for positioning the reactor in either the pool or stall. A locking device prevents accidental or unauthorized movement of the bridge.

The core support tower, suspended from the bridge, is a structural aluminum frame. The walls of the pool contain six aluminum storage racks. The design is such that a critical array, with or without the core being considered, cannot be achieved with elements stored in the racks.

4.2.2 Reactor Core

The reactor core is composed of MTR-type fuel in assemblies inserted in the grid plate, the control-rod fuel assemblies with built-in control-rod guides, graphite reflector elements, and sample irradiation stringers. The elements may be arranged in a variety of lattice patterns depending on experimental requirements.

As shown in Figure 4.5, the grid plate has 54 holes arranged in a 9x6 pattern that can accept fuel elements, control rod fuel assemblies, and reflector elements, plus 40 additional smaller holes located between the fuel element holes to provide required cooling for the various fuel element and experiment arrays. Following placement of the fuel elements and experiments, the unused holes are plugged to divert flow to the core elements and experiments. The current core has 31 full fuel elements, 6 control elements, 5 control rods, and 1 regulating rod. Total fuel in the core is approximately 5 kg of ^{235}U and is dependent on the fuel cycle of the specific core.

4.3 <u>Fuel Elements</u>

4.3.1 Standard Fuel Element Assembly

Each standard fuel element assembly is composed of four major components--the unfueled-aluminum side plates, fuel plates, the lower end fitting, and fuel handling (see Figure 4.6). The two side plates retain the fuel plates in an approximately 3x3-in. assembly. A horizontal rod fastened between the side plates near the upper end of the fuel assembly serves as a handle for the insertion or withdrawal of the assembly from the grid plate. A standard fuel element assembly has 16 fueled plates. The fuel plates are 24 5/8 in. long and fabricated of enriched (93%) uranium-aluminum alloy fuel "meat" sandwiched between high-purity aluminum cladding. Each fuel plate is formed into a convex shape to minimize thermal stress and is fastened to the side plates by swaging. An end fitting, machined to fit into the grid plate, is attached to the lower end of the fuel plate assembly. The nominal fuel content of the element is 196 g ^{235}U at a nominal content of approximately 20 w/o.

4.3.2 Control-Rod Fuel-Element Assembly

Six of the elements composing the reactor core are special control-rod fuel-element assemblies. These assemblies, like the partial fuel element assembly, contain 110 g ^{235}U in nine fueled plates. These control rod

A1 ++++++ 0.0 0.000	B1 ++++++ 0.0 0.000	C1 ++++++ 0.0 0.000	D1 ++++++ 0.0 0.000	E1 ++++++ 0.0 0.000	F1 ++++++ 0.0 0.000
A2 ++++++ 0.0 0.000	B2 UC 452 113.8 0.005	C2 UC 439 107.5 0.008	D2 UC 450 132.4 0.009	E2 UC 423 110.3 0.008	F2 UC 451 109.9 0.005
A3 UC 429 103.5 0.007	B3 ++++++ 0.0 0.003	C3 UC 437 111.0 0.010	D3 UC 448 103.6 0.013	E3 ++++++ 0.0 0.008	F3 UC 455 123.3 0.007
A4 UC 391 109.7 0.009	B4 UC 82S 66.9 0.016	C4 UC 424 149.5 0.017	D4 UC 441 161.6 0.016	E4 UC 72S 86.5 0.018	F4 UC 447 103.7 0.010
A5 ++++++ 0.0 0.000	B5 UC 460 150.7 0.015	C5 UC 482 176.0 0.022	D5 UC 481 183.3 0.022	E5 UC 471 166.8 0.015	F5 ++++++ 0.0 0.000
A6 UC 440 116.7 0.010	B6 UC 83S 96.5 0.020	C6 UC 479 174.5 0.022	D6 UC 462 184.1 0.023	E6 UC 75S 93.6 0.020	F6 UC 443 117.4 0.012
A7 ++++++ 0.0 0.000	B7 UC 421 147.2 0.018	C7 UC 475 180.5 0.020	D7 ++++++ 0.0 0.000	E7 UC 422 153.1 0.017	F7 ++++++ 0.0 0.000
A8 UC 431 122.1 0.011	B8 UC 79S 67.2 0.018	C8 UC 425 157.6 0.016	D8 UC 468 161.8 0.018	E8 UC 74S 72.2 0.018	F8 UC 442 113.9 0.009
A9 UC 427 118.6 0.009	B9 ++++++ 0.0 0.010	C9 UC 435 104.2 0.011	D9 ++++++ 0.0 0.012	E9 UC 433 113.4 0.011	F9 ++++++ 0.0 0.000

Legend: ○ = control rod

⬡ = regulating rod

Figure 4.5 Core configuration

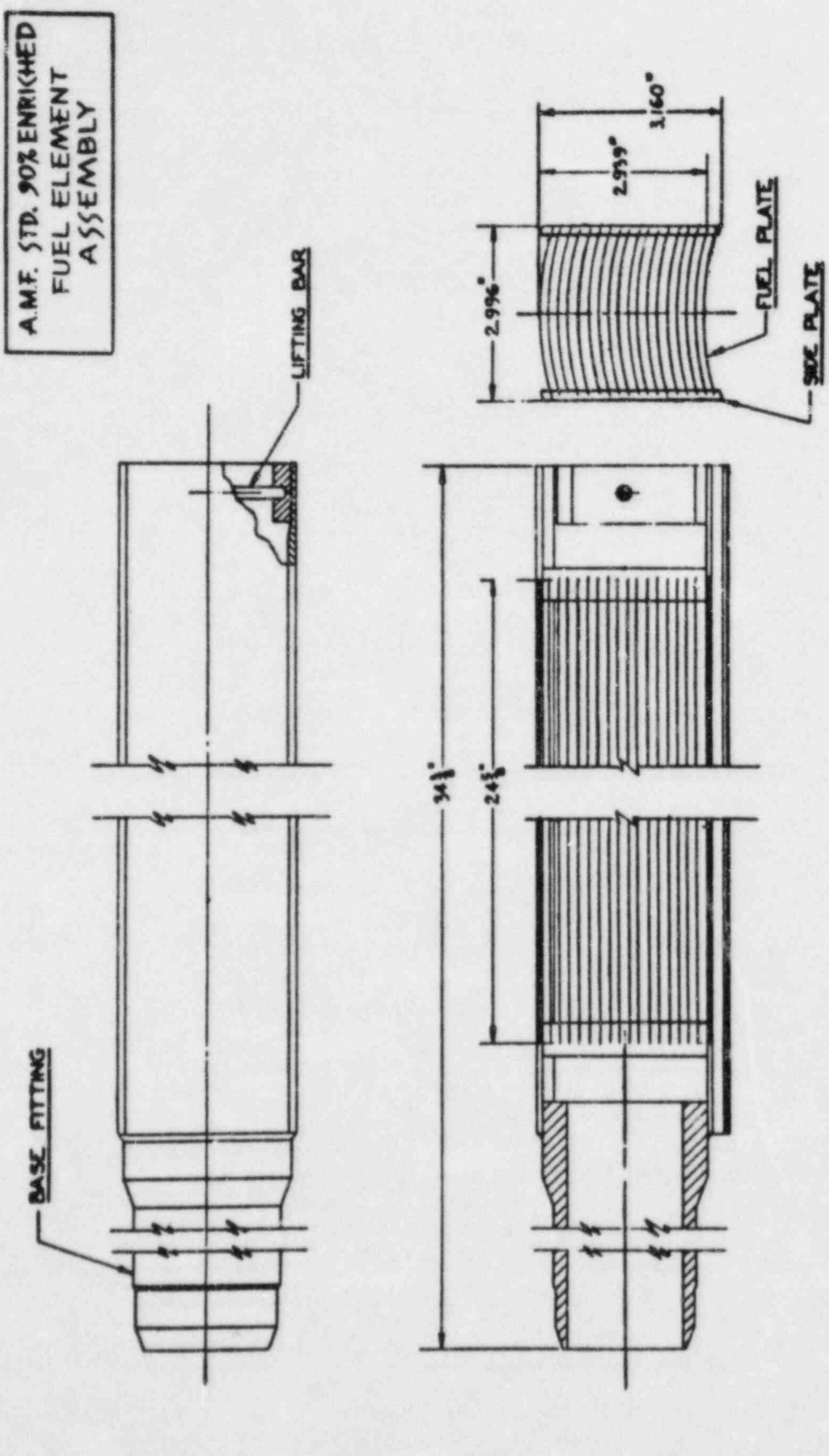

Figure 4.6 Standard fuel assembly

assemblies are similar to the partial assemblies; however, they are longer and contain a centrally located slot into which the reactor control rods are inserted. Assembled at the top of each of these elements is a shock absorber that cushions the fall of the control rods when they are dropped.

The type of fuel element that has seen most service in the UCS reactor is an aluminum clad, uranium-aluminum alloy fuel MTR-type element. This type of fuel element has a long history of satisfactory operation in many research and test reactors. The Technical Specifications also permit the UCS reactor to use two other types of fuel elements that have been demonstrated to give equally satisfactory performance. These are uranium oxide-aluminum powder compacts (or cermets) and uranium aluminide (UAl_x)-aluminum powder compacts.

4.3.3 Partial Fuel Element Assembly

A partial fuel element assembly, used for fast-neutron sample irradiations, is of the same construction as the control rod fuel element assembly and contains nine fueled plates. The nominal fuel content is 110 g ^{235}U.

4.4 Control and Regulating Rods

The reactor control system is typical of those used for pool-type research reactors. The reactor is controlled by means of five thermal neutron-absorbing silver/indium/cadmium control rods and one stainless-steel regulating rod. The control rods provide coarse adjustment of the neutron flux level, and the regulating rod provides fine adjustment. The five control rods can be operated manually and can also be scrammed automatically. The regulating rod may be operated manually or automatically in response to power level demand settings. Console instrumentation provides the operator with the necessary information for proper manipulation of the controls. The following instrument channels are provided and are discussed in detail in Section 7.

(1) counting rate or startup channel
(2) log-N and period channel
(3) linear power level and automatic control channel
(4) three safety channels

USC reactor core has stainless steel control rods. Rod worths vary somewhat, depending on core configuration and fuel distribution. The range for the UCS core is 9-12% Δk/k for the total worth. The effectiveness of these rods, determined for a typical core and rod arrangement, is given below for reactor operation and in the stall position.

Rod	Position	Worth % Δk/k
Control #1	B8	1.3
" #2	B6	2.6
" #3	B4	2.2
" #4	E8	0.8
" #5	E6	2.4
Regulating	E4	0.5
	Total:	9.8

Drive mechanisms are used for remote positioning of control and regulating rods and the fission chamber from the control console. The control-rod drives are mounted near the center of the core support bridge on two aluminum plates. The plates are drilled with a number of mounting holes to permit the drives to be located in a number of positions. The drive mechanism consists of a low-inertia, two-phase motor driving a rack-and-pinion system through a worm and two-spur gear reduction. The mechanism is provided with a long drive shaft and arm, which allows the drive to be mounted above the reactor core with the rack extending over the appropriate rods. The speed of rod withdrawal is limited to ensure a safe rate of reactivity insertion. Means for automatic and manual scrams, rod reversal, and rod inhibits are provided to maintain safe reactor operation.

Each control rod is coupled to the drive mechanism shaft by electromagnets. Scramming or quick insertion is accomplished by de-energizing the electromagnet. The force of gravity separates the control rod from the magnet, and the rod falls into the core. Upon separation, the drive automatically insert until it again touches the control rod. This feature may be bypassed if needed. The time from scram initiation to full insertion is equal to or less than 800 msec.

The regulating rod assembly consists of a stainless-steel rod fastened to a long extension attached to the drive mechanism. Fine and coarse position indicators indicate the regulating rod position.

The regulating rod drive mechanism provides fine control of the reactor. The position of the regulating rod is servocontrolled to maintain constant reactor power. The rod also can be inserted or withdrawn manually.

The regulating rod is designed to have a total worth of 0.3 to 0.6% $\Delta k/k$, depending on the location in the core. This is adequate for the regulating function and has the important advantage that malfunction of the regulating rod or operator error in rod manipulation could not result in prompt criticality.

When the reactor period is $\leq$10 sec (log-N period) the control rods also drive down when the preset power level limits are reached. The temperature settings for the various reverse rod drive indications on the log-N period meter are

Log-N reverse set point	Pool water temperature, Tp
125%	$T_p \leq 15°F$
120%	$115°F < T_p \leq 120°F$
115%	$120°F < T_p \leq 125°F$
110%	$125°F < T_p \leq 130°F$

The staff has reviewed the information pertaining to the design and operation of the UCS reactor fuel, control rods, and regulating devices and concludes that they are adequate to ensure safe operation of the reactor.

4.5 Dynamic Design Evaluation

In addition to the reactor control rods and nuclear instrumentation and controls (Section 7), the UCS reactor has inherent features that ensure its safe operation.

The reactor temperature coefficient is -4.4 x 10-5 Δk/k °F (Burn and Krapp, 1980). If, for any reason, the k_{eff} of the reactor undergoes an increase, there will be an increase in the neutron density and, hence, the fission rate. Because of the increased fission rate, the fuel temperature will rise. The negative temperature coefficient of reactivity will cause k_{eff} to decrease, thereby offsetting the original increase. Thus, the reactor's inherent, negative temperature coefficient provides backup reactivity control and serves to self-limit any potential reactivity excursion. Although small in comparison, the reactor's negative power coefficient, which combines moderator temperature and void coefficients, acts as an additional shutdown method in the event of loss of coolant.

4.5.1 Experiments

The UCS Technical Specifications limit the combined worth of all experiments to 2% Δk/k, with a further limitation on the combined worth of all movable experiments of 1.7% Δk/k. Single secured experiments are limited to a worth of 0.5% Δk/k. Movable single experiments that can be moved when the reactor is critical are limited to 0.25% Δk/k. Single experiments that can be moved while the reactor is subcritical by at least 0.75% Δk/k are limited to worths between 0.25 and 0.5% Δk/k.

The limitation on the maximum combined worth of movable unsecured experiments is based on SPERT reactor power excursion tests, which have shown that the reactor can safely self-limit a step reactivity insertion of $2.14 (USAEC, IDO-17000, 1964). This corresponds to an insertion of 1.73% Δk/k based on a b (eff) of 0.0081.

The licensee has analyzed two step-reactivity insertions. Each analysis assumes that the reactor is operating at 5 MW with forced convection (0.25 MW for natural convection) and that the rods are fully withdrawn. The latter assumption is conservative because it maximizes the rod travel required to offset the step-reactivity insertion. The maximum power achieved is calculated conservatively because it ignores the mitigating effects of rod insertion. The control rod magnet release time and the scram time, as stipulated in the Technical Specifications, have maximum allowable values of 0.05 and 0.80 sec, respectively.

The analyses considered step-reactivity insertions of 0.25 and 0.5% Δk/k, which correspond to the Technical Specifications maximum worth limitations for a movable experiment that can be removed with the reactor critical and for a single secured experiment, respectively.

The maximum power levels resulting from these step insertions at the end of the scram time (0.80 sec are shown in Table 4.1. The power levels shown in Table 4.1 do not exceed the safety limits established in the Technical Specifications for UCS. For forced cooling, the safety limit (1) is dependent on flow rate, (2) requires a pool water level above the core of ≥20 ft, and (3) assumes a pool water temperature of 120°F. For the design flow rate of 2,200 gal/min, the safety limit is ~16 MW. For the natural convection case, the safety limit for fuel cladding temperature is 6.7 MW with the water level ≥20 ft above the core.

Table 4.1 Maximum power resulting from step-reactivity insertions

Reactivity insertion (% Δ/k)	Resulting power level (MW)	
	Forced cooling*	Natural convection**
0.25	7.3	0.36
0.5	14	0.71

*Initial power level = 5 MW
**Initial power level = 0.25 MW

4.5.2 Shutdown Margin

The Technical Specifications limit the total excess reactivity in the core at any time to 10.0% Δk/k, including experiments when the core is in the stall (forced cooling) position, and 8.2% Δk/k when in the open pool position. Included in this amount of excess reactivity is approximately 3.5% Δk/k necessary to overcome the xenon poison of and the negative power coefficent of 0.07 Δk/k per MW, or 0.35% Δk/k. With the control rods out minus [3.5 + 0.35] Δk/k, the net excess reactivity is 10.0% Δk/k minus 3.85% Δk/k or 6.15% Δk/k. Following a scram and with the highest worth rod stuck out (see Section 4.4), the shutdown margin is 6.15% Δk/k minus 6.7% Δk/k = -0.55% Δk/k. The shutdown margin for the core in the bulk pool position would be similarly in excess of 0.5% Δk/k.

4.5.3 Conclusion

On the basis of the information presented above, the staff concludes that (1) the reactivity worth of the control rods; (2) the maximum time for the full insertion of the rods; (3) the limitation on the total experiment reactivity worth of 2% Δk/k*, which includes the limitation of 1.7% Δk/k total reactivity worth for unsecured experiments; (4) a limitation of 0.25% Δk/k per experiment that may be moved when the reactor is critical, and (5) a limitation of 0.5% Δk/k per experiment that may be moved when the reactor is subcritical by at least 0.75% Δk/k provide assurance that these experiments will not lead to a reactivity insertion that will pose a threat to the health and safety of the public. In addition, the staff concludes that the 0.5% Δk/k shutdown margin is sufficient to ensure that the reactor can be shut down under all anticipated conditions.

4.6 Core Thermal and Hydraulic Characteristics

The licensee has performed analyses to determine the limiting safety system settings for the reactor operation for both forced and natural convection

*The combined limits pertain to experiments that can add positive reactivity to the core resulting from a common-mode failure.

cooling. The thermal hydraulic safety analysis considered the burnout heat flux and incipient boiling for both cooling modes. Hot channel factors, flow rates (forced cooling modes), pool temperature effects, bulk coolant temperatures, methodology uncertainties, and available SPERT IV and Oak Ridge reactor (ORR) data have been used in the analyses. A conservative accident analysis assumes a 30-element core (counting control rod fuel elements as a 1/2 element), experiments equivalent to flow in 6 fuel elements, and the core at a 20-ft depth below water level. The staff has reviewed the licensee's analyses (Oak Ridge National Laboratory (ORNL) TM-2421, ORNL-3026, ORNL-3079, ORNL-59-8-39) and finds that the methods used are appropriate and conservative for application to the UCS reactor. The staff further concludes that reactor operation, within the conditions established by the licensee's safety limits and limiting safety system settings as given in Table 4.2 and the Technical Specifications, and which were developed from the thermal-hydraulic safety analyses, is safe and poses no health or safety hazard for the licensee's personnel or the general public.

Table 4.2 Limiting safety system settings

Parameters	Values
Forced Cooling	
Power level for any flow	7.50 MW
Coolant flow for power levels >250 kW	≥1,800 gal/min
Pool water level above core	≥20 ft
Natural Convection	
Power level	≤250 kW
Pool water level above core	≥20 ft

4.7 Operational Practices

UCS operations include a thorough preventive maintenance program that is supplemented by a detailed preoperational checklist to ensure that the reactor is operated at power with the appropriate safety-related components being operable as indicated in the Technical Specifications. The reactor is operated by NRC-qualified personnel in accordance with explicit operating procedures, which include specified responses to any reactor control signal. All proposed experiments involving the use of this reactor are reviewed by the UCS Nuclear Safeguards Committee for potential effects on the reactivity of the core or damage to it, as well as for possible effects on the health and safety of employees and the general public.

4.8 Conclusion

The staff concludes that the UCS reactor is designed and built according to approved industrial practices. It consists of standardized components representing many reactor-years of operation and includes redundant safety-related systems.

The staff review of the reactor facility has included studying its specific design and installation and operational limitations as identified in the Technical Specifications, revisions, and all other pertinent documents associated with the license renewal. Therefore, based upon the UCS reactor design and its operating experience since 1962, plus experience with other similar reactor facilities, the staff concludes that there is reasonable assurance that the UCS reactor is capable of continued safe operation.

5 REACTOR COOLING SYSTEM

The cooling system for the UCS reactor consists of three subsystems--the primary, secondary, and purification subsystems. These are shown in Figure 5.1. These subsystems are interconnected with the instrumentation and control systems discussed in Section 7.

5.1 Primary Cooling System

The primary cooling system consists of demineralized water plus heat exchangers and pumps (see Figure 5.1). Heat generated in the pool water by the reactor is transferred to the heat exchangers where it is subsequently removed by the secondary cooling system. The secondary cooling system, in turn, transfers its heat to the atmosphere by the use of a cooling tower. During forced cooling, pool water flows downward in the pool or stall (depending on the core position used) through the reactor core grid plate and plenum at a flow rate of ~2,200 gal/min and then to a holdup tank. Subsequently, the water is drawn from the holdup tank by a 2,200-gal/min main circulating pump, and is pumped through the shell sides of the two stainless-steel heat exchangers in series and back into the pool.

Butterfly valves are provided in each of the supply and drain lines from the reactor to the holdup tank to adjust the flow rate through the core or to close off the coolant flow from the section of the pool not in use. The valve in the exit from the stall (the usual reactor operating position) is motorized and operated from the control room.

Makeup water is supplied from the research center filtering and demineralizing plant (see Section 5.3 for details).

Primary coolant piping is aluminum; however, the portion of the reactor piping embedded in the concrete is constructed of stainless steel to eliminate the possibility of corrosive attack on the inaccessible piping.

5.1.1 Plenum

The plenum provides an enclosed passage for the cooling water flowing down through the fuel elements and grid plate to the core outlet pipe at either the pool or stall operating position (see Figure 5.2). It is bolted to the lower mounting flange of the grid plate. A short guide tube containing a metal bellows is attached to and extends from the bottom of the plenum. The bellows, which incorporates a special flange and sealing ring insert, may be vertically displaced a small distance. This displacement permits disengagement of the core support structure from the reactor core outlet assemblies at either operating position, before moving the reactor.

A hinged, counterbalanced safety flapper is attached to the side of the plenum. When properly counter-balanced (with flapper ballast weights and adjusting weights) against the suction in the plenum, the safety flapper will immediately

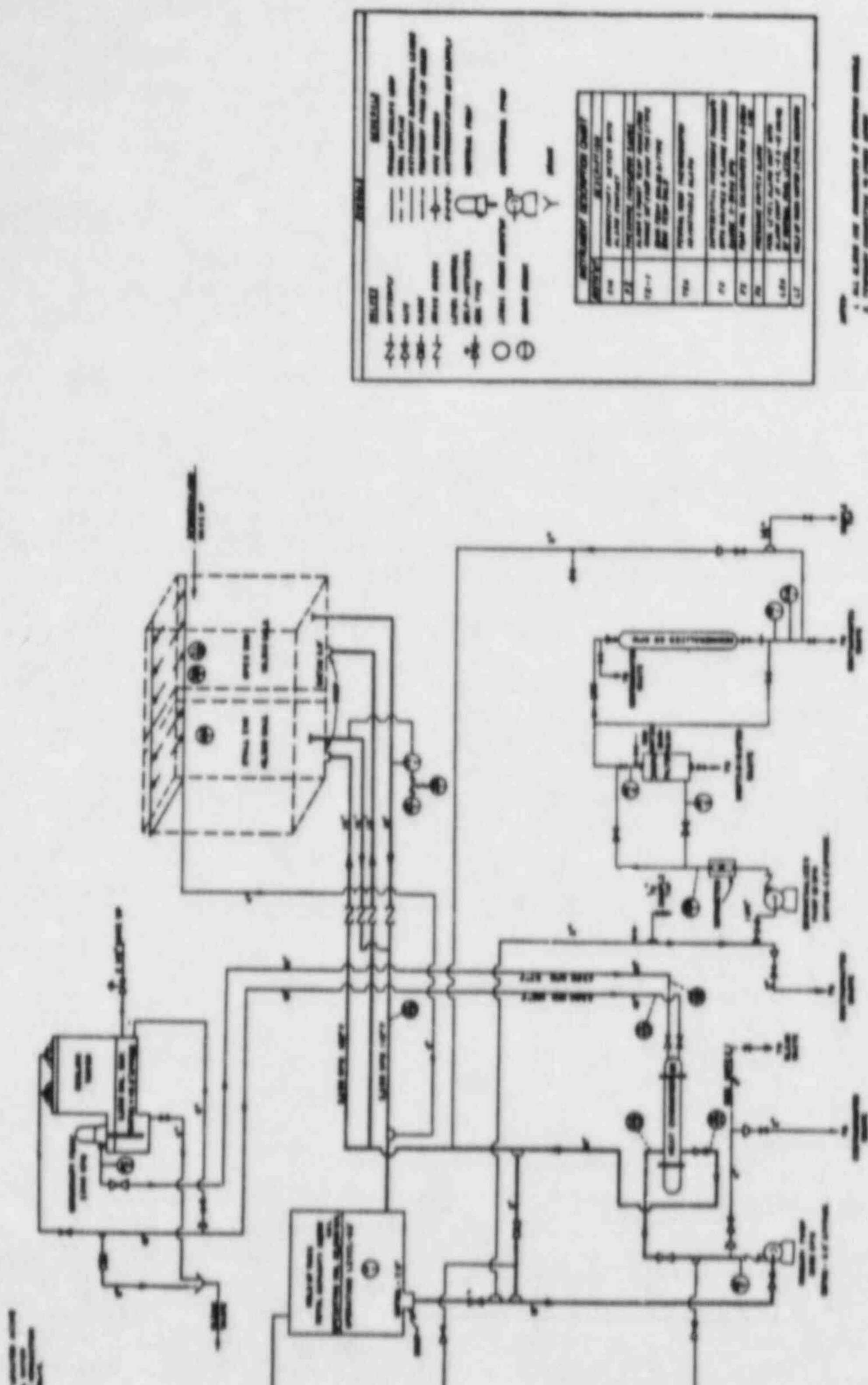

Figure 5.1 Cooling and purification system flow diagram

drop open to provide a path for natural convection cooling of the core, should the cooling water flow fall below 700 gal/min. The safety flapper is closed initially at startup with an actuating rod manipulated from the bridge (see Figure 5.2).

5.1.2 Core Outlet Assembly

A core outlet structure is mounted on the floor of the pool by anchor bolts at both core and stall operating positions. The assembly consists of a vertical length of aluminum pipe and a mating flange. The pipe serves as an extension between the floor drain pipe and the plenum bellows seal.

5.1.3 Holdup Tank

A holdup tank is used to provide a 10-min delay of the pool water in the primary system during normal operation to allow time for decay of ^{16}N and other short-lived isotopes in the coolant before the water enters the pump room. It is an underground concrete enclosure adjacent to the pump room. Although the total capacity is 33,000 gal, the operating volume is normally 11,000 gal.

The holdup tank is located at an elevation so that if the primary pump is shut off, the water levels in the pool and holdup tank equalize at a pool elevation approximately 21 ft above the reactor core. The holdup tank is vented to the main exhaust stack through a solenoid-operated isolation valve and is air-purged to remove radiolytic gases.

5.1.4 Primary Circulating Pump

One primary, 2,200 gpm, electrical centrifugal circulating pump is provided. The pump takes suction from the holdup tank and pumps the cooling water through the shell side of the heat exchangers to the pool. The heated pool water flows downward back to the holdup tank.

5.1.5 Heat Exchangers

Two stainless-steel, shell-and-tube heat exchangers, located in the pump room, are of the fixed tube sheet, two-pass type, and are in series. The pool water is circulated through the shell sides of the units.

5.1.6 Storage Tank

A 100,000-gal aluminum storage tank is provided near the reactor building. This allows for drainage and subsequent storage of pool water while work is being done inside the pool. The bottom of the storage tank is about 8 ft above the pool surface so that the pool cannot be inadvertently drained by gravity. Pool water must be transferred to the storage tank by means of the primary circulating pump.

5.2 Secondary Cooling System

The secondary cooling system transfers the heat from the heat exchanger to the cooling tower where the heat is dissipated to the atmosphere. The secondary system is at a higher pressure than the primary system, thus preventing leakage

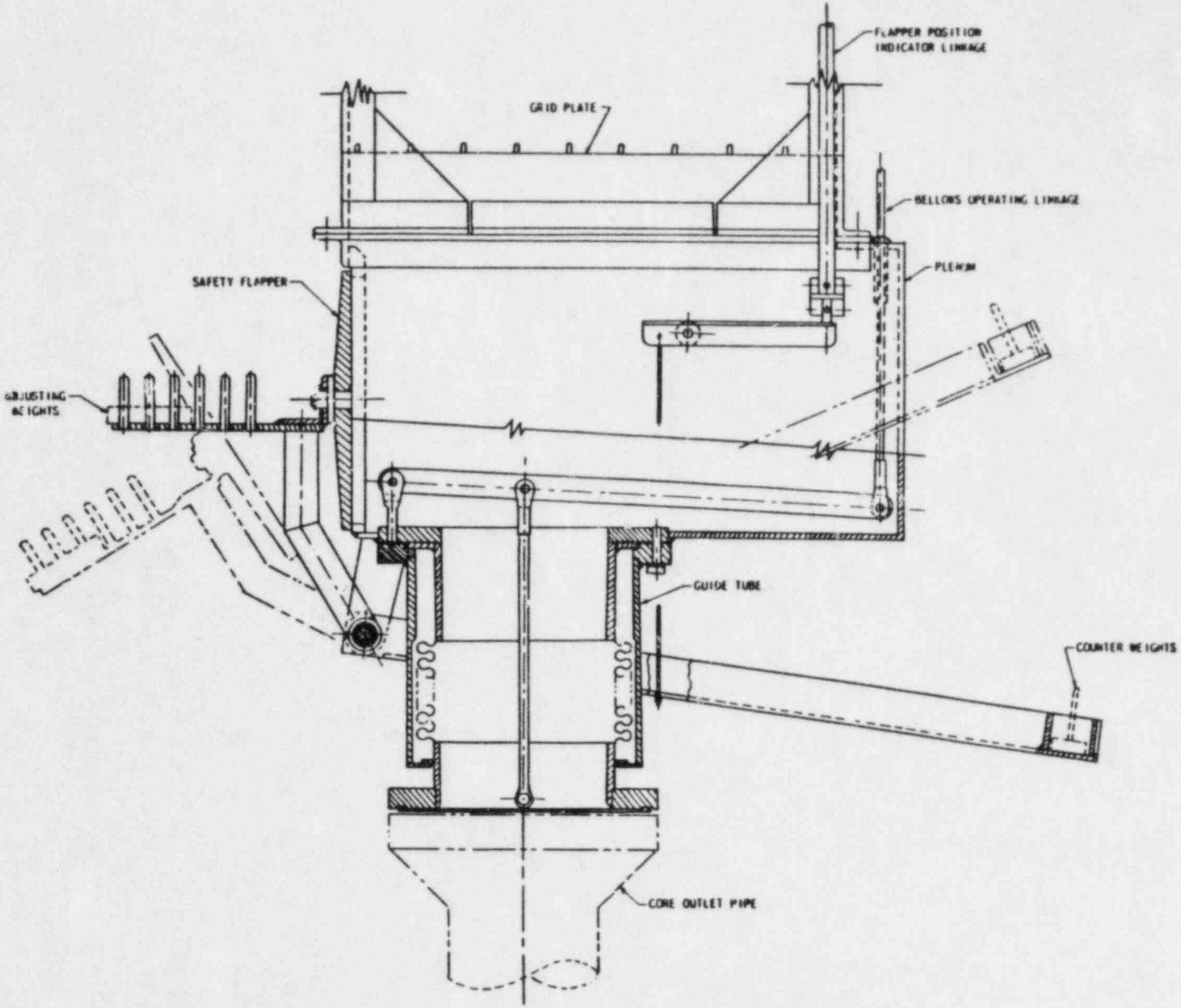

Figure 5.2 Plenum, safety flapper, and grid plate assembly

of primary coolant to the secondary system. The secondary system pump delivers 2,300 gal/min.

5.2.1 Cooling Tower

A two-bay, forced-draft cooling tower, located near the reactor building, dissipates the heat in the secondary water transferred from the heat exchanger. Each of the two fans has three speeds--off, low, and high. Normal operation of the cooling tower is based on an atmospheric wet bulb temperature of 75°F. There is provision for fan reversal to remove ice in winter. Makeup water is supplied from the regular municipal water supply.

5.3 Water Purification System

The purification system maintains the purity of the primary coolant system at a specific resistance of 200,000 ohm-cm and a pH of between 5.0 and 7.5.

A small stream of primary coolant is removed from the holdup tank discharge line by means of a 200-gal/min-capacity pump. This stream is passed through anion and cation exchange columns and a filter. Each column and filter is designed for 100 gal/min flow. The system is designed such that these columns and the filter can be arranged in series, parallel, series-parallel, or individual flow sequences. The system also is designed to accommodate an additional mixed-bed resin ion exchange column. Each bed is normally regenerated once and then disposed of as radioactive waste (see Section 11). The primary coolant makeup water is treated by a separate demineralizer and filter system.

5.4 Conclusions

The cooling and coolant purification systems have been in operation for more than 20 years. They are well instrumented (see Section 7) and well maintained. The low bulk temperature of the coolant provided by the primary and secondary systems reflects the adequacy for cooling of the reactor at 5-MW full power operation. The purity of the coolant is adequate to minimize corrosion of the components in the system as indicated by the virtual corrosion free operation.

The staff, therefore, concludes that the reactor cooling system is adequate to ensure continued safe reactor operation.

6 ENGINEERED SAFETY SYSTEMS

The engineered safety systems are those designed to mitigate the consequences of accidents and include the ventilation system, the core spray system, and the emergency electrical power system. The emergency electrical power system is described in detail in Section 8.

6.1 Ventilation System

The reactor building ventilation system (Figure 6.1) is designed to supply comfort air conditioning, reactor pool sweep air, and to exhaust low-level radioactive emissions from several regions and special equipment during normal operations. It also is designed to minimize releases of airborne radioactivity during emergency conditions.

6.1.1 Normal Ventilation Conditions

Approximately 19,000 ft^3/min of outside air is drawn into the facility fan room, filtered and cooled or heated as necessary and distributed throughout the reactor building by a supply fan. The facility is maintained at 75°F in winter and 80°F in summer. There are about 4 air changes per hour and no air is recirculated. Although designed for a negative pressure of 3/4 in. water, the reactor building is normally maintained at a negative pressure of 1/4 in. water.

During normal operations, 5,000 ft^3/min of air is directed across the reactor pool to detect any radiolytic or fission product gases. This recombines with the other ventilation streams in the reactor building; the air is exhausted by a 20,000 ft^3/min exhaust blower through a 4-ft duct into the exhaust stack that is located high on a ridge overlooking the complex. The exhaust duct is continuously monitored for indications of abnormal radioactivity. Other streams that may be potentially radioactive, and which also are fed into the main exhaust system, include purge air flow from the holdup tank, filtered exhaust from the beam tubes and thermal column, filtered discharge from the pneumatic rabbit tubes, air flow from the pool sweep, and discharge from the xenon irradiation vent system.

6.1.2 Emergency Ventilation Conditions

Under "emergency" conditions (as when there is a release of fission products into the confinement building) the reactor building would be isolated and personnel would be evacuated. This procedure is initiated by one of the following:

(1) manually operating either of two emergency alarm switches in the reactor control room

(2) manually operating the emergency alarm switch in Building 2 (hot laboratory)

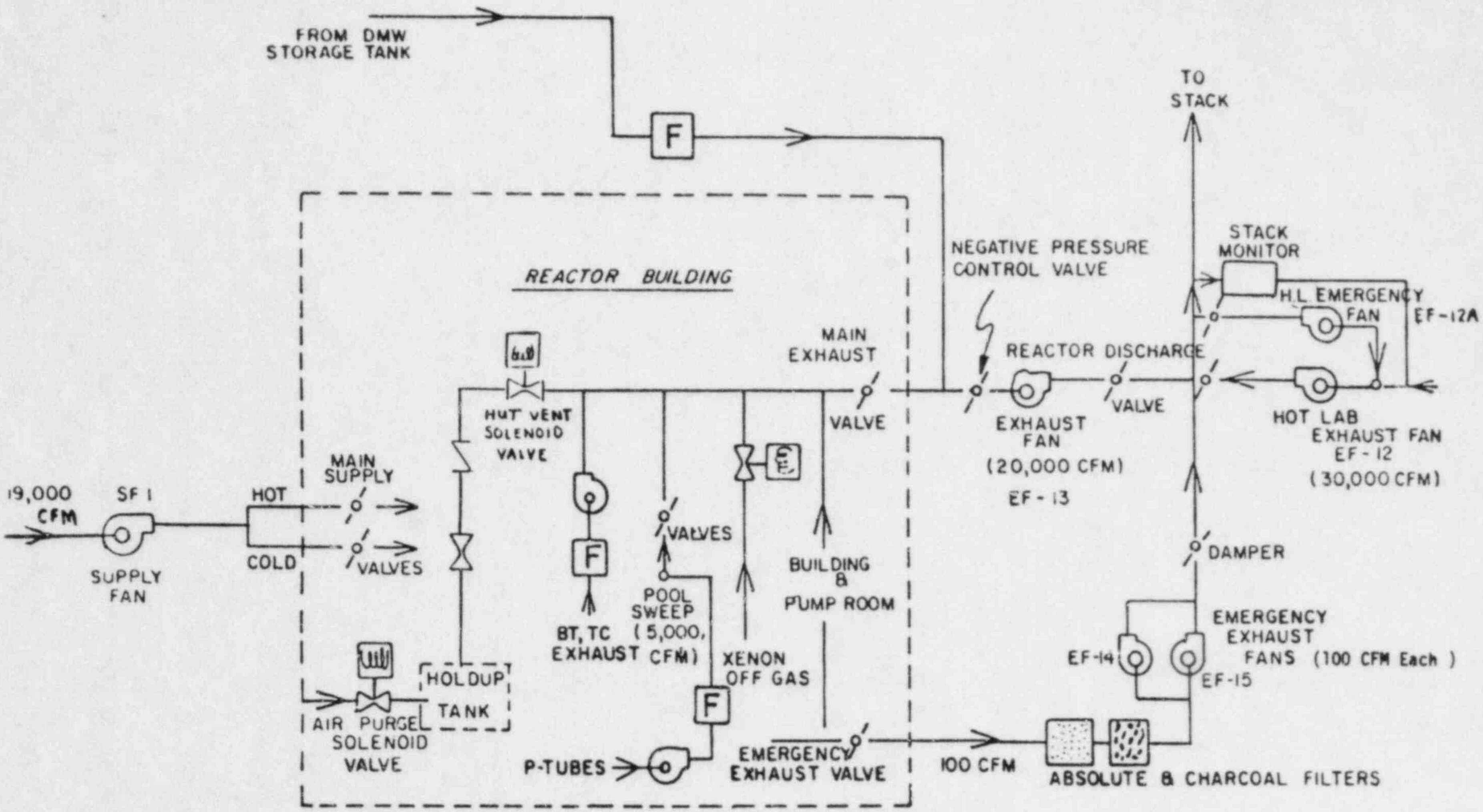

Figure 6.1 Ventilation system

(3) automatically when the radiation level at either of the excursion alarms mounted under the reactor bridge exceeds 5 R/hr

In an emergency condition as defined above, the following events occur immediately and automatically (see Figure 6.1):

(1) The reactor building supply fan shuts down and the air supply dampers close.

(2) The holdup tank air purge supply and exhaust valves close.

(3) The beam tube and thermal column fans shut down.

(4) The pool sweep isolation damper closes.

(5) The xenon offgas valve closes.

As the supply fans are stopped and the exhaust fans continue to operate, the negative pressure in the building will increase. After 7 sec, or when the reactor building reaches a negative pressure of 1 in. H_2O:

(1) The main exhaust fan shuts down.

(2) The exhaust duct damper closes.

(3) One of the two 1 1/2 horsepower emergency ventilation fans starts. The flow is adjusted by the emergency exhaust damper.

(4) The emergency exhaust damper automatically opens fully to provide flow of about 200 cfm through the emergency exhaust system and the emergency charcoal and particulate filters.

The emergency exhaust duct is an auxiliary duct connecting the reactor building through charcoal, roughing, and absolute filters in the fan room to the exhaust stack (Figure 6.1). The building and emergency ventilation system is so designed that, under emergency conditions, a slight negative pressure of at least 0.01 in. H_2O is maintained by the emergency exhaust fan(s) to ensure that building air leakage will be inward during the emergency.

6.2 Core Spray System

The core spray system is planned to function on an emergency. It is operated manually, following indication of a low-water-level signal. It consists of piping and spray nozzles located above the core. In the event that primary coolant is lost and the core is uncovered, the reactor core may be cooled by one of two spray nozzles--one discharging into the pool, the other into the stall. Water is supplied to these nozzles from the municipal water system through manually operated valves. Should the municipal water system fail, a 100,000-gal reservoir, located on a hill above the facility, is available to supply emergency reactor coolant. A valve that isolates this system must be manually operated to activate this water source.

6.3 Emergency Electrical System

In the event of a utility electrical power failure, a 50-kW gasoline-powered electrical generator is available to supply emergency power for minimal operations of critical systems in the reactor building. The system is described in detail in Section 8. The staff has determined that the system design and equipment and testing program are adequate for the emergency electrical system to operate when required and to supply the necessary electrical power to the critical units to ensure safe shutdown of the reactor (see Section 8).

A natural gas-driven 45-kW generator is also available, but is principally assigned to needs in the hot laboratory.

6.4 Conclusions

On the basis of the above review of the engineered safety systems in the UCS reactor complex, the staff concludes that these systems are adequate to ensure safe shutdown and maintenance of the reactor and to prevent the release of radioactivity in excess of the limits imposed by 10 CFR 20, Appendix B, Table II (see Section 14).

7 CONTROL AND INSTRUMENTATION

7.1 Summary

The control and instrumentation systems at the UCS reactor facility are typical of those in wide use for pool-type research reactors in the United States. Control of the nuclear processes is achieved by using five neutron-absorbing control rods and one regulating rod for fine adjustments. The instrumentation system, which is interlocked with the control system, is composed of both nuclear and process instrumentation and is characterized by high-quality components.

The functions and requirements of the control and instrumentation systems are described in detail in the Technical Specifications.

7.2 Control System

The control system is composed of both nuclear and process control circuits and is designed for redundant operation in case of component failure or malfunction.

7.2.1 Nuclear Control System

The nuclear control system consists of the regulating rod drive, the control rods drives, and the fission chamber drive.

The regulating rod drive consists of a low-inertia, two-phase motor with rack-and-pinion gearing that is servocontrolled to maintain constant reactor power.

The control rod drives are low-inertia, two-phase motors with rack-and-pinion and spur gearing connected to drive shafts magnetically attached to the control rods. Controls are provided for operating up to two rods simultaneously. The speed of rod withdrawal is limited to ensure a safe rate of reactivity insertion. In addition to a scram system, the safety instrumentation includes a rod reverse and a rod inhibitor system to maintain the reactor in a safe operating range.

The fission chamber drive is used to remove the fission chamber from the region of high flux.

7.2.2 Process Control System

The process control system consists of the circuitry required to energize and de-energize the cooling pumps, safety flapper, coolant transfer pumps, and the core-support-bridge lock. This system is interlocked with the motor control center (described in Section 8.3).

7.3 Instrumentation System

The instrumentation system is composed of both nuclear and process instrumentation circuits. The electronics system contains both solid-state and tube-type components and provides annunciation and/or indication in the control room in accordance with existing guidelines.

7.3.1 Nuclear Instrumentation

This instrumentation provides the operator with the necessary information for proper manipulation of the nuclear controls.

(1) Log Count Rate Channel

This channel receives data from an in-core movable fission chamber. The primary purpose of the channel is to monitor the reactor while it is in the power range from 1 mW to 10 W.

(2) Log-N Channel

This channel receives data from a gamma-compensated, boron-coated ion chamber. The purpose of this channel is to provide reliable monitoring of the reactor when it is operating between 5 W and 5 MW.

(3) Linear Power Channel

This channel also uses a gamma-compensated ion chamber as the detector. The main purpose of this channel is to provide automatic power control in the power range through interlock with the servocontroller on the regulating rod drive mechanism.

(4) Safety Channels

Two separate and independent channels are provided, that give the desired redundancy required for the main purpose of the safety system, namely to scram the reactor at excessive power. Any one safety channel will scram all control rods. As these channels are used only in the power range, their detectors are uncompensated ion chambers.

All the neutron-sensing ion chambers are located outside of the core and are adjustable over a limited distance to allow their respective channels (log-N, linear-N, safeties) to be standarized to the reactor thermal power derived from primary flow-rate and core ΔT measurements.

Chart recorders are provided for the log count rate, log-N, and linear-N, and the safety channels.

7.3.2 Process Instrumentation

The bulk of this instrumentation is involved with sensing and monitoring parameters associated with the primary and secondary cooling systems. Specific instruments are dedicated to measure or indicate the following parameters:

(1) coolant pool level
(2) coolant pool temperature
(3) primary coolant system flow rate
(4) reactor core differential temperature
(5) primary coolant resistivity
(6) heat exchanger primary differential resistivity
(7) plenum leakage
(8) holdup tank level
(9) reactor core exit temperature
(10) heat exchanger inlet temperature
(11) heat exchanger outlet temperature
(12) cooling tower basin water level
(13) storage tank high and low levels
(14) primary inlet temperature
(15) primary outlet temperature

All readouts, except for resistivity, are in the control room.

Primary flow rate is determined from the differential pressure measured across the orifice plate. The core ΔT is derived from resistanc thermometers above the core and in the core exit line. Pool water level is monitored with three float switches: one set at 6 in. above, one set at 6 in. below, and one set 12 in. below the overflow gutter level. Pool water temperature is measured with a resistance thermometer located above the core. The temperature sensors at the heat exchanger inlets and outlets also are resistance thermometers. The purpose of the differential resistivity measurements across the primary circuit of the heat exchanger is to detect leakage from secondary to primary.

Secondary flow rate is modulated by an automatic control valve controlled by the temperature of the primary water leaving the heat exchanger. This control assists the operator in reducing variation in the temperature of primary water returned to the pool.

The occurrence of a leak in the plenum below the core would bypass some of the core flow into the plenum. This occurrence would be detected by a plenum leak detector. This detector is in reality a differential pressure switch that senses a change in pressure in the core exit line upstream of the low rate orifice plate. This differential pressure switch is preset. An increase in this upstream pressure, which reflects a leak into the plenum, will increase the pressure differential and will activate an alarm in the control room.

An annuciator panel with lamps and alarms in the control room indicates conditions in the various portions of the cooling system, such as cooling-tower fan speeds, primary pump, holdup tank level, storage tank level, secondary pump, tower basin level, and demineralizer pump. The system includes temperature sensing elements that alarm in case of high water temperature. A digital switching meter in the control room shows water temperature in the pool, at the reactor outlet, and at the heat exchanger inlet and outlet for both primary and secondary flow.

Facility radiation monitoring instruments also are included in the process instrumentation. Within the reactor building at various points, monitors are provided to detect local radiation levels and to provide alarms when preset levels are exceeded. The alarms and the levels are indicated in the control room. Duplicate monitors are located at the bridge and serve to initiate the "evacuation sequence" for personnel in the building. In addition, two continuous air monitors (CAMs) continuously sample the building air for radioactive particulates and print the results on a chart recorder. Each CAM gives an alarm at a preset level.

The effluent in the 4-ft stack duct is continuously sampled to provide indication of abnormal levels of airborne radioactive material. This is accomplished by withdrawing a side stream of air from the duct, passing this through particulate, iodine, and gaseous radioactivity monitors, and returning it to the inlet side of the hot laboratory fans. The outputs of these detectors are indicated on local chart recorders equipped with alarm set points and on chart recorders in the reactor control room.

7.4 Safety System

The control and instrumentation systems are interlocked through the safety system at the UCS reactor. This system is described below.

7.4.1 Rod Movement System

Actual pickup or release of the rods (by energization or de-energization of the supporting magnets) is the function of a group of instruments and associated circuitry composing the safety system. The system is designed to shut down the reactor by immediate dropping (scram) of the control rods, driving control rods in, and inhibiting control rod movement if any of the respective conditions shown in Table 7.1 occur.

(1) high power
(2) short period
(3) low flow
(4) unlocking of the core support bridge
(5) low pool water level
(6) lifted guide tube
(7) primary pump failure
(8) flapper open

In addition to the above, the safety system allows the reactor to be shut down quickly by the operator or other personnel by use of the manual scram stations.

7.4.2 Safety Amplifier

The safety amplifier provides the amplification, control, monitoring, and chamber power supply functions for the safety system. Four independent safety amplifiers are provided. Each safety amplifier can accept the signal current from one uncompensated ion chamber. The safety amplifiers also control the current for the control-rod electromagnets through electronic switches. Each magnet has its own switch. Each amplifier supplies current to a pair of high-speed relays whose normally open contacts are in series with those from all the other amplifiers. Opening any one of these contacts causes all the switches to open and cut off current to all the rod electromagnets, thus releasing all the control rods. This action is termed a "fast scram." This

Table 7.1 Scrams, Reverse, and Inhibits

1. Scrams:

   (a) Fast scram at 150% power (linear safety (2))

   (b) Fast scram at 3-sec period (log-N period)

   (c) Slow scram from manual pushbuttons (9 locations)

   (d) Slow scram at water level 20 ft above core

   (e) Slow scram at flow rate of $\leq$1800 gpm

   (f) Slow scram when flapper opens (flow less than 700 gpm)

   (g) Slow scram when primary pump motor loses electrical power

   Bypassed below $2N_L$ (.001)

   (h) Slow scram when any guide tube lifts

   (i) Slow scram when the bridge handle is removed from its locking spline

   (j) Slow scram from console keyswitch

2. Reverses:

   (a) Reactor period less than or equal to 10 sec (log-N period)

3. Inhibits:

   (a) Fission chamber in motion

   (b) Log count rate recorder off, less than 2 cps, more than 9800 cps, or in calibrate position at a log-N level of less than 0.001% power

   (c) Count rate period of less than 30 seconds when log-N below 0.001% power

   (d) Log-N period of less than 30 sec

   (e) Any guide tube lifted

scram mode is reserved exclusively for the rapid shutdown of the reactor required by excessively high power or short period. Additional contacts in the high-speed relays are arranged to de-energize another set of relays that interrupt the supply to the magnet current power supply. The magnet current accordingly decays and results in a backup scram, called the "slow scram."

This scram mode is employed for protective actions where very rapid shutdown is not necessary (see Section 7.4.3).

Each safety amplifier chassis has an annunciator panel for indicating faults and scram location, test buttons for checking scram operation, and panel meters for amplifier output.

7.4.3 Magnet Current Interlocks

Operating power for the safety amplifier is received from two sources through two separate power plugs. One source, the 115-V, single-phase, regulated power line, supplies power for all circuits except the magnet current electronic switches. Power for the latter is obtained from an integral supply that is energized by 115-V, single-phase, regulated ac voltage through a separate plug on the amplifier.

The magnet power supply, ac voltage, is obtained through interlocked contacts in series. The contacts represent and are actuated by conditions important to safe reactor operation. If a condition is unsafe, its associated contact will be open and the reactor cannot be started. Conversely, if a contact should open during operation, the reactor will be shut down immediately because of a loss of magnet current. The magnet power supply interlock contacts are opened under the following conditions:

(1) manual scram buttons depressed
(2) main cooling pump off
(3) safety flapper open
(4) bridge clamp unlocked
(5) low pool water level
(6) low flow
(7) guide tubes lifted

Conditions 2, 3, and 6 are bypassed at low reactor power by a backset switch on the log-N recorder.

The magnet power supply interlock circuit also is used to obtain a rapid manual shutdown of the reactor. The contacts of a console-mounted pushbutton are in series with all the other contacts. The circuit also is carried off the panel through several external scram stations before returning to the panel and then to the magnet power plug. Opening the circuit with any one of these switches also removes power from the magnets.

The console-mounted annunciator provides a means of indicating bypass status when manually bypassing any or all of the following magnet power supply interlocks:

(1) main cooling pump off
(2) safety flapper open
(3) bridge clamp unlocked
(4) low pool water level
(5) low flow
(6) guide tubes lifted
(7) manual scrams

These bypass provisions make it possible to check the interlocks during normal startup.

7.4.4 Fast Scrams

Under certain conditions, shutdown action must be initiated a few milliseconds after an unsafe condition occurs. The two conditions requiring this action are (1) a high neutron density in the reactor and (2) an excessive high rate of increase of neutron flux.

The first condition is detected by the safety amplifier, which uses ionization chambers as the sensing elements. When the neutron density exceeds a preset point, sensitive relays are de-energized in the safety amplifier and magnet current is reduced rapidly below the holding point. The contacts of the sensitive relays accomplish this function by opening the magnet current electronic switches, as described in Section 7.4.2.

The second condition is detected by the log-N amplifier. This amplifier differentiates the signal from its ionization chamber. When the differentiation voltage exceeds a preset point, a sensitive relay is de-energized. A normally open contact of this relay is wired into the safety amplifier in such a way that magnet current is reduced by the same mechanism as described in the preceding paragraph, namely by opening the electronic switches for all the magnets.

Although the Technical Specifications indicate a breakaway time of 50 msec, the normal total time from initiation of control action to breakaway of the rods is actually between 5 and 20 msec.

7.4.5 Annunciation

To provide the operator with a constant indication of all of the critical variables affecting reactor operation, a console-mounted annunciator is provided. The annunciator is energized continuously through the main disconnect switch. There are two lights for each annuciated condition, one red and one green. All conditions are annunciated by means of relay or switch contacts. When the contact, wired to a given point, is opened, the corresponding red light is turned on. When the contact closes, the red light is turned off and the green light is turned on. These lights are affected only by the actual conditions of the external contacts.

The settings in the annunciator are divided into the following three groups, according to the type of annunciation:

(1) alarm buzzer, light
(2) alarm horn, light
(3) light only

7.4.5.1 Alarm Buzzer, Light

For the contacts associated with conditions causing interruption of magnet current, the scram buzzer sounds. A "silence" button, located on the annunciator, can be pressed momentarily to silence the buzzer. The individual light in question remains red until the condition is corrected. The lights for conditions 1 and 2 are located on the safety amplifier and log-N amplifier, respectively, and remain on until reset. The conditions causing annunciation are

(1) high neutron flux level, safety amplifier
(2) short period, log-N amplifier
(3) manual scram, console or external
(4) main cooling pump off
(5) safety flapper open
(6) core support bridge unlocked
(7) low pool water level
(8) low flow
(9) guide tubes lifted

The alarm contacts for conditions 4, 5, 6, 7, 8, and 9 are duplicated in the magnet current interlock circuit. When bypassing a particular function, both the condition itself and its annunciator are bypassed. The method of bypassing uses a phone plug and jack. Each bypass jack is located under the individual alarm light of its respective point. To bypass, it is necessary to insert a phone plug in the jack. Conditions 1 and 2 and the console manual scram cannot be bypassed.

7.4.5.2 Alarm Horn, Light

The following conditions operate the alarm horn. The individual lamps are actuated as described previously.

(1) period reverse
(2) high flux reverse
(3) shimming required
(4) controller power "off"
(5) pool water level abnormal
(6) ion chamber low voltage
(7) high radiation
(8) high core ΔT

Contacts for condition 5 are high and low water level float switches sensing changes in water level of plus-or-minus 6 in. from gutter lip.

The relays for conditions 1 and 2 provide contacts to prevent rod withdrawal and to insert all shim-safety rods by a "Reverse" when the period is less than 10 sec.

7.4.5.3 Lights Only

The following conditions provide lights only:

(1) high pool temperature
(2) period inhibition
(3) count-rate recorder off-scale inhibition
(4) reverse
(5) plenum leak
(6) off-magnet bypass

7.4.6 Warning and Operating Lights

Under the following conditions, warning or operating lights are activated:

(1) When a rod is not in contact with its magnet, a red lamp lights.

(2) When the rod is fully inserted, a green seat lamp lights.

(3) Until rod drives are withdrawn to shim range point, a shim range light remains on.

(4) All pumps operated from the console have on-off lights.

(5) When a rod drive is at its lower limit, a green lamp lights or for its upper limit, an orange lamp lights.

7.5 Conclusion

The control and instrumentation systems at the UCS reactor are well designed and maintained. Redundancy in the crucial areas of power measurements is ensured by overlapping ranges of the log-N and linear power channels.

The overall system is designed so that manual bypasses of the process interlocks can be used to check the function of the interlocks during normal startup. This helps ensure reliable operations.

The control system is designed to scram the reactor automatically if electrical power is lost. However, emergency power is provided to all crucial reactor monitoring systems and to critical ventilation systems to maintain them operational (see Section 8).

On the basis of the above evaluations, the staff concludes that the control and instrumentations system satisfy all existing regulations and are adequate to ensure continued safe operation of the facility.

8 ELECTRIC POWER

The UCS reactor has two separate sources of power: (1) normal operating power, which is utility-furnished, and (2) emergency power, which is provided for monitoring the reactor and operation of the safety systems necessary to ensure safe shutdown of the reactor and to maintain operation of the critical ventilation systems.

8.1 Normal Operating Power

Normal operating power is provided by commercial utility and is standard three-phase, stepped down to 440 V by a dedicated transformer located near the reactor building.

8.2 Emergency Operating Power

Emergency power for the facility is provided by two motor-driven generators. One of these units is gasoline powered and the other is fueled by natural gas. The gasoline-powered, motor-driven electrical generator is 50 kW capacity and provides overall reactor building emergency power in the event of a utility electrical power failure. The following reactor building equipment receives electrical power from this generator.

(1) portion of the reactor building lighting
(2) reactor control console
(3) reactor building exhaust fan (at half speed)
(4) reactor building supply fan (at half speed)
(5) emergency ventilation system
(6) beam tube ventilation system
(7) beam tube flushing pump
(8) swing-type airlock door controls
(9) gasoline pump from storage tank to emergency generator
(10) hot laboratory fan at half speed or hot laboratory standby fan at half speed
(11) certain electrical receptacles for mobile equipment such as radiation monitors

The purpose of this generator is to supply emergency power for operation of critical systems in the reactor building. The reactor is never operated, even at low power levels, on emergency electrical power. Sufficient gasoline is stored to guarantee a 6-day supply for the emergency generator.

The primary purpose of the natural-gas-fueled generator (45 kW capacity) is to provide emergency power to the hot laboratory facility exhaust and supply fans at half speed and to other areas of the complex apart from the reactor building However, the emergency evacuation system in the reactor building can also be powered by this generator.

8.3 Motor Control Center

The main motor control center, in addition to controlling the numerous functions of equipment under normal operation, controls the operations and sequences necessary for the use of the emergency ventilation system and the transfer of power loads from the normal supply to the emergency generator supply. The functions of the motor control center that are of interest in this section are those required to control the latter two emergency operations. Operation of the emergency ventilation system was previously described in Section 6.

The motor control center operational modes described below are those in which a power failure occurs during the following operating conditions:

(1) simultaneously with the startup of the emergency ventilation system
(2) after the initial timing sequence of the emergency ventilation system has been completed
(3) restoration of normal line power after the emergency ventilation system has started

8.3.1 Condition 1

If a power failure occurs in the same instant that the emergency ventilation system is put into operation, either automatically or manually, the loss of electrical power for the few seconds required to start the generator (approximately 5 sec) will completely close the dampers in the air ducts of the entire ventilation system. These dampers are designed to be fail-safe and air-tight so that on loss of either electrical power or pneumatic power, they will close. The closing time for these dampers is less than 3 sec.

After the emergency electrical generator has come up to speed and the automatic transfer switch in the motor control center has transferred power from the normal bus to the emergency bus, the emergency ventilation system will be energized. The signals that initiate the emergency ventilation system are of a type that will not reset automatically after recovery of electrical power. There is no possibility that a power failure coinciding with a need for the use of the emergency ventilation system could return the ventilation system to normal, and thus exhaust quantities of untreated air into the general atmosphere.

8.3.2 Condition 2

If electrical power from the utility should fail after the initial 7-sec timing sequence for the emergency ventilation system has been completed, it might be possible for the system to reset itself when the emergency electrical generators assume the load, thus repeating the 7-sec cycle. To eliminate this possibility, a manual reset relay is connected to the timer that controls the initial operation of the main exhaust fan for the 7-sec period. In this way, the additional operation of the main exhaust fan is prohibited once the timer has cycled through its planned sequence. Loss of normal power and its subsequent replacement by emergency power would have the following effects:

(1) The entire system, including the dampers, would stay shut down for the period of time necessary for the emergency generator to assume the load.

(2) Once the load is assumed by the emergency generator, the system would continue to function as it had immediately before the power failure; that is, the main dampers stay shut.

8.3.3 Condition 3

The emergency electrical generator and its associated automatic transfer switch are interconnected in such a way that once normal power from the local power company is resumed, the electrical load is switched back automatically to the normal bus. The possibility of the emergency ventilation system being programmed automatically through the initial 7-sec phase (described in Section 6) with the resultant discharge of large quantities of untreated air is prevented by the need to manually reset the emergency ventilation system.

8.4 Conclusion

On the basis of the above analysis the staff concludes that the emergency power provisions at the UCS reactor are well designed. The motor generator appears to be well maintained. The power systems (both normal operating and emergency) are well suited to their roles in operation of the facility, and, because the reactor is never operated on emergency power, the possibility of a loss of power-related nuclear accident is extremely remote.

The staff concludes, therefore, that the normal and emergency electrical power provisions for the reactor facility are adequate for continued safe reactor operation.

9 AUXILIARY SYSTEMS

The auxiliary systems include the fuel-handling and storage system, the compressed air system, and the fire protection system. The portion of the ventilation system not associated with emergency operation is normally considered an auxiliary system. The ventilation system was described in Section 6.

9.1 Fuel Handling and Storage System

New fuel elements are stored in wall-mounted racks in a concrete vault equipped with criticality alarms. The fuel element spacing and rack locations are such that a critical assembly would not be possible with the maximum number of elements of the highest possible enrichment stored, even if the vault were to be completely filled with water.

Irradiated fuel elements in the reactor pool are manipulated with specially designed long-handled tools. Fuel elements to be loaded in the core or spent fuel elements are stored in racks installed on sides of the pool or in movable racks located on the pool floor.

As indicated in Section 1.5, the reactor routinely operates on a 14-day cycle, 7-days a week, 24 hours a day. At the end of the 14-day operating cycle, the reactor is shut down to move fuel within the core, to remove spent fuel, and to install new fuel. The reactor is normally on line about 90% of the time.

At a power level of 5 MW and 90% duty cycle, a total of about 20 to 24 fuel elements become spent each year. It can be expected that at intervals of about 2 years, spent fuel shipments will be made to a Department of Energy reprocessing facility. Shipments, each comprising about 24 elements, are made over a 2-to-3-week period. Such shipments are made in a specially licensed fuel cask and in accordance with Department of Transportation (DOT) and NRC regulations.

Spent elements are stored in the pool in wall racks that are spaced to eliminate the possibility of criticality even with dropped fuel elements When these elements are prepared for shipment, the fuel is transferred to the canal where end fittings are cut off. The fuel and fittings are temporarily stored in the canal for subsequent shipment and/or removal.

9.2 Compressed Air System

Dried and filtered air at 100 psi is supplied to the reactor building and hot laboratory by two, 250-ft^3/min air compressors located in the facility boiler house. If the primary compressor fails to operate or more air is required than can be supplied by the one compressor, the standby unit automatically cuts in.

9.3 Fire Protection System

There are fire hydrants located immediately outside the reactor building and hose stations located inside. These are supplied by the municipal water system. Should this supply fail, a 100,000-gal reservoir located on a hill above the facility can provide water to the system. In addition there are portable fire extinguishers located throughout the facility.

All reactor operating personnel are routinely given fire-fighting instruction. Periodic inspections are held to ascertain that the amount of combustible material in and around the reactor building is held to a necessary minimum.

Notification of any fire in the facility is telephoned to the main switchboard located in the administration building. The switchboard operator initiates the fire alarm and calls the Sterling Forest Fire Department. Fire department personnel are given periodic familiarization tours of the facility.

9.4 Conclusion

On the basis of the above, the staff concludes that the UCS reactor facility's auxiliary systems are adequate to support reactor operations in a safe and reliable manner.

10 EXPERIMENTAL PROGRAMS

The reactor serves as a source of radiation for research, radiochemical and radiopharmaceutical production. In addition to in-pool irradiation capabilities, experimental facilities include a thermal column, six beam tubes, and three pneumatic transfer systems.

10.1 Experimental Facilities

10.1.1 Pool Irradiation

The open end of the pool permits bulk irradiations and provides storage space for irradiated fuel and activated equipment. A permanent shelf is attached to one end of the core support assembly to facilitate positioning samples in reproducible geometries. The decision to perform experiments in the reactor pool--as opposed to using a pneumatic transfer system or a beam tube--is dictated by specimen size and the type and intensity of radiation field required. The actual placement of experiments or samples in the core region is controlled by their radiation needs and any effect on core excess reactivity, which is limited by the Technical Specifications. As indicated in Section 4, the core grid plate has 40 core coolant holes that can be adjusted to direct coolant to those sections of the core that have ongoing sample irradiations that may require special cooling.

10.1.2 Thermal Column

A steel and aluminum chamber is cast integrally within the stall wall and magnetite concrete shield at core level. This chamber is square and extends horizontally from the inside wall of the stall to the outer surface of the concrete shield. The inner surfaces of the chamber are lined with boral sheet. A closely packed arrangement of graphite blocks is stacked within this boral liner for the length of the chamber.

With the reactor positioned against the inner face of the horizontal chamber, the graphite moderates the energetic neutrons escaping from the reactor, providing an external beam of thermal neutrons for experimental use. The ventilation system maintains a negative pressure on the thermal column so that air flows into the chamber and is discharged through a filter to the exhaust stack (see Figure 4.4).

10.1.3 Beam Tubes

Two 8-in.-diameter and four 6-in.-diameter beam tubes radiate in a horizontal plane outward from the reactor core through the shield wall. The tube assembly consists of an embedded stainless-steel sleeve, a retractable aluminum liner, and a set of shielding plugs of canned magnetite concrete and lead.

The beam tubes can be filled with demineralized water to reduce the number of shielding plugs required and to eliminate voids near the reactor core. When not water-filled, the beam tubes are continuously vented to the same filtered ventilation system that exhausts the thermal column to prevent buildup of ^{41}Ar.

When the beam tubes are used, external shield walls or beam catches are installed to control radiation levels in the experimental areas.

10.1.4 Pneumatic Transfer Systems

One 1-1/2-in. and two 3/4-in. pneumatic tubes are available to deliver samples for irradiation into the high-flux region of the core. The samples can be inserted or removed while the reactor is in operation through a constant exhaust system that is vented through a filter to the exhaust duct. Each of these pneumatic tube systems has automatic timing controls and shielded containers for timing the irradiation duration and for receiving the irradiated specimens, respectively.

10.2 Experiment Review

A Nuclear Safeguards Committee appointed by and reporting to the General Manager provides an independent review and audit of reactor facility operations (see Section 13 for additional details).

All new experiments or classes of experiments that can affect reactivity or result in the release of radioactivity and the respective procedures thereof must be reviewed by this Committee before insertion into the reactor. The approval of the Reactor Supervisor is required before previously approved experiments can be inserted into the reactor.

In addition to ensuring safe reactor use, this review and approval process allows personnel specifically trained in radiological safety and reactor operations to consider and recommend alternative operational conditions (such as different core positions, power levels, or irradiation times) that might decrease personnel exposure and/or the potential release of radioactive materials to the environment.

10.3 Conclusion

The staff concludes that the design of the experimental facilities, combined with the detailed review and administrative procedures applied to all research activities, is adequate to ensure that experiments (1) are not likely to fail, (2) are not likely to release significant radioactivity to the environment, and (3) are not likely to cause damage to the reactor systems or its fuel. Therefore, the staff considers that reasonable provisions have been made so that the experimental programs and facilities do not pose a significant risk of radiation exposure to the public.

11 RADIOACTIVE WASTE MANAGEMENT

Radioactive waste resulting from reactor operations is either discharged to the environment in gaseous form or shipped to a licensed disposal site in a solid form. The small amounts of radioactive liquid waste are evaporated. The resulting concentrates are mixed with cement to form a solid and disposed of as solid waste.

11.1 Waste Generation and Handling Procedures

11.1.1 Airborne Waste

The potential airborne waste includes gaseous ^{16}N and ^{41}Ar, fission products from tramp uranium and neutron-activated dust particulates. No fission products escape from the fuel cladding during normal operations. The radioactive airborne waste is produced principally by the neutron irradiation of the air dissolved in the pool water and the air and airborne particulates in the thermal column and beam tubes.

Exposure to ^{41}Ar and the limited fission products by personnel is minimized by constantly sweeping the air from the reactor room, from above the pool surface, and from the experimental facilities. A separate ventilation system provided for the thermal column and beam tubes is exhausted into the main ventilation system exhaust. A 5,000-ft^3/min supply duct, installed beneath the reactor bridge, removes the ^{41}Ar and fission products evolved from the surface of the pool above the reactor core. The ventilation system was described in detail in Section 6.

The coolant flow down through the core to the holdup tank and heat exchanger at elevated power levels precludes the release of measurable quantities of ^{16}N, as this isotope ($T_{1/2}$ = 7.1 sec) has essentially decayed within the piping system by the time the water returns to the open pool.

As shown in Figure 6.1, during normal operations, the exhaust from the reactor facility is combined with the hot laboratory exhaust and the stream is discharged from a tall stack located on the ridge above the complex. This combined effluent stream is continuously monitored. The reactor discharge accounts for about 40% of the total air flow and for less than 10% of the activity released from the stack. Ninety percent of the radioactivity released from the stack is generated in the hot laboratory.

11.1.2 Liquid Waste

The radioactive liquid waste disposal system in shown in Figure 11.1. Normal reactor operations produce no radioactive liquid waste. However, many associated activities conducted within the reactor facility are capable of generating such waste. The largest volume of contaminated water from the reactor systems is produced by the regeneration of the demineralizer.

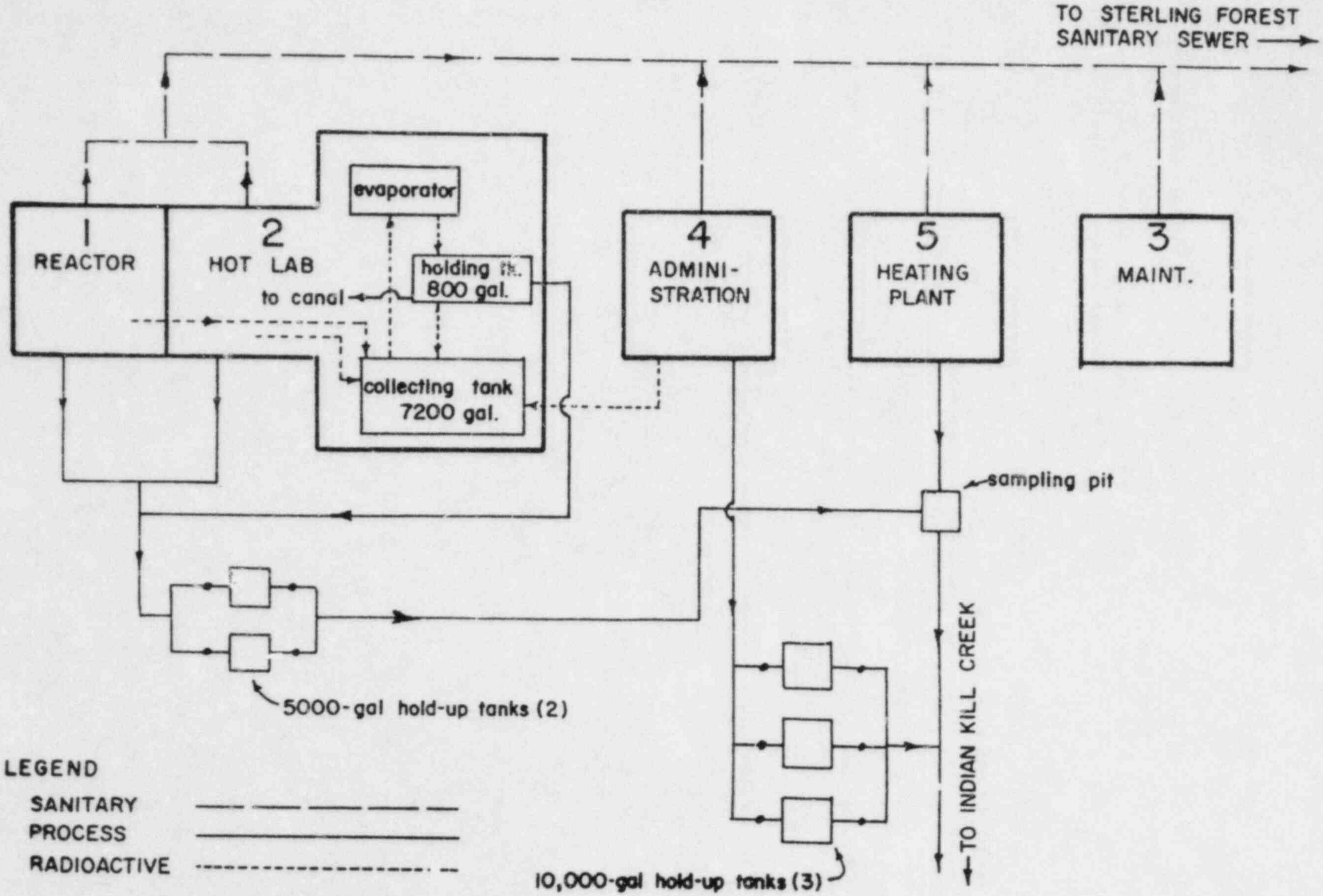

Figure 11.1 Radioactive liquid waste disposal system

All radioactive liquid waste from the reactor (and from the hot laboratory) is collected in a 7,200-gal tank located in the basement of the hot laboratory. These waste liquids are treated by evaporation and/or ion exchange to reduce the volume to a quantity that can be solidified in concrete.

All nonradioactive waste liquids from these facilities, including the decontaminated process waste stream, are collected in one of two 5,000-gal tanks that are operated on a collect-hold-sample-analyze-release philosophy. If unacceptable concentrations are detected, the contents can be pumped to the 7,200-gal waste storage tank and processed by the evaporation/ion exchange system. This provides a positive method of preventing accidental discharge of radioactive liquids to the off-site environment.

11.1.3 Solid Waste

Low-level solid waste generated as a result of reactor operations consists primarily of ion exchange resins and filters, contaminated paper and gloves, and occasional small, activated components, in addition to the solidified evaporator concentrates. These are packaged and stored on-site in a special waste storage facility and are eventually shipped to an approved disposal site in accordance with applicable NRC and DOT regulations.

High-level solid waste generated by routine reactor operations consists of handling 20 to 24 spent fuel elements per year. Spent elements are stored in the open end of the reactor pool until the accumulation justifies shipment to a Department of Energy fuel reprocessing plant. At this time the end plugs are removed.

11.2 Conclusions

The staff concludes that the waste management activities of the UCS reactor facility have been conducted and are expected to continue to be conducted in a manner consistent both with 10 CFR 20, with as-low-as-is-reasonably-achievable (ALARA) principles (see Section 12.1), and with the methods and principles of ANSI/ANS 15.11, "Radiological Control at Research Reactor Facilities," 1977.

Because ^{41}Ar is the principal potentially significant radionuclide released by the reactor to the environment during normal operations, the staff has reviewed the history, current practice, and future expectations of operations. The staff concludes that the doses in unrestricted areas as result of actual releases of ^{41}Ar have always been a small fraction of the limits specified in 10 CFR 20 when averaged over a year. Furthermore, the staff's conservative computations of the dose beyond the limits of the reactor facility give reasonable assurance that potential doses to the public as a result of ^{41}Ar will not be signficant. This dose has been verified by calculations conducted in connection with the UCS renewal of their Special Nuclear Materials license, SNM-639 (NRC, 1983).

12 RADIATION PROTECTION PROGRAM

UCS has a structured radiation safety program with a health physics staff equipped with radiation detection equipment to determine, control, and document occupational radiation exposures at its reactor and hot laboratory facilities. In addition, detectors located throughout the complex monitor both liquid and airborne effluents at the points of release to comply with applicable regulations. UCS also has an environmental monitoring program to verify that radiation exposures in the unrestricted areas around the facility are within regulations and guidelines and to confirm the results of calculations and estimates of environmental effects resulting from the research programs.

12.1 ALARA Commitment

The corporate administration has formally established the policy that all operations are to be conducted in a manner to keep all radiation exposures ALARA. All proposed experiments and procedures at the reactor are reviewed for ways to minimize the potential exposures of personnel. All unanticipated or unusual reactor-related exposures are investigated by both the health physics and the operations staffs to develop methods to prevent recurrences.

12.2 Health Physics Program

12.2.1 Health Physics Staffing

The normal full-time health physics staff at the UCS reactor facility consists of two professionals and several technicians. The onsite staff has sufficient training and experience to direct the radiation protection program for a research reactor. The health physics staff has been given the responsibility, authority, and adequate lines of communication to provide an effective radiation safety program.

The health physics staff provides radiation safety support to the entire research complex, including a large hot laboratory facility. The staff believes that the UCS health physics staff is adequate for the proper support of safe reactor operation.

12.2.2 Procedures

Detailed written procedures have been prepared that address the health physics staff's various activities and the support that it is expected to provide to the routine operations of the UCS reactor facility. These procedures identify the interactions between the health physics staff and the operational and experimental personnel. They also specify numerous administrative limits and action points as well as appropriate responses and corrective action if these limits or action points are reached or exceeded. Copies of these procedures are readily available to the operational and research staffs and to the health physics and administrative personnel.

12.2.3 Instrumentation

The UCS reactor facility has a variety of detecting and measuring instruments available for monitoring potentially hazardous ionizing radiation. The instruments and their calibration procedures and techniques ensure that any credible type of radiation and any significant radiation intensities will be detected promptly and measured correctly.

12.2.4 Training

All reactor facility personnel are given an indoctrination in radiation safety before they assume their work responsibilities. Additional radiation safety instructions are provided to those who will be working directly with radiation or radioactive materials. The training program is designed to identify the particular hazards of each specific type of work to be undertaken and methods to mitigate their consequences. Retraining in radiation safety is provided as well. As an example, all reactor operators are given an examination on health physics practices and procedures at least every 2 years. Retraining requirements are determined by the examination results. All of the above-mentioned radiation safety training is provided by the health physics staff.

12.3 Radiation Sources

12.3.1 Reactor

Sources of radiation directly related to reactor operations include radiation from the reactor core, sources activated in the core, ion exchange columns, filters in the water and air cleanup systems, and radioactive gases, primarily ^{41}Ar and fission products from tramp uranium.

The reactor fuel is contained in aluminum cladding. Radiation exposures from the reactor core are reduced to acceptable levels by water and concrete shielding. The ion exchange resins and filters are routinely changed before high levels of radioactive materials have accumulated, thereby minimizing personnel exposure.

Personnel exposure to the radiation from chemically inert ^{41}Ar is limited by dilution and prompt removal of this gas from the reactor room and experimental areas and its discharge to the atmosphere, where it is diluted and diffused further before reaching offsite occupied areas.

12.3.2 Extraneous Sources

Sources of radiation that may be considered as incidental to the normal reactor operation but that are associated with reactor use include radioactive isotopes produced for research, activiated components of experiments, and activated samples or specimens.

Personnel exposure to radiation from reactor products, as well as from the required manipulation of activated experimental components, is controlled by stringently developed and reviewed operating procedures that use the standard protective measures of time, distance, and shielding.

Targets subsequently handled in the hot laboratory represent another source of radiation. Operations and health physics personnel move freely between the reactor and hot laboratory facilities and many have assigned task in both areas.

12.4 Routine Monitoring

12.4.1 Fixed-Position Monitors

The UCS nuclear reactor facility uses several fixed-position radiation monitors and continuous air monitors. These include an air particulate monitor in the reactor room and two monitors on the bridge above the reactor. There are several fixed-head air samplers in the reactor building that are changed and analyzed daily.

Area radiation monitors and/or criticality monitors are at strategic locations throughout the building in regions where radiation levels might increase and reflect an abnormality or hazard. All monitors have adjustable alarm set points and read out in the control room.

12.4.2 Experimental Support

The health physics staff participates in planning of experiments by reviewing all proposed procedures for ways to minimize personnel exposures and limit the generation of radioactive waste. Approved procedures specify the type and degree of health physics involvement in each activity. As examples, standard operating procedures require that changes in experimental setups include a survey by health physics personnel using portable instrumentation, and all items removed from the reactor room or beam room must be surveyed and tagged by health physics personnel.

12.4.3 Special Work Permits

Occasionally, one-of-a-kind, short-term, low-to-intermediate-risk tasks such as simple but nonroutine maintenance activities in potential radiation or contamination areas are performed, but only after staff review. The work is then performed with constant health physics coverage or, in some cases, with intermittent health physics support under a special work permit (SWP). Each SWP requires documentation of the radiation safety review and concurrence of operations personnel; the SWP includes details of any special actions or precautions that are needed to minimize personnel radiation exposures and/or the spread of radioactive contamination.

12.5 Occupational Radiation Exposures

12.5.1 Personnel Monitoring Program

The UCS reactor facility personnel monitoring program is described in its Radiation Safety Instructions. The program requires that personnel exposures be measured by the use of film badges assigned to individuals who might be exposed to radiation. In addition, thermoluminescent dosimeters (TLDs), non-self-reading pocket chambers, and instrument dose rate and time measurements are used to ensure that administrative occupational exposure limits are not exceeded. These limits are in conformance with 10 CFR 20.

12.5.2 Personnel Exposures

During each of the last 5 years, 12 to 15 operators and health physics personnel performing work in both the reactor and hot laboratory areas have received whole-body exposures in excess of 1.25 rem. These annual exposures have ranged from about 1.5 to 4.5 rem, with an average near 2.5 rem. The radiation dose standards in 10 CFR 20.101 are 5 rem per year for whole-body dose.

12.6 Effluent Monitoring

12.6.1 Airborne Effluents

As discussed in Section 11, airborne effluents from the reactor facility consist principally of activated gases. In the emergency mode, the entire effluent stream is filtered to remove most particulate materials and iodines before discharge to the environment through the stack. This filter installation consists of a roughing filter to reduce the loading of the final filters, a high-efficiency particulate air (HEPA) filter (which removes more than 99% of the solid matter in the air stream), and a bank of activated carbon filters to remove any iodine.

The combined facility airborne effluent is monitored to provide prompt indication of any abnormal concentrations being discharged to the environment. This is accomplished by withdrawing a representative side stream from the main discharge duct, passing this through particulate, iodine and gaseous monitors, and returning it to the suction side of the exhaust fans. The particulate monitor is a moving paper filter passing in front of an anthracene scintillation crystal. The iodine monitor is a charcoal filter viewed by a scintillation detector. The gas monitor is a shielded volume containing a sodium iodide crystal. The monitors indicate on meters having adjustable alarm set points. These outputs are repeated on chart recorders in the reactor control room.

The individual reactor and hot laboratory effluent streams are sampled with fixed-head particulate and charcoal (for iodine) filters. These filters are normally changed and analyzed weekly with the results used for the official effluent reports.

12.6.2 Liquid Effluents

As stated in Section 11, the reactor generates very limited radioactive liquid waste during routine operations. However, all radioactive liquids from the reactor and the hot laboratory are collected in a large holdup tank for subsequent treatment by ion exchange and/or evaporation to reduce the volume of radioactive solution. The concentrated liquid wastes are solidified, monitored, tagged, and stored for eventual offsite shipment. Decontaminated process waste is collected, sampled, and analyzed to ensure that levels of contained radioactivity are below the levels specified in 10 CFR 20.303 before release to the chemical sewer that discharges to Indian Kill Brook.

12.7 Environmental Monitoring Program

An environmental monitoring program was started in 1957, before the initial criticality of the reactor, and has continued to the present with minor changes. The current program is directed toward measuring airborne activities and direct radiation as well as potential ingestion pathways to man.

12.7.1 Airborne Release and Direct Radiation

Two sampling stations for radioiodine (charcoal canister) and for particulates are operated continuously; both are located downwind of the prevailing wind direction, one near the site boundary and the other at the nearest public habitation. Samples are collected every 7 days for analysis. For direct radiation monitoring, the stations also have gamma dosimeters that are read at 1-to-3-month intervals.

12.7.2 Ingestion Pathways

Water samples are taken monthly for gross-beta analysis from five separate locations, namely, Indian Kill Brook inlet and outlet, Warwick Brook, Sterling Lake, and the Ramapo River.

These measurements are supplemented by the extensive environmental monitoring program conducted by the New York State Department of Health on water, milk, and certain flora and fauna.

12.8 Potential Dose Assessments

Natural background radiation levels in the Bear Mountain area result in an exposure of about 125 mrem per year to each individual residing there. At least an additional 7.5% (approximately 8 to 10 mrem per year) will be received by those living in brick or masonry structures. Medical diagnosis X-ray examinations will add to this natural background exposure.

Conservative calculations by the staff, based on the amount of ^{41}Ar released by the reactor operations, predict a maximum annual exposure of about 1 mrem in the unrestricted areas. The radiation levels measured by the environmental radiation dosimeters have shown fluctuations up to 30% of normal background. However, these variations cannot be readily correlated with reactor operations.

12.9 Conclusion

The staff concludes that (1) radiation protection receives appropriate support from the corporate administration, (2) the program is properly staffed and equipped, (3) the UCS health physics staff has adequate authority and lines of communication, (4) the procedures are properly integrated into the research plans, and (5) surveys verify that operations and procedures achieve ALARA conditions.

The staff also concludes that the effluent and environmental monitoring programs conducted by Union Carbide personnel are adequate to promptly identify significant releases of radioactivity and confirm possible effects on the environment, as well as to predict maximum exposures to individuals in the unrestricted areas. These predicted maximum exposure levels are a small fraction of applicable regulations and guidelines specified in 10 CFR 20.

Thus, the staff concludes that the UCS reactor radiation protection program is acceptable, and there is reasonable assurance that the personnel and procedures will continue to protect the health and safety of the public.

13 CONDUCT OF OPERATIONS

13.1 Organization

The organization for the management and operation of the UCS reactor facility is shown in Figure 13.1.

Four levels of authority provided are

Level 1 - General Manager
Level 2 - Operations Manager
Level 3 - Reactor Supervisor
Level 4 - Operating Staff

In addition to the above line of authority, there are two independent groups that oversee reactor operators: the Health, Safety, and Environmental Manager and the Nuclear Safeguards Committee.

The General Manager is responsible for the overall policy and safe operation of the UCS reactor facility and the hot laboratory.

The Operations Manager is responsible for ensuring the safe operation of the reactor and coordinating the irradiation services required by the hot laboratory.

13.2 Radiation Safety Staff

The radiation safety staff is supervised by the Health, Safety, and Environmental Manager. Within this group is the Health Physics staff, which is responsible for ensuring minimum exposures of onsite and offsite personnel commensurate with the activities conducted within the UCS complex. The health physics activities include instruction, radiation surveys, controlling the discharge of effluents, maintenance and calibration of radiation detection equipment, checking of incoming and outgoing shipments for contamination and radiation, maintenance and operation of meteorological equipment and data, and maintenance of radiation records for personnel and environment.

13.3 Nuclear Safeguards Committee

The Nuclear Safeguards Committee is responsible for the independent review and audit of reactor facility operations. It is composed of five members whose knowledge and experience enable them to provide a broad spectrum of expertise in reactor technology. The Committee reports directly to the General Manager.

The Committee reviews

(1) proposed changes in equipment, systems, tests, experiments, or procedures to determine that they do not involve an unreviewed safety question

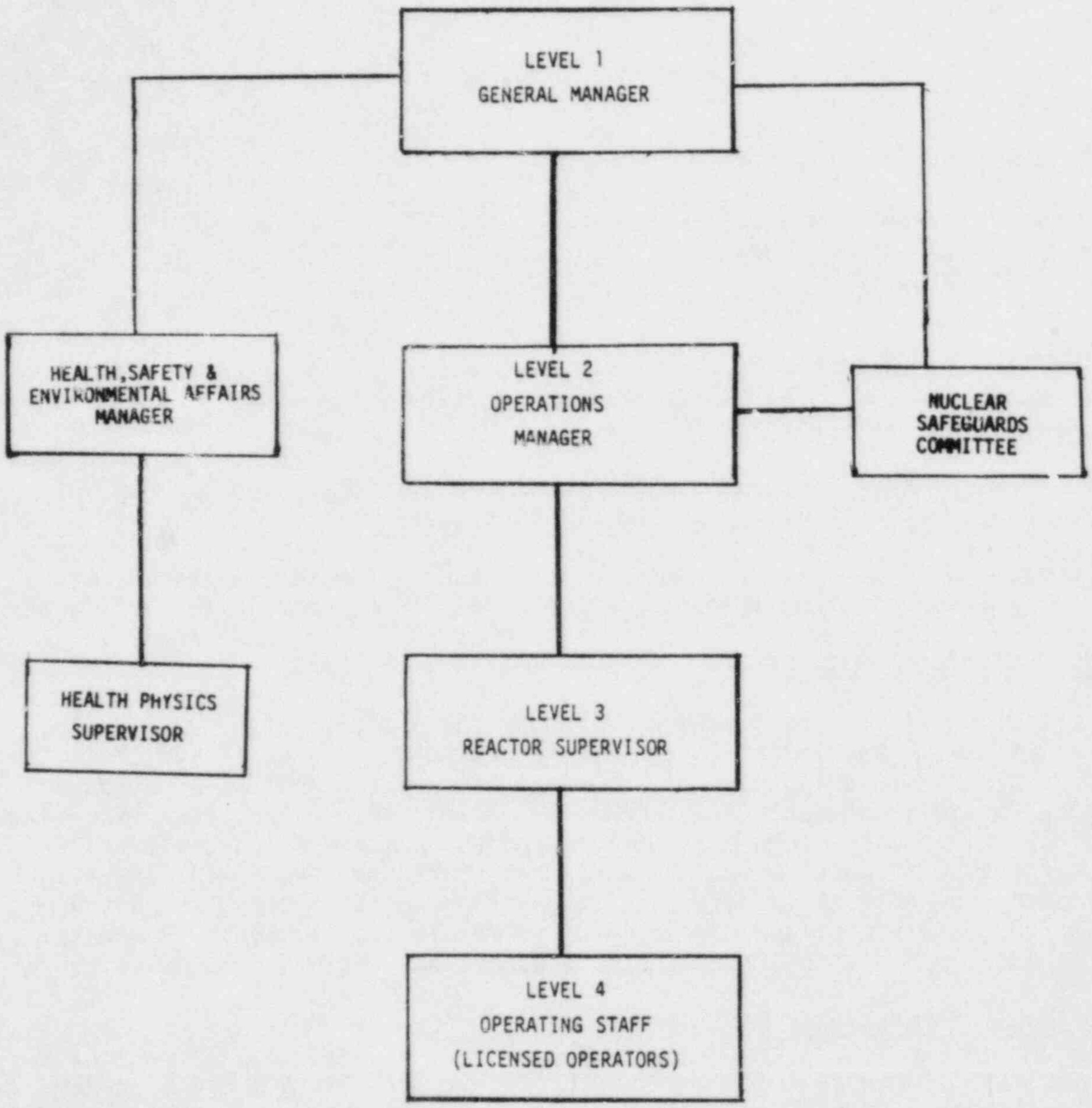

Figure 13.1 UCS organization

(2) all new procedures and major revisions having safety significance, proposed changes in reactor facility equipment or systems having safety significance

(3) tests and experiments that have not been previously reviewed

(4) proposed changes in Technical Specifications, license, or charter

(5) violations of Technical Specifications, license, or charter and violations of internal procedures or instructions having safety significance

(6) operating abnormalities having safety significance and audit reports

(7) reportable occurrences listed in the Technical Specifications

The audit function includes selective and comprehensive examination of operating records, logs, and other documents and, as necessary, discussions with responsible personnel. Items that are audited include

(1) the conformance of facility operations to the Technical Specifications and applicable license or charter conditions, at least once per calendar year (interval not to exceed 18 months)

(2) the retraining and requalification for the operating staff, at least once every other calendar year (interval not to exceed 30 months)

(3) the results of actions taken to correct deficiencies occurring in reactor facility equipment, systems, structures, or methods of operations that affect reactor safety, at least once per calendar year (interval not to exceed 18 months)

(4) the reactor facility Physical Security Plan and implementing procedures at least once every other calendar year (interval not to exceed 30 months).

13.4 Training

The requirements for requalification and retraining of the operating staff are indicated in the Technical Specifications. The UCS Operator Requalification Program, submitted as part of the renewal application, indicates that training and requalification of staff is conducted biennially. The UCS program has been reviewed and approved by the staff and determined to meet the requirements of the ANS 15.4, "Selection and Training of Personnel for Research Reactors."

13.5 Operational Review and Audits

As stated in Section 13.3, reactor safety review and audits are performed by the Nuclear Safeguards Committee.

13.6 Emergency Planning

10 CFR 50.54 and Appendix E to 10 CFR 50 require that nonpower reactor applicants/licensees develop and submit emergency plans. The licensee submitted a

plan that was developed following the recommended guidance in RG 2.6 (1979, For Comment Issue) and guidance in American Nuclear Society (ANS) 15.16 (1978 Draft). However, both of these guides have been revised. (Revision 1 to RG 2.6 was issued for comment in March 1982; Draft 2 of ANS 15.16 was issued November 1981.) By letter dated September 1982, UCS submitted a revised Emergency Plan for staff review and approval. This was amended by letter dated August 8, 1983.

On the basis of its review and evaluation, the staff concludes that the emergency plan for the UCS facility, dated September 3, 1982, as amended, demonstrates that the licensee has the capabilities to assess and respond to emergency events. The plan provides assurance that necessary emergency equipment is available and describes a plan of action to protect the health and safety of workers and the public. For the above reasons, the staff concludes that the emergency plan for the UCS facility meets the requirements of the regulations and, therefore, is acceptable.

13.7 Physical Security Plan

UCS has established and maintains a program designed to protect the reactor and its fuel and to ensure its security. The NRC staff has reviewed the plan and visited the site. The staff concludes that the plan, as amended meets the requirements of 10 CFR 73.67 for special nuclear materials of moderate strategic significance. The UCS license authorization for reactor fuel falls within that category. Both the Physical Security Plan and the staff's evaluation are withheld from public disclosure under 10 CFR 73.21.

13.8 Conclusion

On the basis of the above discussions, the staff concludes that the licensee has sufficient training experience, management structure, and procedures to provide reasonable assurance that the reactor will be managed safely and will cause no significant risk to the health and safety of the public.

14 ACCIDENT ANALYSIS

In establishing the limiting safety system settings and the limiting conditions for operation for the UCS reactor, the licensee analyzed potential transients to ensure that these events would not result in any safe limits being exceeded. Hypothetical accidents and their effects on the core and the health and safety of the public were similarly analyzed. In addition, the licensee analyzed the potential effects of natural hazards and minor accidents.

Among the accidents postulated, the one with the greatest potential effect on the environment and the unrestricted area outside the exclusion area is the meltdown of an amount of fuel containing 10% of the total accumulated fission products with a concurrent total loss of water. A less severe accident involving an in-core fueled experiment also has been analyzed and is probably more credible than the meltdown accident.

The meltdown accident is designated as the maximum hypothetical accident (MHA). The MHA is defined as an accident for which the risk to the public is greater than that from any other credible event. The staff assumed that the accident occurred without trying to describe or evaluate the mechanistic details of the accident or the probability of its occurrence. Only the consequences are described.

In summary, the following postulated transients and accidents were evaluated:

(1) natural phenomena
(2) minor accidents
(3) maximum startup accident
(4) credible serious accidents
(5) in-core fueled experiment accident
(6) meltdown accident (MHA)

14.1 Natural Phenomena

The licensee has considered the potential effects of windstorms, floods, and earthquakes on the UCS reactor. The licensee concludes that the hazards from these are negligible. As indicated in Section 3, the staff agrees with this conclusion.

As indicated in Section 2.3, no strong earthquakes have occurred in the area. The reactor pool is placed on very firm, hard rock. In addition, should a violent shock occur, it would cause the reactor to fail safe either from power failure or because the rods would drop from the magnets. An earthquake event of sufficient magnitude (MMI VII) that it would result in the loss of pool water through cracks in the concrete and a simultaneous break in the pool water seal was considered in the MHA accident, described in detail in Section 14.6. The staff concurs with the licensee's conclusion that the combination of earthquake rarity, low seismic intensity, rock foundations for the reactor building, and structural design of the pool renders significant hazard from seismic events to be remote.

A shock of MMI VII could cause a leak or break in one or more of the four primary cooling system pipes, resulting in draining of the pool if there were no intervening valves. An analysis by the licensee for clean break of 4 to 10 in. lines concludes that the minimum time to drain the pool is 8 min. The decay heat generated in the scrammed reactor would be about 1% of that at operating power and, based on work by Wett on the Oak Ridge Research Reactor fuel elements (ORNL-2892), the maximum fuel temperature would reach about 950°F without any emergency cooling, which is below the temperature required for fuel melting (1,180°F). Two manually operated spray nozzles located at the top of the pool at both reactor operating positions can flood the core with about 100 gal/min of water and can be operated from outside the reactor building.

On the basis of the above and the licensee's analysis, the staff concludes that natural phenomena do not constitute a hazard to safe operation of the UCS reactor. Therefore, there is reasonable assurance that such events pose no significant threat to the health and safety of the public.

14.2 Minor Accidents

Minor accidents are those whose results are less severe than those identified as credible serious accidents (Section 14.4). These are discussed briefly in the following sections.

14.2.1 Loss of Beam Tube Ventilation

If the ventilation system fails, radioactive gases from the beam tubes can contaminate the reactor room atmosphere. The major source of the contamination is ^{41}Ar from the $^{40}Ar(n,\gamma)\,^{41}Ar$ reaction. If the content of an 8-in. beam tube (assumed unplugged volume of 0.75 ft^3, or 21.4 l) is released to the reactor building atmosphere by removal of the beam port cover, the resulting ^{41}Ar concentration would be above that of Table 1 of Appendix B of 10 CFR 20. To prevent this (1) the reactor is equipped with a warning light that indicates beam tube ventilation failure and (2) administrative rules allow the beam ports to be opened only after reactor has been shut down and the ^{41}Ar has decayed to safe levels.

14.2.2 Loss of Pool Surface Ventilation

Failure of the pool-top ventilating system does not constitute a health hazard. If fission product gases are released from damaged fuel elements at the same time, the two high radiation monitors located under the bridge would indicate "high radiation" at much lower activity levels than if the sweep ventilation system were operating.

14.2.3 Loss of Fuel Cladding Integrity

This accident assumes loading of a damaged or defective fuel element or the cladding being corroded or eroded from the fuel element. This accident would result in an increase in radioactivity in the pool primary coolant. The regularly scheduled water sampling and analysis program would detect the increase before the activity level became significant. In addition, the pool top detectors or the continuous stack and building air monitors would detect an increase in any airborne radioactivity.

14.2.4 Drop of Pool Water Level

Loss of water by accidental draining into the holdup tank would drop the water level by about 3.7 ft. This would not result in a significant increase in the radiation level above the pool. Further, if the reactor is operating, the low-level water scram would be activated.

An empty beam tube could conceivably be sheared off by a falling object, causing a drop in water level that could expose the core, which would result in high radiation levels in the building. This accident can occur only with beam tubes that have unbolted cover plates and are unplugged, because a plugged and bolted tube can withstand pool water pressure with only some minor leakage around the outer plug. Administrative procedures require that the reactor be in the open end of the pool whenever the beam tubes are unbolted and unplugged, thus eliminating the possibility of a beam-tube shear-off accident.

In the case of an unnoticed beam tube leak, a 6-in. drop in the pool water level will result in a pool water low-level alarm. There is sufficient time with a beam tube leak to permit the reactor to be moved to the open pool position before the pool water level drops appreciably.

Water can be pumped from the holdup tank (in essence from the pool) into the storage tank. Because of the valving and pump system, this cannot be done accidentally. The low-level water alarm and scram discussed above will operate if this occurs.

14.2.5 Fuel Handling Accident

A fuel handling accident was considered. It included dropping a fuel element out of a transfer cask onto the operating floor. It was concluded that although the fuel cladding may rupture to the drop, it would not result in a release of fission products or result in a dose that would be greater than that considered for the MHA in Section 14.6.

14.2.6 Explosions

The reactor concrete biological shield is an extremely efficient explosion barrier. It is not credible that any nominally-sized explosion that is external to the shield could damage the core.

14.2.7 Conclusion

On the basis of the above information, the staff concludes that the minor accidents described above will not result in fuel melting, cladding failures, or significant radiation exposures. Therefore, there is reasonable assurance that these events would pose no significant threat to the health and safety of the general public or the UCS reactor staff.

14.3 Maximum Startup Accident

In this accident analysis the licensee assumes that as a result of a circuit malfunction, all rods are able to be withdrawn simultaneously. It is further assumed that (1) the reactor is initially at a very low (source) power level, (2) no rod inhibits are operative, (3) at criticality the rods are in the most effective region (50% withdrawn), and (4) the total rod bank maximum reactivity worth is 11.6% $\Delta k/k$. These assumptions maximize the accident and make the analysis conservative. The analysis showed that with a 200% of power scram trip level, the total energy of the excursion is only 15 MW-sec. This is about 2.5 times less than the so-called "BORAX threshold of 37 MW-sec." An additional analysis was made of the self-limited excursion that would result if no safety system were present. The results of the SPERT-I and SPERT-IV tests were used in this analysis (Phillips Petroleum Co. IDO-16528 and IDO-17000). The self-shutdown characteristics of the UCS reactor core serve to limit the power and energy generated in such an excursion.

In particular, the fuel-plate surface temperature for the reactor period corresponding to the 200% of power scram level is less than 220°C, which is more than 400°C below the cladding melting temperature.

The staff has reviewed the licensee's analysis of the maximum startup accident and finds that the methodology used was appropriate and conservative. Therefore, the staff concludes that this postulated accident does not present a potential hazard to the health and safety of the public or UCS reactor personnel.

14.4 Credible Serious Accidents

Accidents that are credible and that could have serious implications include fuel element mishandling, improper fuel element loading, and experimental facility accidents.

14.4.1 Fuel Element Mishandling

Radiation hazards can result from the mishandling of irradiated fuel elements or experimental samples. The former have the greatest potential hazard. Although normal fuel handling procedures do not constitute a potential exposure hazard, several credible equipment malfunctions could result in a fuel element being removed from all shielding. These are considered in the following paragraph.

The building crane can be used to transport elements by hanging the fuel element handling tool (with attached fuel element) from the crane lift hook. During such an operation, personnel manipulate the crane controls while standing on the reactor bridge or at the pool side. At that time they are within 90 ft of the crane disconnect switch. The maximum height of the crane hook is 22 ft above the pool water surface. The fuel handling tool is 25 ft long because of the reactor pool depth. Thus, if the crane control circuit fails in such a manner as to cause the hook to reach its maximum height the fuel element could rise to within 3 ft of the pool water surface. A radiation field of 20 R/hour would result with a freshly discharged element, setting off the radiation alarm located under the pool bridge. Because it takes 1 min for the element to be raised to this position, the operator would have sufficient time to traverse the distance to the crane disconnect switch, thus decreasing the potential for radiation exposure from this accident. If the operator did not disconnect the crane travel switch, the radiation alarms would sound to evacuate the building.

It is conceivable that an element would be moved in a similar operation with a handling tool less than 25 ft long, although such a fuel handling tool is not now available. Given the event that the crane control circuit fails in the same way, the building evacuation alarm would sound when the radiation level reaches 5 R/hour, corresponding to an element about 4.5 ft from the pool water surface. This would warn personnel of the condition and give the operator sufficient time to reach the crane disconnect switch. The time from initiation of alarm until the element would reach the pool water surface is about 12 sec. If the crane operator leaves the building rather than disconnecting the crane power, the radiation level outside of the reactor building wall resulting from a freshly discharged element centered in the reactor building would be less than 160 R/hour. Assuming the operator does not leave the building until the element is exposed and that he takes an additional minute to reach an area of lower radiation, the estimated dose to the operator would be less than 3 rems. Once the building is emptied, there would be time to plan corrective action.

A similar accident, lifting the pool bridge with the crane, was considered. However, this mishap is prevented because the bridge structure is clamped to the bridge rails and cannot be inadvertently raised off the rails.

14.4.2 Fuel Element Loading Accidents

While the possibilities of a fuel loading accident are extremely remote. four loading conditions are considered in the context of the technique employed during fuel loading. To put the analysis of the accident potential of these loading operations in perspective, it should be noted that the total worth of the UCS reactor control rods is about 9.4%, the worth of the regulating rod is not more than 0.6% $\Delta k/k$, and fuel loading is done with the all control rods fully inserted. The staff has considered the consequences of a dropped-fuel-element accident and has determined that the consequences are less severe than those analyzed in the following paragraphs.

The first loading operation considered assumes an approach to criticality with a new core configuration. In this operation, elements are added at the outer faces of in-place elements. After the addition of no more than two elements, a criticality experiment is performed, which permits an estimate of the fuel mass required for criticality. The procedure is repeated until criticality is achieved.

The worth of two elements added to the outer face is not more than 5% $\Delta k/k$. It is improbable that a loading accident would occur during the buildup of a new core because it would require the insertion of at least four additional elements without a criticality check simply to offset the negative worth of the fully inserted rods.

The second loading operation considered is reloading in a known core configuration. The standard procedure requires that all rods are to be fully inserted and spent elements are to be replaced with elements with fuel masses within ±10% of that of the spent elements at the beginning of the previous cycle. There is no change in core size or geometry. The largest loading error would, therefore, involve a net increase of core mass of 10%, or a maximum of 540 g ^{235}U. If all mass deviations were +10%, the reactivity increase would be 8% $\Delta k/k$, which approaches the negative worth of all the inserted control rods.

Therefore, the maximum number of elements that can be loaded without a criticality check is administratively limited to 15, or about half the core, which restricts the possible reactivity increase to a more conservative level of about 4% $\Delta k/k$.

The third loading condition considered is the replacement of sufficient elements to allow reactor restart with significant xenon poisoning present. This is essentially the same situation as analyzed above for reloading a known core configuration. The 15-element criticality check replacement-restriction is applied to this situation to ensure that the reactivity change is limited to a safe value.

The fourth loading operation considered involves replacement of an element in a central core positioning (for example, replacement of a central flux trap). The procedure used requires that all control rods are fully inserted and that a minimum of three outside elements be removed for each central element to be inserted. A criticality check is made after the addition of a central element to determine how many of the outer elements are to be returned to the core. Operator error could result in loading a central element without removing the outer elements. This could result in a positive reactivity insertion of up to 4% $\Delta k/k$. Although this is a large fraction of the total worth of the inserted control rods it is not sufficient to cause an excursion.

14.4.3 Experimental Facility Accidents

Rapid activity increases resulting from incidents involving experiments in the core are considered to be the most credible cause for serious reactivity accidents. For these reactivity increases to result in a serious accident, they must take place in less than 50 msec (the release time of control rod electromagnets). Reactivity insertion times that are longer than the response time of the safety system require a coincident failure of the safety system to achieve serious consequences.

Reactor power response to potential reactivity transients associated with experiments are discussed in Section 14.3

14.4.4 Multiple and Sequential Failures of Safety Components

Of the many accident scenarios hypothesized for the UCS reactor, none produce consequences more severe than the accidents reviewed and evaluated as the MHA. The only multiple-mode failure of more severe consequences would be melting of more than one fuel element. No credible scenario contructed by the staff has included a mechanism by which the failure of integrity of one fuel element can cause or lead to the failure of additional elements. Therefore, if the melting of multiple fuel elements should occur, the failures would be random and not a result of the same primary event. Additionally, the reactor contains redundant safety-related measuring channels and control rods. The staff review has revealed no mechanism by which failure or malfunction of one of these safety-related components could lead to a failure of a second component.

14.4.5 Conclusion

On the basis of the above considerations, the staff concludes that there is no credible serious accident associated with fuel element mishandling, improper fuel element loading, or experimental facilities that would result in excessive radiation exposures or that could exceed the safety limits for the fuel.

14.5 In-Core Fueled Experiment Accidents

The licensee irradiates fuel-bearing samples for the production of radioisotopes for use in nuclear medicine. Amendment 16 to the facility Operating License (No. R-81) changed the Technical Specifications to allow the licensee to increase the quantity of iodine allowed per capsule to 1,000 Ci and to have a single encapsulation of the target material. As part of the review for Amendment 16, potential accidents involving the in-core target materials were analyzed by the staff. Two accident scenarios were considered. One analysis considered a release of the capsule contents while in the core (capsule melt) with gaseous iodine transport through 22 ft of water. The other analysis considered release from the capsule while in the core caused by mechanical damage, material defects, or improper seals with gaseous iodine transport through 10 ft of water.

14.5.1 Release From Capsule Melt

For the case where the capsule and its contents melt, it was assumed that 100% of the iodine was released to the reactor pool. The iodine release value should be much less inasmuch as the temperature is relatively low (54°F) and that any released iodine would still have to find, reach, and traverse any break or rupture in the capsule.

14.5.2 Release From Capsule Mechanical Damage

The release fraction for the noble gases in this scenario was estimated to be 2.5% based on thin-film oxide data. The upper bound on the release fraction of the iodine was taken to be equal to the noble gas release fraction. It was noted that if capsule integrity is lost, the area associated with the breach should be very small relative to the inside surface area of the capsule. Therefore, once the iodine is released from the oxide film of the target material, it must still reach the capsule break and ooze through it against the water pressure into the bulk coolant water.

14.5.3 Water Transfer Coefficient

Water transfer coefficients range from 10^{-2} to 10^{-4} for depths between 9 ft and 22 ft. There appears to be little dependency on the rate of iodine injection or the size of bubble formed during the release process. There is a large dependency on the carrier media. The more vapor in contact with the iodine formed, the smaller the transfer coefficient.

14.5.4 Results

For the capsule melt scenario, the amount of iodine reaching the surface of the pool is 10^{-4} of the total content assuming 100% release to the pool water. Therefore, for this accident scenario the amount of iodine released to the confinement atmosphere is 0.1 Ci.

For the capsule damage scenario with loss of capsule integrity in the transfer chute (shallowest depth of capsule transfer) and a release fraction of 2.5%, the amount of iodine reaching the pool surface. Thus, the total iodine released to the confinement atmosphere is (1,000 Ci $\times$ 2.5 $\times$ 10^{-6}) = 2.5 $\times$ 10^{-3} Ci.

The iodine reaching the pool surface will be

(1) diluted by the confinement building volume 272,000 ft^3 (7,700 m^3)
(2) plated out on the various surfaces, for a factor of 2 reduction
(3) further reduced by the emergency exhaust system flow (200 ft^3/min or 9.44 $\times$ 10^{-2} m^3/sec)
(4) reduced by absorbtion on the charcoal filters by a factor of approximately 20

Assuming uniform mixing, the iodine concentrations in the confinement building and the quantities leaving the charcoal filters are as shown in Table 14.1:

Table 14.1 Iodine releases

Accident	Confinement building concentration, Ci/m^3	Release rate to environment, Ci/sec
Capsule melt	1.3 $\times$ 10^{-5}	3.1 $\times$ 10^{-8}
Capsule damage	3.2 $\times$ 10^{-7}	7.6 $\times$ 10^{-10}

The concentration at the site boundary using the licensee's calculated dispersion factor of 1.8 $\times$ 10^4 sec/m^3 and a 0- to 2-hour time period yields the site boundary (250 m) concentrations as shown in Table 14.2:

Table 14.2 Site boundary iodine concentration

Accident	Site boundary concentration μCi/cm^3
Capsule melt	5.6 $\times$ 10^{-12}
Capsule damage	1.4 $\times$ 10^{-13}

10 CFR 20, Appendix B, Table II, specifies a permissible offsite concentration for ^{131}I of 10^{-10} μCi/Cm^3. Therefore, the exposure of a person standing offsite directly in the plume for a continuous 2-hour period is less than that allowed in 10 CFR 20.

A recent failure of a UCS fission product molybdenum irradiation capsule verified the conservativeness of the above assumptions and calculations. In that incident a target source ruptured, releasing $\frac{1}{7.3\times10^6}$ of the ^{131}I inventory to the reactor room. The staff evaluation for the amendment that allowed single encapsulation concluded, from the literature, that 1/10,000th of the iodine in the capsule would be released to the reactor room. This actual release was 700 times less than the calculated release, and resulted in a small fraction of 10 CFR 20 limits.

14.5.5 Conclusion

On the basis of the above analysis and the capsule failure incident, the staff concludes that calculations for ^{131}I concentrations in the event of an incident use conservative assumptions, and they indicate an offsite concentration that is only 1/100 to 1/10,000th of the maximum permissible concentration (MPC), as delineated in 10 CFR 20, Appendix B, Table II. Accordingly, the staff concludes that no significant hazard can occur from these operations. A singly encapsulated target capsule containing 1,000 Ci of iodine can be safely handled in the UCS reactor, and there is reasonable assurance that the operations can be conducted without endangering the health and safety of the public.

14.6 Meltdown Accident

As indicated previously, an accident leading to the loss of coolant resulting in a meltdown of a portion of the reactor fuel containing 10% of the total core-accumulated fission products is the MHA for the UCS reactor. In addition, the licensee hypothesized that the reactor has operated for a long time at a power level of 7.5 MW (150% of licensed maximum power). It is further assumed that 10% of the total core noble gases and 5% of the total core halogens are released from the fuel. Further, because of the loss of pool water, there is no reduction of halogens resulting from absorption in the coolant. These assumptions are considered to impressible to achieve because of operating limitations of 5 MWt and the high percentage of fission products assumed to released (NUREG/CR-1386).

The licensee's offsite thyroid and whole-body dose calculations were performed based on the following additional assumptions and guidelines:

(1) The iodine released is reduced by a factor of 2 because of plateout on the interior surfaces of the confinement building.

(2) Effluent, after mixing with building air, is released through the emergency exhaust system at a rate of 200 ft^3/min.

(3) Iodine is further reduced by a factor of 20 by the charcoal filters in the emergency exhaust system.

(4) Noble gases are unaffected by the emergency exhaust system.

(5) The confinement building free air volume is 7,700 m^3 (272,000 ft^3).

(6) Decay corrections are made only for ^{137}Xe and ^{138}Xe during the initial 2-hour release period, but are made for all isotopes for subsequent release periods.

(7) Burnup and buildup are taken into account for ^{135}Xe.

(8) The guidelines of RG 1.3 were followed to calculate doses at the site boundary distance of 250 m (775 ft) for the stack release height of 69 m (214 ft) above the reactor elevation.

The accumulated thyroid dose was calculated using the method in an AEC report (TID-24190). Thyroid uptake factor, thyroid mass, the effective ^{131}I decay constant, and the effective energy absorbed per ^{131}I disintegration were taken from the International Commission on Radiological Protection (ICRP, 1960) report, whereas the breathing rates were taken from RG 1.3. No allowance was made for the fraction of the iodine absorbed in the thyroid that would be eliminated with time. That is, it was assumed that all iodine absorbed by the thyroid remained in the thyroid until it decayed.

The thyroid and whole-body doses to an individual standing in the plume at the site boundary (250 m) for exposure times of 0-to-2 hours and 8-to-24 hours were calculated. The total whole-body doses shown in Table 14.3 are based on noble gases only because the iodine contribution to the whole-body dose is negligible.

Table 14.3 Whole-body dose at site boundary from exhaust plume radiation (rem)

	0-2 hours	2-8 hours	8-24 hours	Total
Dose (β-γ)	0.75	0.17	0.23	1.2 rem*

*Rounded to nearest 0.1 rem

The thyroid doses are shown in Table 14.4.

Table 14.4 Thyroid dose from plume iodine (rem)

	0-2 hours	2-8 hours	8-24 hours	Total
Dose	3.4	2.0	2.6	8.0

Thus, the total dose to the whole body and thyroid of a person standing at the site boundary for 24 hours would be 1.2 and 8.0 rem, respectively. There are no exposure limits for accident situations in 10 CFR 20. However, 10 CFR 20 and ICRP guidelines provide limits for an average annual dose to be less than or equal to 1.5 rem whole body and 8 rem/13 weeks thyroid, respectively.

The staff has reviewed and verified the licensee's MHA scenario and associated calculations. The staff concurs with the designation of this scenario as the MHA and has confirmed the validity of the calculations.

Although the calculations for thyoid dose are equal to the ICRP allowable dose, the staff considers that the dose calculations are conservative for the following reasons:

(1) No credit is taken for increase in effective stack height because of velocity of the effluent air stream.

(2) No credit for solution of iodine with any vapor or chemical combination with other reactive chemicals was considered. The actual available iodine release would be much less than the assumed value.

(3) Conservative continuous atmospheric conditions were used in accordance with RG 1.3. Not only are there diurnal changes in wind direction and velocity, but there are also hourly changes that would dramatically lower the exposure.

(4) The charcoal efficiency factor for iodine was assumed to be 95% compared to an actual measured efficiency in excess of 98%; therefore, the assumed removal is 2.5 times less efficient than the 98% removal.

(5) It was assumed that the exposed individual stands at the site boundary continuously for 24 hours and the individual would always be immersed in the plume. In all likelihood a person would only be in the plume for several minutes, which, by itself would reduce exposure to less than 10 CFR 20 guidelines.

(6) It was assumed that the reactor had been operated at 7.5 MW (150% of licensed power). The steady-state operating power limit is 5 MWt, a 50% decrease in power and fission product inventory.

On the basis of the above conservative analysis, the staff concludes that the MHA for the UCS reactor does not result in undue risk to the health and safety of the general public. The analysis demonstrated that even if an inordinately conservatively high fission product release was assumed, the radiation doses to a person located 250 m from the stack and continuously in the plume for 24 hours would be only 1.2 rem and 8.0 rem to the whole body and the thyroid, respectively. A more realistic evaluation would place these values well below the 10 CFR 20 and ICRP guidelines, respectively, for whole-body and thyroid doses.

14.7 Conclusions

The staff has reviewed various postulated and credible transients and accidents at the UCS reactor facility. On the basis of this review, the only events that are postulated to result in the release of fission products to the environment are the failure of an in-core fueled experiment capsule and a hypothetical loss-of-coolant accident with the assumed meltdown of an amount of fuel containing 10% of the total core accumulated fission products. The analysis, using conservative assumptions, has shown that if either of these events should occur, the resultant doses would be at or below the limits specified in 10 CFR 20 and ICRP. The staff also notes that because of the low water temperature and pressures in the primary system, the probability of an MHA is extremely unlikely. In addition, actual operating parameters would decrease the calculated exposures by a factor of 10-100. The staff concludes, therefore, that the design of the facility, together with the Technical Specifications, provide reasonable assurance that the UCS reactor can be operated at 5 MWt without significant risk to the health and safety of the general public or the UCS staff.

15 TECHNICAL SPECIFICATIONS

The licensee's Technical Specifications evaluated in this licensing action define certain features, characteristics, and conditions governing the operation of this facility. These Technical Specifications are explicitly included in the license renewal as Appendix A. Formats and contents acceptable to the NRC have been used in the development of these Technical Specifications, and the staff has reviewed them using the ANSI/ANS 15.1-1982 standard, "The Developement of Technical Specifications for Research Reactors," as a guide.

On the basis of its review, the staff concludes that normal plant operation within the limits of the Technical Specifications will not result in offsite radiation exposures in excess of 10 CFR 20 limits. Furthermore, the limiting conditions for operation and surveillance requirements will limit the likelihood of malfunctions and mitigate the consequences to the public of off-normal or accident events.

16 FINANCIAL QUALIFICATIONS

By letters dated January and March 1984, UCS transmitted its 1982 and 1983 Financial Reports. This supplemented the financial information provided in the renewal application of May 1980.

Union Carbide Subsidiary B, Inc., is part of the Medical Products Division of Union Carbide Corporation, which is a multi-billion dollar diversified corporation.

The staff reviewed the licensee's financial status and concludes that funds will be made available to support continued operations and, when necessary, to shut down the facility and maintain it in a safe shutdown condition. The licensee's financial status is in accordance with the requirements of 10 CFR 50.33(f)(ii). Therefore, the staff concludes that the UCS financial qualifications are acceptable.

17 OTHER LICENSE CONSIDERATIONS

17.1 Prior Reactor Utilization

Previous sections of this SER concluded that normal operation of the reactor causes insignificant risk of radiation exposure to the public and that only an off-normal or accident event could cause some significant exposure. The maximum hypothetical accident (MHA) was shown to result in radiation exposures within applicable guidelines and regulations (10 CFR 20, ICRP).

The staff has reviewed the impact of prior operation of the facility on the risk of radiation exposure to the public. The two parameters involved are the likelihood of an accident and the consequences if an accident occurred.

Although the staff has concluded that the reactor was initially designed and constructed with both inherent safety and additional engineered safety features, the staff considered whether continued operation would cause significant degradation in these features. Furthermore, because loss of integrity of fuel cladding is possible, the staff considered mechanisms that could increase the likelihood of failure. Possible mechanisms are (1) radiation degradation of cladding strength, (2) corrosion or erosion of the cladding leading to thinning or other weakening, (3) mechanical damage as a result of handling or experimental use, and (4) degradation of safety components or systems.

The staff's conclusions regarding these parameters, in the order in which they were identified above, are

(1) As all the fuel in the core is replaced with new fuel at intervals of about 2 years, radiation damage to the cladding is unlikely.

(2) The relatively short time that the fuel is in the core, coupled with the high purity of the coolant, make corrosion damage of the fuel cladding unlikely.

(3) Mechanical damage as a result of fuel handling is a possibility if fuel elements are dropped during fuel relocation manipulations. However, as fuel is moved about under water, incidents involving dropping of fuel have never resulted in any damage to the fuel. See Section 14.4 for additional potential fuel handling incidents.

Damage to the core resulting from experiments is very remote because all experiments are reviewed by the Nuclear Safeguards Committee.

(4) UCS performs regular preventive and corrective maintenance and replaces components as necessary. Nevertheless, there have been some malfunctions of equipment. However, the staff review indicates that most of these malfunctions have been random one-of-a-kind incidents, typical of even good quality electromechanical instrumentation. There is no indication of significant degradation of the instrumentation, and the staff further concludes that the UCS procedures, calibration, testing, and preventive

maintenance program would lead to adequate identification and replacement before significant degradation occurred. Therefore, the staff concludes that there is strong evidence that any future degradation will lead to prompt remedial action by UCS, and there is reasonable assurance that there will be no significant increase in the likelihood of occurrence of a reactor accident as a result of component malfunction.

17.2 Conclusion

On the basis of the above considerations, the staff concludes that there are no other credible events that could produce effects greater than those already analyzed in Section 14.

18 CONCLUSIONS

On the basis of its evaluation of the application as set forth above, the staff has determined that

(1) The application for renewal of Operating License R-81 for its reactor filed by UCS dated May 23, 1980, as supplemented, complies with the requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR, Chapter I.

(2) The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission.

(3) There is reasonable assurance (a) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public; and (b) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR, Chapter I.

(4) The licensee is technically and financially qualified to engage in the activities authorized by the license in accordance with the regulations of the Commission set forth in 10 CFR, Chapter I.

(5) The renewal of this license will not be inimical to the common defense and security nor to the health and safety of the public.

19 REFERENCES

Aggarwal, Y. P., and L. R. Sykes, "Earthquakes, Faults and Nuclear Power Plants in Southeastern New York and Northern New Jersey," Science 200 425-429, April 1978.

Burn, R. R., and S. H. Krapp, Research, Training Test and Production Reactor Directory, American Nuclear Society, La Grange Park, Ill., Vol. II, p. 1367, 1980.

Code of Federal Regulations, Title 10, "Energy" (including General Design Criteria), U.S. General Services Administration, Office of Federal Register, National Archives and Records Service, U.S. Government Printing Office, Washington, D.C., January 1981.

International Commission on Radiological Protection, "Report of ICRP Committee II on Permissible Dose for Internal Radiation," Health Physics, 3, 1960.

Kafka, A. K., "Earthquake Hazard Studies in New York State and Adjacent Areas," NUREG/CR-3079, 1983.

Mitronovas, W., "Earthquake Hazard in the State of New York," Empire State Geogram, Vol. 16, No. 2, New York Geological Survey, 1980.

Oak Ridge National Laboratory, Report ORNL-59-8-39, J. A. Cox, "ORR Operations for Period April 1958 - April 1959," 1959.

---, ORNL-2892, J. W. Wett, Jr., "Surface Temperatures of Irradiated ORR Fuel Elements Cooled in Stagnant Air," March 23, 1960.

---, ORNL-3026, W. R. Gambill and R. D. Bundy, "Burnout Heat Fluxes for Low Pressure Water in Natural Circulation," December 20, 1960.

---, ORNL-3079, W. R. Gambill and R. D. Bundy, "HFIR Heat Transfer Studies of Turbulent Water Flow in Thin Rectangular Channels," June 1961.

---, ORNL-TM-1207, E. B. Johnson and P. K. Reedy, Jr., "Critical Experiments With SPERT-D Fuel Elements," July 14, 1965.

---, ORNL-TM-2421, W. R. Gambill, "Design Curves for Burnout Heat Flux in Forced-Convection Subcooled Light Water Systems," November 14, 1968.

Phillips Petroleum Co., Report IDO-16528, "Analysis of Self-Shutdown Behavior in the SPERT-1 Reactor," July 1959.

---, IDO-17000, "Reactor Power Excursion Tests in the SPERT-IV Facility," August 1964.

Stover, G. W., L. M. Barnhard, B. G. Reagor, and S. T. Algermissen, "Seismicity Map of the State of New Jersey," U.S. Geological Survey, 1980.

---, Seismicity Map of the State of New York," U.S. Geological Survey, 1981.

U.S. Atomic Energy Commission, Report AECD-3668, J. R. Dietrich, "Experimental Investigation of the Self-Limitation of Power During Reactivity Transients in a Sub-Cooled Water-Moderated Reactor," BORAX-1 experiments, 1954.

---, Report IDO-17000, "Reactor Power Excursion Power Tests in the SPERT IV Facility," August 1964.

---, Report TID-24190, D. H. Slade, Ed., "Meteorology and Atomic Energy 1968," July 1968.

U.S. Nuclear Regulatory Commission, NUREG/CR-1386, "Fission Product Release From Highly Irradiated LWR Fuel Heated to 1300-1600°C in Steam," December 1980.

Woodward-Clyde Consultants, "Response to Comments made in the Sandia Draft Letter Report on Review and Evaluation of the Indian Point Probabilistic Safety Study," 1982.

<u>Industry Codes and Standards</u>

American National Standards Institute/American Nuclear Society (ANSI/ANS)the 15 series

---, ANSI/ANS 15.11, "Radiological Control at Research Reactor Facilities," 1977.

---, ANSI/ANS 15.16, "Standard for Emergency Planning for Research Reactor," Draft 1978 and Draft 2, November 1981.

American Nuclear Society (ANS) 15.1, "Standard for the Development of Technical Specifications for Research Reactors," September 1981.

---, ANS 15.4, "Selection and Training of Personnel for Research Reactors," 1977.

---, ANS 15.6, "Review of Experiments for Research Reactors," 1974.

NRC FORM 335 (11-81)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC)	NUREG-1059
4. TITLE AND SUBTITLE (Add Volume No., if appropriate)	Safety Evaluation Report related to the renewal of the operating license for the Union Carbide Subsidiary B,Inc. research reactor
2. (Leave blank)	
3. RECIPIENT'S ACCESSION NO.	
7. AUTHOR(S)	
5. DATE REPORT COMPLETED	MONTH: June / YEAR: 1984
DATE REPORT ISSUED	MONTH: June / YEAR: 1984
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)	Division of Licensing Office of Nuclear Reactor Regulation U.S. Nuclear Regulatory Commission Washington, DC 20555
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. FIN NO.	
13. TYPE OF REPORT	Safety Evaluation Report
PERIOD COVERED (Inclusive dates)	
15. SUPPLEMENTARY NOTES	Docket No. 50-54
14. (Leave blank)	

16. ABSTRACT (200 words or less)

This Safety Evaluation Report for the application filed by the Union Carbide Subsidiary B,Inc.(UNC) for a renewal of operating license R-81 To continue to operate a research reactor has been prepared by the Office of Nuclear Reactor Regu;ation of the U.S. Nuclear Regulatory Commission. The facility is owned and operated by the Union Carbide Subsidiary B,Inc. and is located in the City of Tuxedo, Orange County, New York. The staff concludes that the reactor facility can continue to be operated by UNC without endangering the health and safety of the public.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

Non Power Reactor
Union Carbide Susidiary,Inc
Pool-type Reactor
License Renewal

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT	Unlimited
19. SECURITY CLASS (This report)	Unclassified
20. SECURITY CLASS (This page)	Unclassified
21. NO. OF PAGES	
22. PRICE	$

NRC FORM 335 (11-81)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

FIRST CLASS MAIL
POSTAGE & FEES PAID
USNRC
WASH D C
PERMIT No G67

120555078877 1 1AN
US NRC
ADM-DIV OF TIDC
POLICY & PUB MGT BR-PDR NUREG
W-501
WASHINGTON DC 20555

UNION CARBIDE SUBSIDIARY R, INC. RESEARCH REACTOR