NUREG-0982

Safety Evaluation Report
related to the renewal of the operating license for the Research Reactor at the State University of New York at Buffalo

Docket No. 50-57

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

May 1983

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ABSTRACT

This Safety Evaluation Report for the application filed by the State University of New York at Buffalo for a renewal of Operating License R-77 to continue to operate a research reactor has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is owned by the State University of New York and is located on the campus in Buffalo, New York. Based on its technical review, the staff concludes that the reactor facility can continue to be operated by the university without endangering the health and safety of the public or endangering the environment.

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1 INTRODUCTION

The Nuclear Science and Technology Facility (NSTF) of the State University of New York at Buffalo (SUNYAB) submitted a timely application to the U.S. Nuclear Regulatory Commission (NRC) (staff) for renewal of the Class 104 Operating License (R-77) for its pulse training assembled reactor (PULSTAR). The application was by letter (with supporting documentation sent separately) dated June 14, 1979 for renewal for a period of 20 years, 3 months. The NSTF (licensee) currently is permitted to operate the reactor within the conditions authorized in past license amendments in accordance with Title 10 of the Code of Federal Regulations (10 CFR), Paragraph 2.109, until NRC action on the renewal request is completed.

The staff's technical review with respect to issuing a renewal of the operating license to the NSTF has been based on the information contained in the renewal application and supporting supplements, plus responses to requests for additional information. The renewal application includes: the Physical Security Plan as supplemented through March 11, 1983; Technical Specifications as supplemented through March 10, 1983; Environmental Impact Appraisal Data as supplemented through April 3, 1981; a Safety Analysis Report as supplemented through March 10, 1983; the Reactor Operator Requalification Program; Emergency Plan; and responses to additional questions sent by letters dated March 10, April 20, and May 6, 1983. This material is available for review at the Commission's Public Document Room at 1717 H Street N.W., Washington, D.C.

The renewal application contains the information regarding the original design of the reactor facility and includes information about modifications to the facility made since initial licensing. The Physical Security Plan is protected from public disclosure under 10 CFR 2.790(d)(1) 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 NSTF PULSTAR 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 NSTF reactor at steady-state thermal power levels up to and including 2 MW. The current license also authorizes pulsed operation, but the licensee has requested that that feature be removed from the renewed license. The facility was reviewed against the requirements of 10 CFR 20, 30, 50, 51, 55, 70, and 73, and applicable Regulatory Guides (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 specific accident-related regulations for research reactors, the staff has at times compared calculated hypothetical radiation dose values with related standards in 10 CFR 20, the standards for protection against radiation, both for employees and the public.

This SER was prepared by Robert E. Carter, Project Manager, Division of Licensing, Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission. Major contributors to the technical review include J. E. Hyder, D. B. Jensen, C. A.

Linder, and C. C. Thomas, Jr., of Los Alamos National Laboratory (LANL) under contract to the NRC.

The NSTF reactor has been in operation since June 1961. From 1961 to 1963 the reactor was fueled with materials-testing-reactor (MTR)-type fuel elements and operated at a maximum steady-state power level of 1 MW. In 1963 the reactor was shut down and the core and control systems were modified so that the reactor could operate with PULSTAR*-type fuel at power levels up to 2 MW. The license was amended in June 1963, authorizing the 2 MW operation. The original core grid plate was retained and the MTR fuel elements were replaced with PULSTAR pin-type fuel clusters designed to use the same grid spacing. On May 12, 1965, after extensive testing, the reactor was licensed to operate in the pulse mode also, with routine energy per pulse up to 35 MW·sec, and a maximum size pulse of 44 MW·sec.

The NSTF reactor has operated with PULSTAR fuel since 1964, with one major refueling in 1978. The annual use in the experimental and instructional programs has averaged approximately 6.8×10^3 MW-hours since 1970.

The original reactor facility and its control and safety systems were designed and built by American Machine and Foundry (AMF), who produced and installed approximately sixteen similar research reactors around the world during the 1950s and 1960s. Although several of these reactors have now been shut down, the total accumulated operational experience amounts to more than 340 reactor years. The fuel in the NSTF reactor is unique for research reactors, there being only one other similar one operating in the United States; however, this fuel design is based on extensive tests at the special power excursion reactor test (SPERT) facility (Spano, 1963) and on power reactor technology and experience.

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. In addition, as part of the licensing review, the staff reviewed analyses of several accidents postulated for the PULSTAR-type reactor.

The principal matters reviewed for the NSTF reactor and the conclusions reached were the following:

(1) The design, testing, and performance of the reactor structure and 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 serious credible accidents and determined that the calculated

*PULSTAR also is a trade name for the type of research reactor fuel used in the NSTF reactor, based on power nuclear reactor concepts and technology, developed jointly by NSTF and American Machine and Foundry.

potential radiation doses outside of the reactor room are not likely to exceed 10 CFR 20 doses in unrestricted areas.

(3) The licensee's management organization, conduct of training and research activities, 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 the Commission's regulations and are as low as reasonably achievable (ALARA).

(5) The licensee's Technical Specifications, which provide operating limits controlling 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 financial data and information provided by the licensee are such that the staff has determined that the licensee has sufficient revenues to cover operating costs and to ensure protection of the public from radiation exposures when operations are terminated.

(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 licensee submitted an Emergency Plan using NRC guidance that was current at the time of license renewal application. The licensee has submitted a revised Emergency Plan that follows new guidance developed since the NSTF renewal request was tendered. This item is discussed further in Section 13.3.

1.2 Reactor Description

The NSTF PULSTAR is a heterogeneous pool-type reactor, the core is cooled by forced convective cooling at higher power levels and by natural convective cooling at lower power levels. The coolant/moderator is light water, and the reflector may be water or graphite. The core is immersed in an 87,000-l (23,000-gal),* aluminum-lined reinforced concrete pool. The coolant is circulated through external systems for heat removal and for purification. Reactor experimental facilities include incore irradiation positions, a thermal column, beam tubes, pneumatic sample transport systems, a dry gamma chamber, and a gamma irradiation facility.

The current reactor core configuration has been operated since 1964, with one major refueling. The fuel design is similar to nuclear power reactor fuel, consisting of pellets of sintered uranium dioxide stacked in long thin-walled

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*In general, metric units are used in this SER with English equivalents given in parentheses.

metal tubes. The uranium is enriched to 6% in the ^{235}U isotope, and the reactor exhibits a large negative temperature coefficient of reactivity including a Doppler effect of broadening of ^{238}U absorption resonances.

1.3 Reactor Location

The NSTF reactor building is located near the southwestern edge of the Main Street Campus of the State University of New York at Buffalo (SUNYAB), which is located in the northeast corner of the city of Buffalo, New York.

1.4 Shared Facilities and Equipment and any Special Location Features

The reactor building is constructed of reinforced concrete and is attached to a laboratory complex dedicated primarily to nuclear science-related research and instruction. Some of the spaces adjacent to the reactor building are associated with reactor operations and use. Utilities such as municipal water and sewage, natural gas, and electricity are provided to the complex for joint use.

The reactor building is air conditioned with the air exhausted through absolute filters and discharged from a 36-in. duct at roof level. Air from certain experimental facilities and the hot chemistry laboratory is exhausted separately and discharged from a stack ~50 m above grade, which also functions as the smokestack of a campus power plant. The nearest public residence is approximately 122 m (400 ft) from the reactor building, and approximately 70 m (230 ft) from the exhaust stack.

1.5 Comparison With Similar Facilities

The reactor core is similar to that of one other licensed research reactor and is based on nuclear power reactor fuel technology. The instruments and controls are typical both of NRC-licensed and other operating research reactors.

2 SITE CHARACTERISTICS

2.1 Geography

The general location of the Main Street Campus of the State University of New York at Buffalo (SUNYAB) is in the northeast corner of the city of Buffalo, New York, which lies at the eastern end of Lake Erie along the Niagara River. The Township of Tonawanda is northwest of the site, Amherst is northeast, and Cheektowaga is east and southwest.

Buffalo is located on the eastern end of Lake Erie and along the Niagara River on a gently sloping plane. The country surrounding the campus is low and level to the west with gently rolling hills to the east and south. There are pronounced hills within 19 to 30 km (11 to 18 mi) to the south-southeast, which rise to a height of 300 m (984 ft) above the level of Lake Erie at a distance of approximately 56 km (34 mi) from Buffalo.

The actual reactor site is near the southwestern edge of the campus, which is in a triangle bounded by three principal streets. The nearest off-campus public residence is approximately 122 m (400 ft) from the reactor building.

2.2 Demography

According to the 1980 census, the population of the city of Buffalo is 358,000 and the population of the Buffalo metropolitan area is 1.24 million. Both of these figures are smaller than the analogous data compiled in 1960 when the reactor was built. Figure 2.1 shows the location of the campus in relation to this area.

The student body of the Main Street Campus is composed mostly of commuting students, with an average day-time population of approximately 10,000. Most of the student dormitories of SUNYAB are on the Amherst Campus several kilometers away. There is a large Veteran's Administration Hospital some 640 m (2,000 ft) from the reactor, just off the campus to the northeast. Generally, the city of Buffalo, which has a high-density population, occupies the quadrant to the south and southwest, starting at the corresponding border of the campus.

2.3 Nearby Industrial, Transportation, and Military Facilities

2.3.1 Transportation Routes

The Greater Buffalo International Airport is approximately 10 km (6 mi) east of the SUNYAB campus, but only a minority of the larger commercial flights have air-routes over the campus. There are several streets bordering or close to the campus that carry high-density commuter traffic, but there are no major interstate highways within 5 km (3 mi). There are no major military facilities involving transport flights or heavy vehicular traffic any closer than the interstate highway.

2.3.2 Nearby Facilities

The campus is located in an area containing no major industries, but instead has nearby suburban shopping centers, parks, hospitals, schools, etc.

2.3.3 Conclusion

There is no heavy industry or heavy air or ground traffic to constitute an external threat to the integrity of the reactor facility. Therefore, the staff concludes that there is no significant risk from accidents to the reactor because of activities related to the military, industry, or heavy transportation traffic.

2.4 Meteorology

Buffalo is located near the mean position of the polar front. Its weather is varied and changeable, characteristic of its latitude. Wide seasonal swings of temperature from hot to cold are tempered appreciably by the proximity of lakes. Lake Erie lies to the southwest, the direction of the prevailing wind. Wind flow is somewhat high throughout the year as a result of this exposure. The vigorous interplay of warm and cold air masses during winter and early summer causes one or more wind storms. Precipitation is moderate and fairly evenly distributed throughout the twelve months.

The data collected for the meteorological regime were obtained from various prepared annual meteorological summaries and from unprocessed data at the Weather Bureau itself. Much information was gathered at the former Buffalo Airport at el 213 m (698 ft), approximately 4 km (2.5 mi) from the campus, at the same elevation, with no intervening geographic features that might influence the data. Within a 4-km radius of the campus there is no point lower than 177 m (580 ft), nor higher than 220 m (720 ft). Consequently, the data can be assumed valid for the campus and immediate vicinity.

The wind regime in Buffalo is one of moderate velocity predominantly from the southwest. The wind is most frequently south and southeast when precipitation occurs. Within the first hundred meters or so of the surface the wind at times varies as much as 90 to 180° in direction from the actual surface wind. In general, however, the winds of the lower layers of air above the surface are within 45° of the direction of the surface wind. Though the wind circulation is basically southwest, local geographical features change this pattern somewhat. The local land and sea breezes previously mentioned are of considerable influence. Because the prevailing wind for Buffalo is southwest, the trajectory is over Lake Erie. This path of wind over the water is less affected by friction than if a land route were followed. Furthermore, with the lower relief of the lake, the wind is funneled or held in the lake path enroute, and the velocity is thereby built up. Therefore, higher wind velocities are maintained by the time the wind reaches Buffalo. In fact, velocities are some 20 to 30% higher in Buffalo than only a few kilometers either north or south of the city, and stagnant air conditions exist less than 1% of the time.

The wind data of Table 2.1 indicates the wind variance. Through the year the wind blows from the southwest quadrant 61% of the time, including west and south winds. (The city of Buffalo lies southwest of the reactor facility.)

During the winter and to a less extent the spring, there is a high degree of cloudiness and a low percentage of sunshine. Numerous storms pass over the area and the polar front swings over the area periodically during this time. During the summer and in the early fall a large percentage of clear and partly cloudy weather prevails in the Buffalo area. The lake tends to moderate conditions and prevent the formation of many thunderstorms that are found in the interior in mid-afternoon during the warmer weather.

Table 2.2 summarizes the precipitation data in the vicinity of the airport, and Table 2.3 gives the data for sky cover.

Based on the information presented in the licensee's SAR, as summarized above, the staff concludes that the meteorological conditions of the NSTF reactor site are acceptable for the relatively rapid dispersal of airborne radioactivity released from the reactor.

2.5 Geology

The SUNYAB campus is covered with dense glacial clay overburden generally 3-to-7-m (10-to-20-ft) thick. Such boulder clay is general over the entire Buffalo Area. To the north, the bed of glacial Lake Tonawanda is surfaced by varved clays, silty sands with some peat and marl.

The bedrock of the area consists of Silurian and Middle Devonian marine shales, dolomites, and limestones. The structure is very simple, there being a general southerly drop of approximately 24 m per kilometer (50 ft per mile) (1/2%). The rock formation is a joint system consisting of two joint sets intersecting at about 90° and, in the campus area, these two sets are arranged north-south, east-west. Where the overburden will allow entrance, water moves readily along the joints presumably reaching the surface along the scarp face of the cuesta.

The uppermost formations and members are shown on the campus profile (Figure 2.1). These rocks are underlain by a similar marine series of Ordovician and Cambrian Age to a depth of some 1,000 m (3,000 ft) where they rest on Precambrian.

On the campus the overburden lies on a well-developed and markedly level glacial pavement developed on the strong, dense Onondaga limestone. The transition from the overburden to the Onondaga is not at all gradational but extremely sharp. Figure 2.2 represents data from core drillings on the campus.

The thickness of the overburden from the bedrock to the surface averages about 4.6 m. The soil proper occupies about 0.3 m while the subsoil is made up of dense boulder clay that extends uniformly down to the bedrock.

Under average moisture conditions, the soil to a depth of 0.2 m consists of moderately compacted, brownish-gray, heavy, silty clay loam; beneath that is a 0.1 m stratum of compacted, dull yellowish-brown, silty clay containing a considerable amount of fine pebbles. This layer is underlain by dull brown plastic, very heavily compacted, silty clay that contains some crystalline pebbles throughout. This boulder clay material extends down to the bedrock without any significant changes.

Water movement within the overburden, including the soil proper, is extremely slow. Only within the first 0.3 m does water percolation take place at a very slow rate, nowhere exceeding 0.5 cm per hour. In the subsoil and weathered material to a depth of 4.6 m (15 ft) from the surface, permeability ceases for all practical purposes. Standing surface water, a good indication of extremely slow internal drainage, is typical for the area. Its removal is accomplished by either artificial surface drainage or by evaporation.

On the basis of the information above, there are no known geologic formations at the reactor site that could lead to such hazards as cavernous conditions, tectonic depressions, surface subsidence or uplifts, or volcanoes. Also there are no conditions present that could produce rockfalls, avalanches, or floods. Therefore, the staff concludes that geologic formations at the NSTF site do not pose a significant risk to the reactor facility as to make the site an unacceptable location for the NSTF reactor.

2.6 Hydrology

The campus is located on the extreme edge of the Onondaga Cuesta which follows a northeast to southwest course in the immediate vicinity. Here the cuesta faces northwest and stands above the land in that direction by some 24 m (80 ft). This change in elevation forms a steep slope to the northwest of approximately 21 m (70 ft) over 0.8 km (0.5 mi) (2.7% grade). The southeastern part of the campus is located on the dip slope of the cuesta. As modified by glacial deposits, this, in the vicinity of the reactor site, amounts to approximately 0.3 m (10 ft) over 0.8 km in a southerly direction.

The general area is extremely flat except for the Onondaga Cuesta. As a consequence streams are slow flowing and widely spaced. Ellicott Creek flows north over the cuesta at Williamsville some 6.4 km east of the campus. A small unnamed tributary to Ellicott Creek has its headwaters approximately 3 km north of the campus. No other streams are as close.

On the NSTF reactor site, surface water will flow in a southerly direction, to be intercepted by an adequate storm sewer system. Normally it would flow to the sewage treatment plant. A portion of the storm sewage at peak bypasses into a large retaining pool and is pumped back into the system during normal flow periods.

The staff concludes that the hydrologic characteristics of the site do not make it unacceptable for the location of the reactor facility.

2.7 Seismology

Four earthquakes since 1857 have had their epicenters within 48 km of Buffalo. Table 2.4 lists available data concerning those incidents. No faults are known in this area nor in the area of the epicenters. The shocks perhaps were related to the general isostatic readjustment following the Wisconsin glaciers.

The first three earthquakes referred to in Table 2.4 were quite minor shocks. The 1929 quake intensity in Buffalo (acceleration 1-50 $cm \cdot sec^{-2}$) was appreciable if the upper part of the intensity given by the Coast and Geodetic Survey was reached.

With the favorable bedrock conditions on the NSTF site, a properly constructed building resting on the Onondaga Cuesta would, in the opinion of the applicant's consultant, not be adversely affected by any but the most severe quake.

Based on the geology of the SUNYAB area and the past seismic activity, the probability of the occurrence of significant earthquakes in the future can be reasonably expected to be very small. The staff concludes that seismic hazards associated with the reactor facility are, therefore, very small and pose no unacceptable risk to the reactor.

2.8 Conclusions

The staff has reviewed and evaluated the NSTF reactor site for both natural and man-made hazards and has concluded that there are no significant risks associated with the site that make it unacceptable for the continued operation of the research reactor.

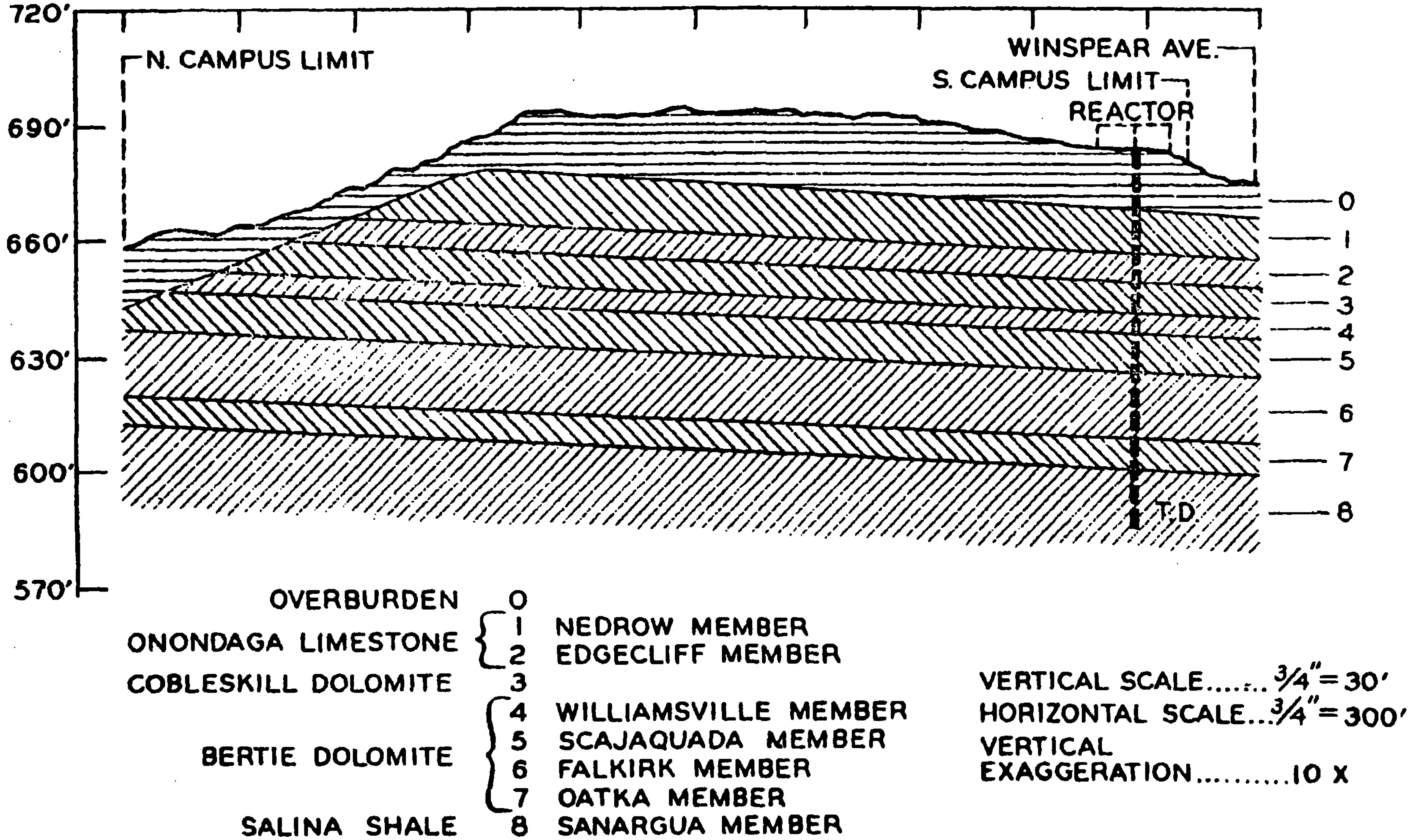

Figure 2.1 Profile through nuclear reactor site

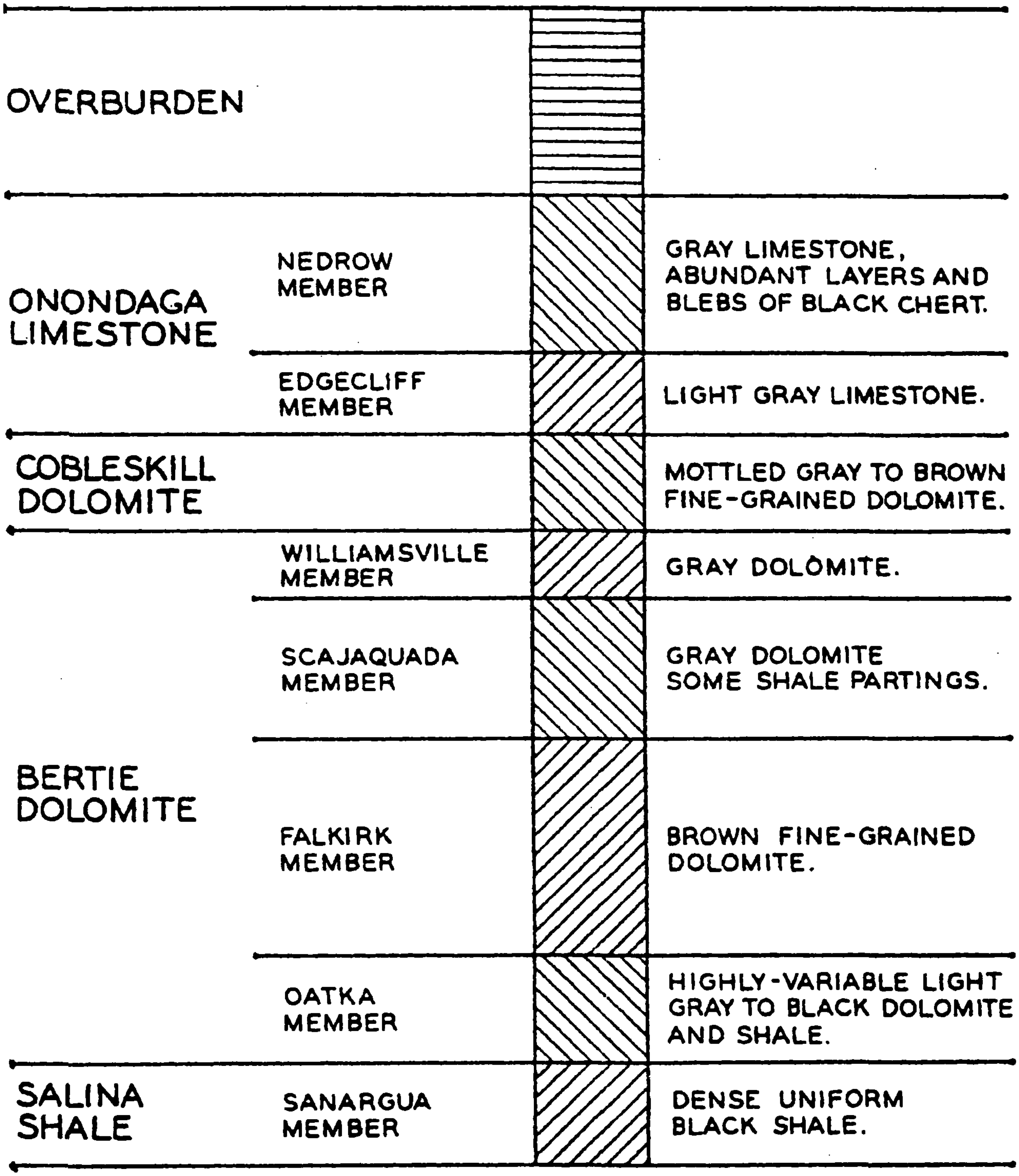

Figure 2.2 Stratigraphic column at nuclear reactor site

Table 2.1 Monthly winds, 1969 Through 1978

Month	Normal mean mph *	Average mph	Maximum mph	Prevailing direction
J	14.6	14.4	60	W
F	14.1	13.1	47	WSW
M	13.7	12.7	56	WSW
A	13.0	12.3	51	WSW
M	11.8	10.8	47	SW
J	11.2	10.4	37	SW
J	10.6	10.2	39	SW
A	10.0	9.0	41	SW
S	10.6	9.8	41	SW
O	11.4	10.5	38	SW
N	12.9	11.7	47	SWW
D	13.5	12.4	49	WSW

*1941 - 1979 period
**≥1 minute duration

Table 2.2 Precipitation, 1970 through 1978, inches water equivalent

Month	Normal*	Maximum monthly	Minimum monthly	Maximum in 24 hours	Inches snow maximum monthly
J	2.90	6.47	1.03	2.40	68.3
F	2.55	5.80	0.81	2.31	54.2
M	2.85	5.59	1.20	2.14	29.2
A	3.15	5.90	1.27	1.17	15.0
M	2.97	6.39	1.21	2.03	2.0
J	2.23	6.06	0.11	3.04	0.0
J	2.93	6.43	0.99	3.38	0.0
A	3.53	10.67	1.10	3.88	0.0
S	3.25	8.99	0.77	3.63	(Trace)
O	3.01	9.13	0.30	3.49	3.1
N	3.74	6.37	1.44	2.51	31.3
D	3.00	8.02	0.69	2.16	60.7

*1941 through 1970

Table 2.3 Sky cover, 1943-1978, mean number of days

Month	Clear	Partly Cloudy	Cloudy
J	1	7	23
F	2	5	21
M	4	7	20
A	5	8	17
M	6	9	16
J	6	12	12
J	7	13	11
A	7	12	12
S	7	9	14
O	7	8	16
N	2	5	23
D	1	6	24

Table 2.4 Earthquakes affecting Buffalo

Characteristics	1857 Oct. 23	1873 July 6	1879 Aug. 21	1929
Location	43.2 N 78.6 W	43.0 N 79.5 W	43.2 N 79.2 W	42.9 N 78.3 W
Area affected	47,000 km^2	78,000 km^2	3,400 km^2	259,000 km^2
Intensity at epicenter modified Mercali (MM)	VI	V-VI	V-VI	VIII-VIII+
Buffalo distance	48 km	56 km	48 km	48 km
Estimated intensity at Buffalo (MM)	II	I-II	I-II	IV
Estimated acceleration $cm \cdot sec^{-2}$ at Buffalo	1-5	1-5	1-5	1-50

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Wind Damage

Meteorological data indicate a low frequency of tornadoes and effects of tropical disturbances at SUNYAB. Further, the reactor building is a poured reinforced concrete cylinder with walls approximately 0.60 m thick, and the reactor pool is a massive reinforced concrete structure located mostly below grade. Therefore, the staff has concluded that wind or other storm damage to the NSTF reactor is very unlikely.

3.2 Water Damage

The reactor building is situated on a gently sloping terrain well above the floodplain. Therefore, the staff has concluded that there is reasonable assurance that damage to the reactor by flood is small.

3.3 Seismic-Induced Reactor Damage

The data on past seismic activity and future likelihood of earthquakes in the Buffalo area, summarized in Section 2.7, indicate that SUNYAB is located in a region of low probability of seismic activity. In the event of an earthquake and catastrophic damage to the reactor building and/or the reactor pool, water might be released. However, Section 14 shows that loss of coolant in itself does not lead to core damage. These considerations lead the staff to the conclusion that the risk of radiological hazard resulting from seismic damage to the reactor facility is small.

3.4 Mechanical Systems and Components

The mechanical systems of importance to safety are the neutron-absorbing control blades suspended from the superstructure. The motors, gear boxes, electromagnets, switches, and wiring are above the level of the water and readily accessible for testing and maintenance. An extensive preventive maintenance program has been in operation for many years for the NSTF to conform and comply with the performance requirements of the Technical Specifications.

The effectiveness of this preventive maintenance program is attested to by the small number and types of malfunctions of equipment over the years of operation for the NSTF reactor. These malfunctions have generally been one of a kind (that is, few repeats) and/or of components that were fail-safe or self-annunciating. Therefore, the staff concludes that there appears to be no significant deterioration of equipment with time or with operation. Thus, there is reasonable assurance that continued operation of the NSTF reactor for the requested period of renewal will not increase the risks to the public.

3.5 Conclusion

The NSTF reactor was designed and built to withstand all credible wind and water damage contingencies associated with the SUNYAB site. A seismic event has a small likelihood of occurring and the consequences of such an event would not be great; therefore, the staff has concluded that a seismic event need not be evaluated explicitly.

4 REACTOR

The NSTF PULSTAR is a fixed-core, pool-type research reactor using light water as the moderator, coolant, and partial shield, and using solid pin-type fuel assemblies. The reactor is currently authorized to operate both in the steady-state mode at power levels up to and including 2 MW and in the pulsing mode with maximum energy generation per pulse up to 44 MW·sec.

The reactor core is immersed in an aluminum-lined, water-filled, open-topped pool. The pool is spanned by a fixed structure that supports the control rod systems, reactor instrumentation, and some experimental facilities. The core itself is located near the bottom of the pool and is supported on a plenum structure that rests on the pool floor.

Reactor control is achieved by inserting or withdrawing neutron-absorbing control elements suspended from the drive mechanisms. Heat generated by fission is transferred from the fuel to pool water. At higher power levels, the water is forced downward through the fuel to an external heat exchanger. At lower power levels, cooling is provided by natural convection of the water within the pool. The following discussions are based on information obtained from licensee reports and during visits at the licensee's facility.

4.1 Reactor Building

The NSTF building housing the reactor is a poured concrete right cylinder, 21.3 m in diameter and 15.8 m high. The reinforced concrete walls and roof (supported by concrete beams) are 0.61 m thick and 0.1 m thick, respectively. The building walls and the first level are laid on bedrock.

This gas tight cylinder encloses the reactor pool and all necessary auxiliary facilities, including the control room and storage spaces for radioactive materials. The reactor building is penetrated by two sets of personnel airlocks and a truck entrance hatch, ventilation ducts, electrical conduit, and piping. The truck hatch is sealed with an inflatable gasket when the reactor is in operation. The ventilation ducts are equipped with hydraulic dampers, and all electrical conduits penetrations are sealed with epoxy resin.

The NSTF reactor building is air conditioned with about four air changes per hour, exhausted through absolute filters from a 36-in. duct at the building roof level. The reactor building is a part of the structure that houses the entire nuclear research center. Figure 4.1 is a vertical cross section of the building, and Figure 4.2 is a horizontal cross section at the level of the reactor core (neutron deck). Figures 4.3 and 4.4 show cross sectional views of the reactor itself.

4.2 Reactor Core

The reactor core is composed of low-enriched uranium dioxide (UO_2) fuel assemblies inserted in the grid plate together with control blades and control blade guides, graphite-reflector elements (if used), sample irradiation stringers, and incore experiments. The fuel assembly end fittings are similar to those of the MTR-type fuel elements so that the original grid plate could be used when conversion to the PULSTAR design was made in 1964.

The reactor is approximately critical with 17 fresh fuel assemblies. Typical core loadings contain 22 to 27 fuel assemblies depending on burnup and experimental needs. The assemblies may be arranged in a variety of lattice patterns depending on experimental requirements. Special handling tools are used for underwater insertion or removal of any of the assemblies from the grid plate.

4.2.1 Fuel Assemblies

The NSTF reactor uses a type of fuel commonly referred to as PULSTAR. It consists of sintered UO_2 pellets in a pin geometry, similar to current light-water power reactor fuels.

The fuel assembly is shown in Figure 4.5. Each assembly consists of 25 fuel-bearing pins. The pins are thin-walled (0.51 mm) Zircaloy-2 tubes filled with sintered UO_2 pellets with welded Zircaloy-2 end plugs. The uranium is enriched to 6% in the isotope ^{235}U. The UO_2 pellets are about 1.1 cm in diameter and about 1.5 cm long. A finished pin is 1.184 cm in diameter and 66.0 cm long with spacers brazed around the circumference (90° apart) of the pin near the ends and at the center.

Approximately 30.7 g ^{235}U (513 g U) is contained in each pin or 768g ^{235}U (12.8 kg U) per assembly. The pins are fastened mechanically in groups of 25 with aluminum end fittings and are constrained in a Zircaloy-2 box. A guide tube (nosepiece) machined to fit the grid plate is attached to the lower end of the fuel pin assembly. A dowel is welded between two sides of the box near the upper end of the fuel assembly and serves as a handle for insertion or removal of the assembly from the grid plate. The finished assembly is about 96.5 cm long with a cross section of about 6.96 cm by 8 cm. The nosepiece is inserted in a large hole in the grid plate that supports the entire fuel array. A small pin set in the grid plate mates with a hole in the nosepiece shoulder to position the assembly axially. Both ends of the assembly are open so that cooling water can flow up or down around the fuel pins.

4.2.2 Control Elements

The reactor control system is typical of those used for pool-type research reactors with the exception that the control blades operate between fuel assemblies rather than within a control rod fuel element. The reactor is controlled by five thermal neutron-absorbing control-safety blades and one control blade, which is discussed in more detail in a later paragraph. The six blades are made of a nickel-plated silver/indium/cadmium alloy. One of the control-safety blades may be used for automatic servocontrol of reactor power, thus serving the function of a regulating rod. The control-safety blades provide coarse adjustment of the neutron flux density, and the control (regulating) blade provides fine adjustment. As discussed below, the licensee has proposed that the sixth blade (transient rod) used previously for pulse-mode operation be converted to a nonscramming control blade.

Drive mechanisms are actuated from the control console for remote positioning of the control elements. The drives could be relocated to cover any of the spaces in the grid plate.

The five control-safety blades have extension rods reaching above the surface of the pool water that terminate in armatures. These mate with electromagnets on the bottom of the drive mechanism, providing a scram capability. When the electromagnet current is interrupted, the armature is released and gravity causes the control blades to fall in their slots in the core. The elapsed time from scram initiation to full insertion of the blades is approximately 0.65 sec. Means are provided for automatic or manual scrams, blade reversal, and blade inhibits, to maintain the reactor in a safe operating range or for safe shutdown.

For usual core loadings of 22 to 27 fuel assemblies, the typical reactivity worth of the set of six blades is approximately 12.6% $\Delta k/k$, with the worth of the nonscramming blade being about 1.8% $\Delta k/k$. The maximum worth of a single blade is about 3 to 3.4% $\Delta k/k$. All of these values will vary with the nuclear characteristics for any specific core loading.

4.2.3 Conclusion

The staff has reviewed the information pertaining to the design and construction of the NSTF PULSTAR fuel, control-safety blades, control blade, and all control element drives, and has concluded that the design and operation of these core-related components are adequate to ensure safe operation of the reactor.

4.3 Reactor Pool

The reactor core is located near the bottom of a water-filled, aluminum-lined, reinforced-concrete pool that is roughly hexagonal in cross section, with a width of 4.27 m at the top. Approximately 4.57 m from the top, the pool interior is stepped to a width of 2.44 m to provide increased thickness of wall shielding in the immediate vicinity of the core. The aluminum liner is 6.35 mm thick, and a sealant is used to prevent corrosion of the liner that might otherwise occur because of contact with the concrete shield walls. The liner is penetrated as follows:

(1) Five 15.24-cm round beam-tube ports radiate from the core around the lower tank section.

(2) One pneumatic conveyor system enters near the top of the upper tank and terminates in the reflector region above the beam tubes.

(3) The primary coolant exits the tank through a penetration that formerly housed a 30.48-cm square beam port.

(4) The primary coolant returns to the pool through a penetration that formerly housed a 15.24-cm round beam port.

(5) A passthrough canal (tube) provides a passage between the upper portion of the tank and the hot cell.

(6) Eight emergency pool fill nozzles are located in the lower tank section, just below the step.

Storage cylinders for fuel assemblies are arranged around the upper section of the lower tank on all faces except the back wall where a dry chamber nosepiece is located. In addition, a rack for eight used elements is located on the tank wall common with the hot cell to provide an experimental gamma irradiation facility.

4.4 Core Support Structure

The grid plate is a 12.7-cm-thick aluminum plate bolted to a plenum chamber. Thirty-six holes capable of accommodating the nosepieces of the fuel assemblies are arranged in a 6-by-6 pattern on the plate. Small holes are located between the nosepiece holes to provide additional passages for water flow past the sides of the fuel assemblies. Holes not required for fuel assemblies or incore experiments are plugged to confine coolant flow to core assemblies and experiment positions. Small pins set in the grid plate mate with holes in the nosepiece shoulders to position the assemblies axially.

The plenum chamber, which channels the coolant flow to the discharge pipe during forced convection cooling, is supported by four legs. These legs are welded to the plenum and to the floor of the tank liner. The aluminum superstructure above the core provides a guide rack for the neutron detection chambers.

4.5 Reactor Instrumentation

The reactor instrumentation is similar to that found at research reactor installations at other laboratories. The initial control console and associated instruments were typical of those for approximately 17 research reactors supplied by the same vendor. During the past several years, instruments have been improved or replaced to maintain a state-of-the-art facility.

The nuclear instrumentation provides the operator with the necessary information for proper manipulation of the controls. The following instrument channels are provided and are discussed in more detail in Sections 4.8 and 7.

(1) count-rate or startup channel (fission chamber)
(2) linear power level and automatic control channel
(3) log-N power and period channel
(4) two safety channels
(5) ^{16}N power monitor
(6) core differential temperature
(7) cooling system temperatures

4.6 Biological Shield

The reactor core is shielded in the lateral directions by pool water and the concrete walls of the pool. Vertical shielding is provided by about 7 m of water above the core, and 1 m between the core and the pool floor. The concrete walls vary in thickness from top to bottom, being approximately 0.46 m at the top and stepping to 1.8 m in the lower section. For more effective shielding, the lower section was poured with an ilmenite aggregate, while the upper section was made of ordinary concrete.

The staff concludes that the shielding was designed adequately to reduce external radiation exposure rates to acceptable levels.

4.7 Dynamic Design Evaluation

To ensure safe and responsive operation, the reactor is provided with multiple control elements and nuclear instrumentation. The PULSTAR's inherent negative bulk temperature and Doppler temperature coefficients provide reactivity control during steady-state operation and provide a self-limiting mechanism on transients initiated by rapid additions of excess reactivity (Spano, 1963). As discussed by the licensee, most of the negative temperature coefficient of reactivity is caused by the Doppler broadening of ^{238}U neutron absorption resonances in the low enrichment (6% ^{235}U) fuel of the PULSTAR.

4.7.1 Core Thermal and Hydraulic Characteristics

A thermal-hydraulic analysis of the NSTF PULSTAR core was performed in 1963 (Western New York Nuclear Research Center, Inc. (WNY) SAR 1963). The licensee's pulse test program carried out since 1963, as well as extensive steady-state operation, provided considerable information on the thermal-hydraulic behavior of the core. In particular, the licensee has demonstrated that boiling in the coolant outlet channel does not cause significant risk of damage to fuel or cladding (WNY-017, 1964; WNY PULSTAR Summary Report, 1966). The licensee's proposed safety limits and the limiting safety system settings for forced convection and natural convection cooling are discussed in the following sections and are based on a recent analysis (NSTF Safety Analysis Report (SAR), 1981).

4.7.1.1 Forced Convection Cooling

The limiting criterion for safety is the assurance of integrity of the fuel and the cladding. The licensee, for purposes of his analysis, has assumed fuel or cladding integrity to be compromised if either fuel centerline melting or departure from nucleate boiling (DNB) in the coolant were to occur.

The licensee's analysis includes the assumptions that (1) the depth of the water above the core was 5.18 m, (2) the primary coolant flow rate was 63 l per second, and (3) the coolant inlet (pool) temperature was 60°C. The maximum reactor power levels corresponding to calculated critical heat flux (DNB) for core loadings ranging from 16 to 35 assemblies were calculated using the assumed process variables and the heat flux hot-spot factor (NSTF SAR, 1981). The correlations established in the WNY SAR (Rev. II, 1963) were used to calculate the fuel centerline, fuel surface, and inner and outer cladding temperatures corresponding to the maximum reactor power shown in Table 4.1. The melting points of UO_2 and Zircaloy-2 are approximately 2760°C and 1815°C, respectively. The 25-assembly core loading is a typical operating core. Initial steady-state criticality with a clean cold Xenon-free core required 17 assemblies.

The staff has reviewed the licensee's analysis (WNY SAR, 1963; NSTF SAR, 1981; Bernath, 1955) and has determined that the methods used are appropriate for application to the NSTF reactor and are very conservative. The staff, therefore, concludes that reactor operation, within the conditions established by the licensee's specified safety limits for operation with forced convection cooling given in Table 4.2, NSTF Technical Specifications Appendix A, and NSTF SAR update (1981) give reasonable assurance that fuel and cladding integrity will not be lost during normal operation with forced cooling at licensed power levels.

4.7.1.2 Natural Convection Cooling

The licensee conducted extensive tests in 1966 (WNY Technical Note J-435, 1966) on natural convective cooling in the NSTF reactor. These tests demonstrated that PULSTAR fuel can be operated in the natural-convection-cooling mode at power levels in excess of 1 MW without exceeding the critical heat fluxes (DNB), and thus exceeding the criterion for safety established for forced convection cooling. In order to ensure that the results of the referenced tests remain applicable, the height of water in the pool above the core must be no less than 6.1 m, the same as for forced convective cooling. The licensee has conservatively established 0.5 MW as the maximum operating power using natural convective cooling.

The staff concurs with the licensee's analysis and evaluation, and has concluded that operation of the PULSTAR with natural convective cooling at power levels up to and including 0.5 MW poses no significant risk of fuel or cladding damage resulting from high temperatures.

4.7.2 Pulse-Mode Operation

During 1964-1965, the licensee carried out a program to determine the characteristics, limitations, and safety of the PULSTAR low-enrichment (6% ^{235}U) UO_2 core under transient operation. This effort proved that the reactor meets most of the original design objectives and showed that the PULSTAR fuel can survive relatively large power transients; therefore, it is an inherently safe reactor. The NSTF reactor has been licensed to operate in the pulse (power transient) mode with a routine energy release per pulse of 35 MW·sec, corresponding to a reactivity insertion of 1.5% $\Delta k/k$, and a maximum energy release of 44 MW·sec. The NSTF reactor was the prototype for the North Carolina State University PULSTAR reactor.

Because the pulse mode of operation has never been used extensively with the NSTF reactor and is not required for the current experimental program, the licensee has eliminated pulse mode operation from the proposed revised Technical Specifications. The licensee cites the following disadvantages to maintaining pulse capability and authorization in the license:

(1) Maintenance of unused instrumentation and mechanical systems is a disadvantage.

(2) Maintenance of operator proficiency is a problem because pulsing is cumbersome to carry out and significantly interferes with the normal operation schedule. It should be noted that the more recently licensed operators have not been tested on pulse-mode operation and are qualified and licensed for steady-state operation only.

(3) Potential damage to the fuel is another consideration. Pulsing produces transient thermal stress in the cladding and fractures the UO_2 pellets.

In view of the high cost of fuel and the licensee's commitments to three-shift steady-state operation, the added risk to the fuel is not warranted.

The licensee has proposed that the transient rod be converted to a nonscramming control blade in conjunction with the elimination of pulse-mode operation. This would require plugging the inlet ports of the transient operation pneumatic cylinder, thus rendering it inoperative, and rigidly coupling the blade extension to the rod drive. The licensee has considered several alternatives. The first would convert the transient blade to a control-safety blade identical to the others. This would be prohibitively expensive because currently there is no commercial source of the required system components; therefore, this alternative was rejected. The second alternative was to continue steady-state operation with the transient rod always fully withdrawn, as has been the practice since 1964. This is undesirable because with the blade fully withdrawn the neutron flux tilts towards the corner of the core, where flux peaking generally is not needed for the experimental program. Further, the licensee indicates that it would be desirable to have two-axis symmetry in the power distribution, which requires an even number of control blades. Use of the transient blade as a nonscramming control blade would provide an even number of blades. The third alternative is embodied in the proposed Technical Specifications, by which the nonscramming control rod (1) is limited to flux distribution control, (2) is not considered in computations of shutdown margin, (3) may be left fully or partially withdrawn during operation, and (4) could, with prior notification to the NRC, be converted to a scram-safety blade at a future time as long as its resulting performance characteristics are similar to the other five control-safety blades.

The staff concurs with the licensee's evaluation of the elimination of pulsing from the Technical Specifications and from the license. The staff has determined that all available experience has proven that the NSTF reactor can be operated safely in both the steady-state and pulse modes as currently authorized, and that elimination of the pulsing feature by the proposed method does not constitute an unreviewed hazard and would not endanger the environment or the health and safety of the public. Further, the reason for deletion of the pulsing capability is for current programmatic purposes of this licensee, and this decision has no impact on the retention or elimination of pulsing authorization at other licensed nonpower reactors. The staff has also concluded that the licensee's proposed modification of the pulse-mode electro-mechanical systems is acceptable, and does not introduce an unreviewed safety question.

4.7.3 Shutdown Margin

The proposed Technical Specifications prescribe a minimum reactivity shutdown margin of 0.5% $\Delta k/k$ in a cold, xenon-free core with the highest worth control blade fully withdrawn. Depending on the actual core loading, the reactivity worth of this maximum control blade ranges from approximately 3% to 3.4% $\Delta k/k$, and the total worth of all control-safety blades, excluding the nonscrammable control blade, is between 9% and 12% $\Delta k/k$, with a typical value of 10.8% $\Delta k/k$ (NSTF Justification, 1981). Generally, the core loading producing the higher total worth of all blades also will correspond to the higher worth of the most reactive control blade. Therefore, as long as the total excess reactivity loaded into the core, including that resulting from experiments, is no more than 5.1% $\Delta k/k$, the shutdown margin can certainly be achieved. Furthermore, because higher total worth of all blades is closely related to higher worth of individual blades, the required shutdown margin can generally be achieved with even more than 5.1% $\Delta k/k$ excess.

4.7.4 Excess Reactivity

The total excess reactivity that the NSTF is authorized to have loaded into the reactor during operation is 5.2% $\Delta k/k$. This amount provides for the various negative-reactivity effects associated with operation and use of the reactor as well as for operational flexibility. Typical excess reactivity requirements, excluding experiments, as given in the NSTF SAR update (1981), are as follows:

Xenon override	1.7% $\Delta k/k$
Temperature coefficient	0.15 $\Delta k/k$
Power defect (0-2 MW)	0.35 $\Delta k/k$
Total	2.20% $\Delta k/k$

The limitation of 5.2% $\Delta k/k$ excess allows up to 3% $\Delta k/k$ associated with experiments, without significant core rearrangement. While it is apparent that the fundamental criterion is maintaining ensured capability to shut the reactor down, hence the minimum shutdown margin, also imposing a limit on excess reactivity helps ensure that the SAR analyses are applicable to the operational core.

4.7.5 Experiments

The licensee's proposed Technical Specifications limit the combined absolute reactivity worth of all experiments to 3.0% $\Delta k/k$. The staff has analyzed this limitation based on information provided by the licensee in Appendix A to the Technical Specifications (1981) and the update to his SAR (1981).

If this excess reactivity were added to an operationally loaded reactor, the total excess would be 3.0 + 2.20 = 5.2% $\Delta k/k$. This is consistent with the authorized excess reactivity discussed in Section 4.7.4. Further, this also is consistent with the required minimum shutdown margin, if the effect of the maximum worth control safety blade is no more than 3.4% and the total worth of all control safety blades is not simultaneously less than 9.1% $\Delta k/k$. As noted in Section 4.7.2, this is the likely situation. In the event that either the shutdown margin or the maximum excess reactivity authorization would be exceeded by a proposed loading of experiments, these limits would prevail.

The proposed Technical Specifications for the NSTF reactor (1) define a movable experiment as one that can be inserted, removed, and manipulated while the reactor is critical and (2) limit the reactivity worth of such experiments to $\pm$0.3% $\Delta k/k$ per experiment. Experience at the NSTF PULSTAR has shown that this worth is adequate for isotope production needs and that a combination of temperature coefficient, Doppler effect coefficient, and operator action provides easy control of any change resulting from the insertion or removal of the experiment. In addition, the 0.3% $\Delta k/k$ worth is well below the 1.5% $\Delta k/k$ positive reactivity insertion required to obtain the previously authorized routine pulse (35 MW·sec thermal energy released).

Unsecured experiments are defined in the proposed Technical Specifications as those that are not held in position mechanically with sufficient force to overcome the expected effects of hydraulic, pneumatic, buoyant, or other forces that are normal to the operating environment or by forces arising from likely credible malfunctions. Unsecured experiments are limited by the proposed Technical Specifications to ±0.6% Δk/k per experiment. This worth is less than β (i.e., less than 1.0$) for the NSTF reactor and is well below the reactivity worth required to produce the previously authorized routine energy release (35 MW·sec) for pulse-mode operation.

The staff has reviewed the proposed limitations on the worth of movable, unsecured, and secured experiments and concludes that they are conservative and provide reasonable assurance that failure of single experiments resulting in a positive reactivity insertion would not result in damage to the fuel or reactor components. However, simultaneous removal of a combination of movable and unsecured experiments equivalent to the maximum absolute worth limitation for all experiments (3% Δk/k) has the potential for an energy release that could exceed that previously tested. For this reason, the staff has required that the Technical Specifications limit the combined worth of movable and unsecured experiments to 1.7% Δk/k. A positive reactivity insertion of 1.7% Δk/k corresponds to the licensed maximum energy release (44 MW·sec) for the NSTF reactor operating in a pulse mode. The PULSTAR test program demonstrated that a 44 MW·sec energy release would not result in fuel damage (WNY-017, 1964; WNY Summary Report, 1966).

On the basis of the information presented above, the staff concludes that (1) the limitation on total absolute experiment reactivity worth of 3% Δk/k with a further limitation of 1.7% Δk/k total reactivity worth for movable and unsecured experiments, (2) a limitation of ±0.3% Δk/k per experiment for experiments that may be moved when the reactor is critical, (3) a limitation of ±0.6% Δk/k per unsecured experiment that may be moved when the reactor is subcritical by at least 3% Δk/k, and (4) operation in compliance with minimum shutdown margin requirements of the Technical Specifications provides 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 plus the worth of the highest worth rod is sufficient to ensure that the reactor can be adequately shutdown under all likely conditions.

4.8 Functional Design of Reactivity Control System

4.8.1 Standard Control Element Drives

The drive units for the standard control-safety blades are reversible electric motors with a rack-and-pinion-drive mechanism that raises or lowers an electromagnet. The drive mechanisms are activated by switches from the control console. The control blade drives could be positioned to cover any of the spaces in the grid plate. The limits of stroke of the control elements are set by adjustable, cam-operated, snap-action switches mounted on the rack guide. Except for the control blade (former transient rod) drive, the other five control elements may be activated both individually and in groups of two, three, four, or five. If electrical power is removed from the electromagnet, the control-safety blade falls into the core by the force of gravity.

4.8.2 Transient Control Rod Drive

As discussed earlier, the system is being disabled, so it will not be described further here.

4.8.3 Scram-Logic Circuitry

The NSTF reactor is equipped with a scram-logic safety system that receives signals from core instrumentation (neutron flux density detectors) and other reactor parameters to initiate a scram by removing electrical power from the control-safety blade magnets.

The reactor parameters that can initiate these scrams are

(1) high reactor power
(2) low coolant flow (forced convection)
(3) low pool water level
(4) flapper valve open
(5) high pool/core coolant inlet temperature
(6) dry chamber door not fully closed
(7) operator/personnel manual scram
(8) loss of safety chamber high voltage

The safety system is discussed in more detail in Section 7.

4.8.4 Conclusion

The NSTF PULSTAR is equipped with a safety and control system typical of non-power reactors, incorporating multiple control-safety blades and multiple and redundant sensors that can initiate a scram. There is sufficient redundancy of control-safety blades that the reactor can be shut down safely even if the most reactive control-safety blade fails to insert upon receiving a scram signal.

In addition to the electromechanical safety controls for both normal and off-normal operation, the negative bulk temperature coefficient and the large, prompt, negative Doppler effect temperature coefficient typical of a low enrichment uranium core provide an inherent backup safety feature.

In accordance with the above and the details presented in Section 7, the staff concludes that the reactivity control systems of the NSTF reactor are designed and function adequately to ensure safe operation and safe shutdown of the reactor under all normal operating conditions.

4.9 Operational Practices

The NSTF has implemented a preventive maintenance program that is supplemented by a detailed preoperational checklist to ensure that the reactor is not operated at power unless the appropriate safety-related components are operable. The reactor is operated by NRC-licensed personnel in accordance with explicit operating procedures, which include specified responses to any reactor control signal. All proposed experiments involving the use of the NSTF reactor are reviewed by the Operating Committee for potential effects on the reactivity of the core or damage to any component of the reactor, as well as for possible malfunction of the experiment that might lead to the release of contained radioactivity. The Operating Committee can request additional review of specific experiments by the Nuclear Safety Committee. Since approximately 1980, staffing of the NSTF seems to have been minimal for performing all surveillance functions in a timely manner. The licensee has recently taken steps to improve the staffing complement of the NSTF, such as an engineer joining the staff in April 1983.

4.10 Conclusions

The staff concludes that the NSTF PULSTAR is designed and built according to good 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 original and proposed Technical Specifications revisions and other pertinent documents associated with the reactor. The design features are similar to the Bulk Shielding Reactor at Oak Ridge as well as to other pool-type research reactors manufactured by the suppliers of the NSTF reactor, with the exception of the fuel and fixed core position. The fuel, which is Zircaloy-2 clad low-enriched sintered UO_2, is used in two research PULSTARs and is very similar to power reactor fuel and the fuel used in some SPERT tests. On the basis of its review of the NSTF reactor operating experience since 1964 with the present fuel and experience with other pool-type facilities, the staff concludes that there is reasonable assurance that the reactor can continue to operate safely, as limited by its proposed revised Technical Specifications for the proposed duration of the license.

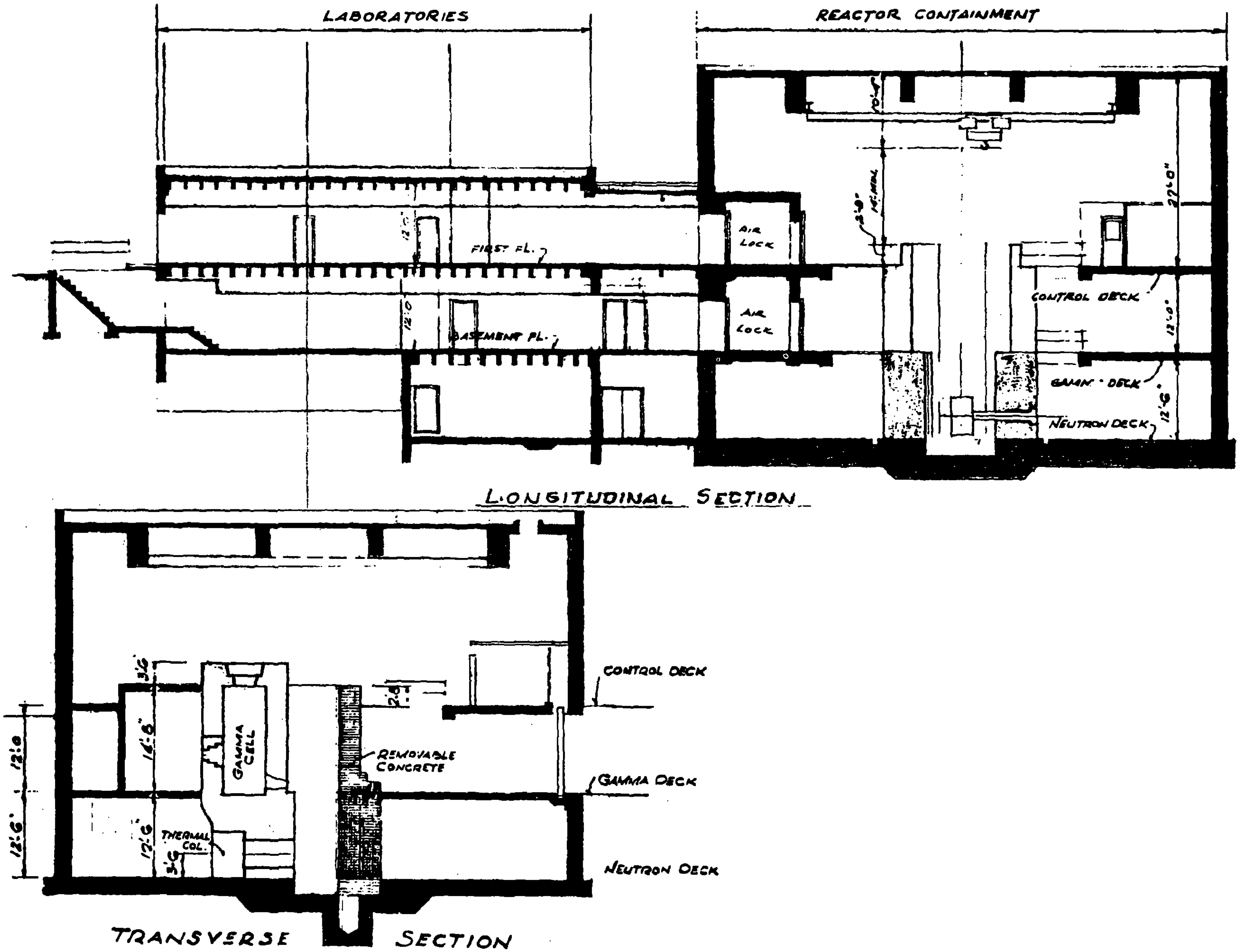

Figure 4.1 Vertical cross section of building
Source: Western New York Research Center

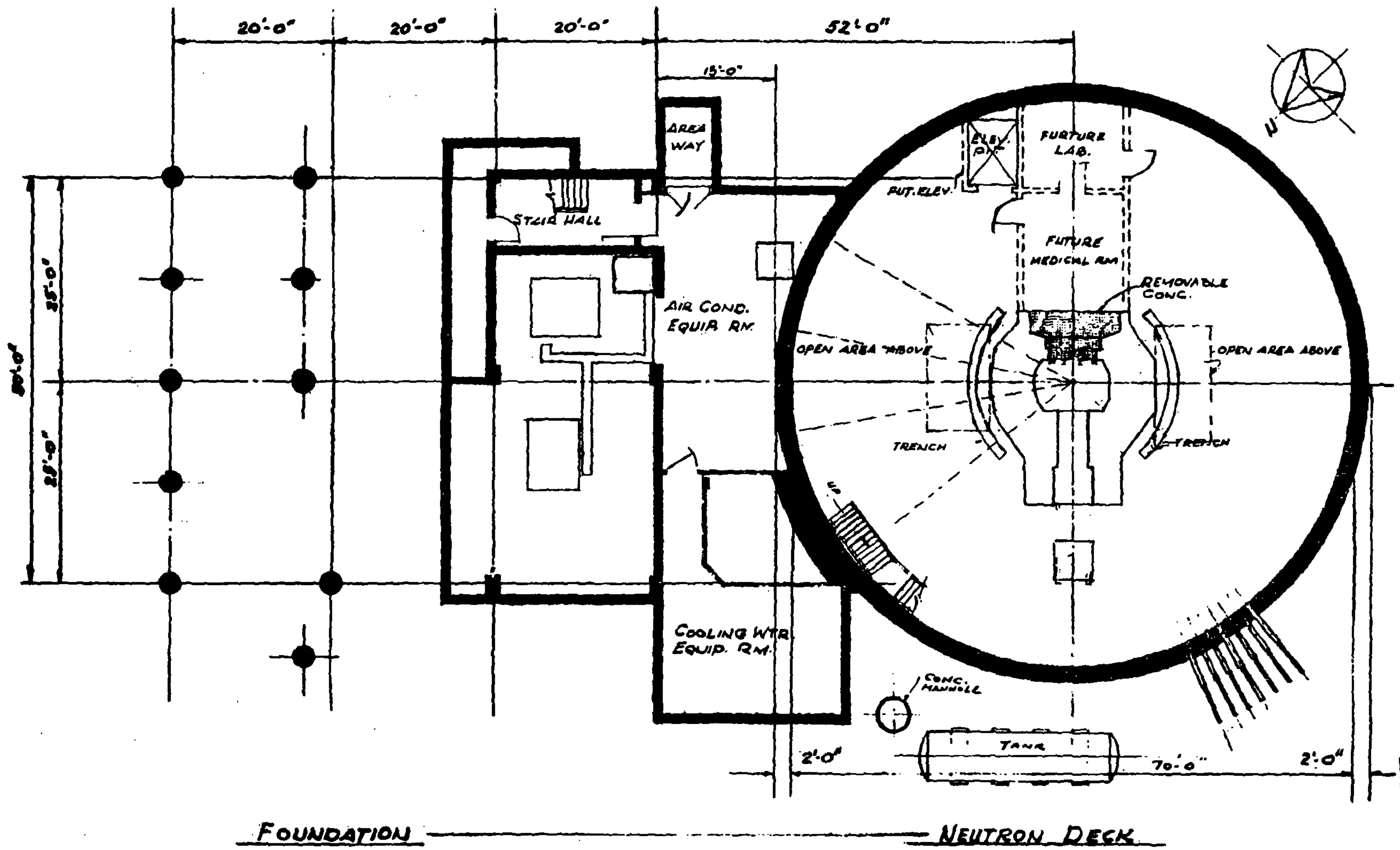

Figure 4.2 Cross sectional view of reactor facility at core level

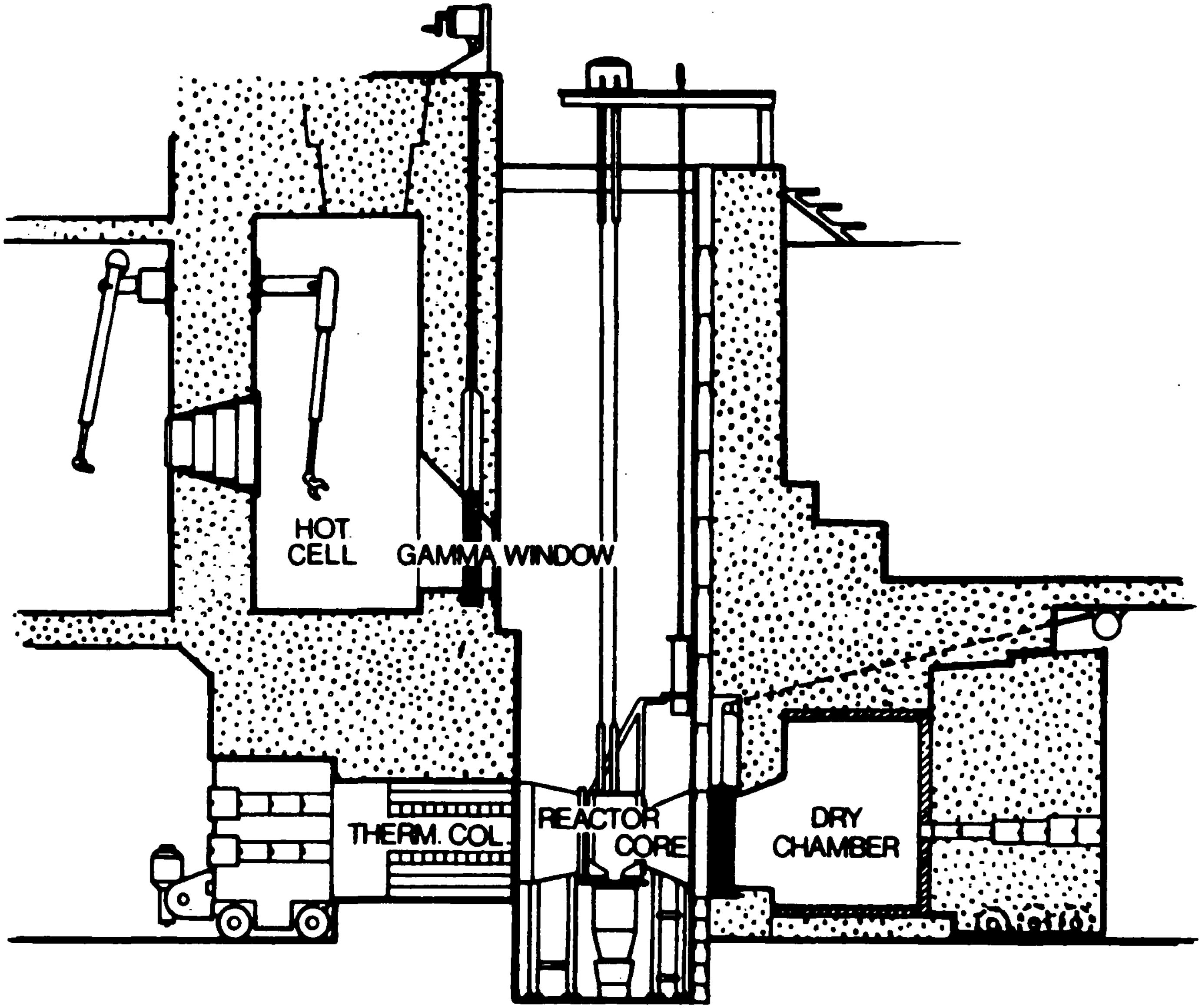

Figure 4.3 Cross sectional view of reactor

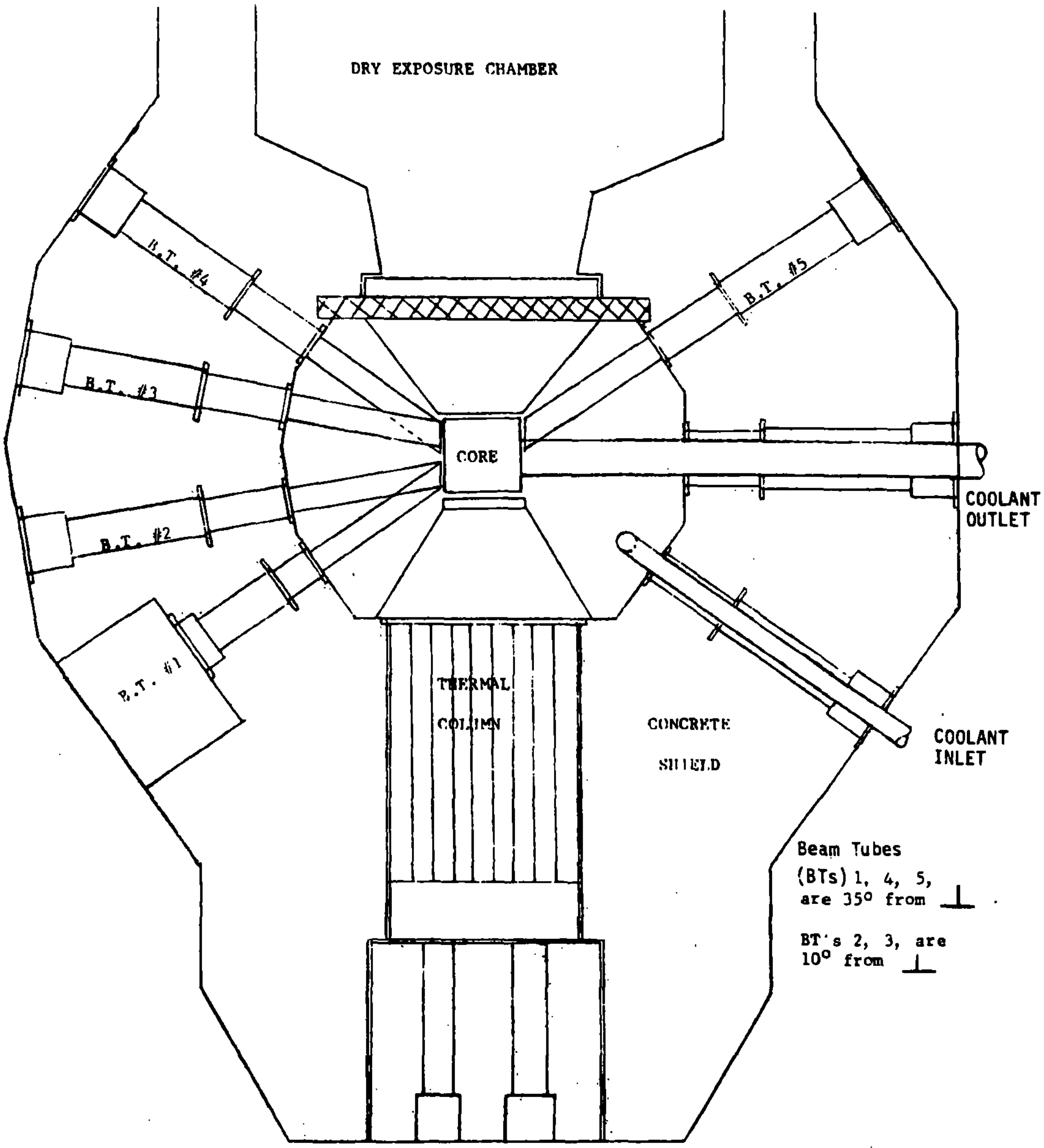

Figure 4.4 Horizontal cross section of the reactor

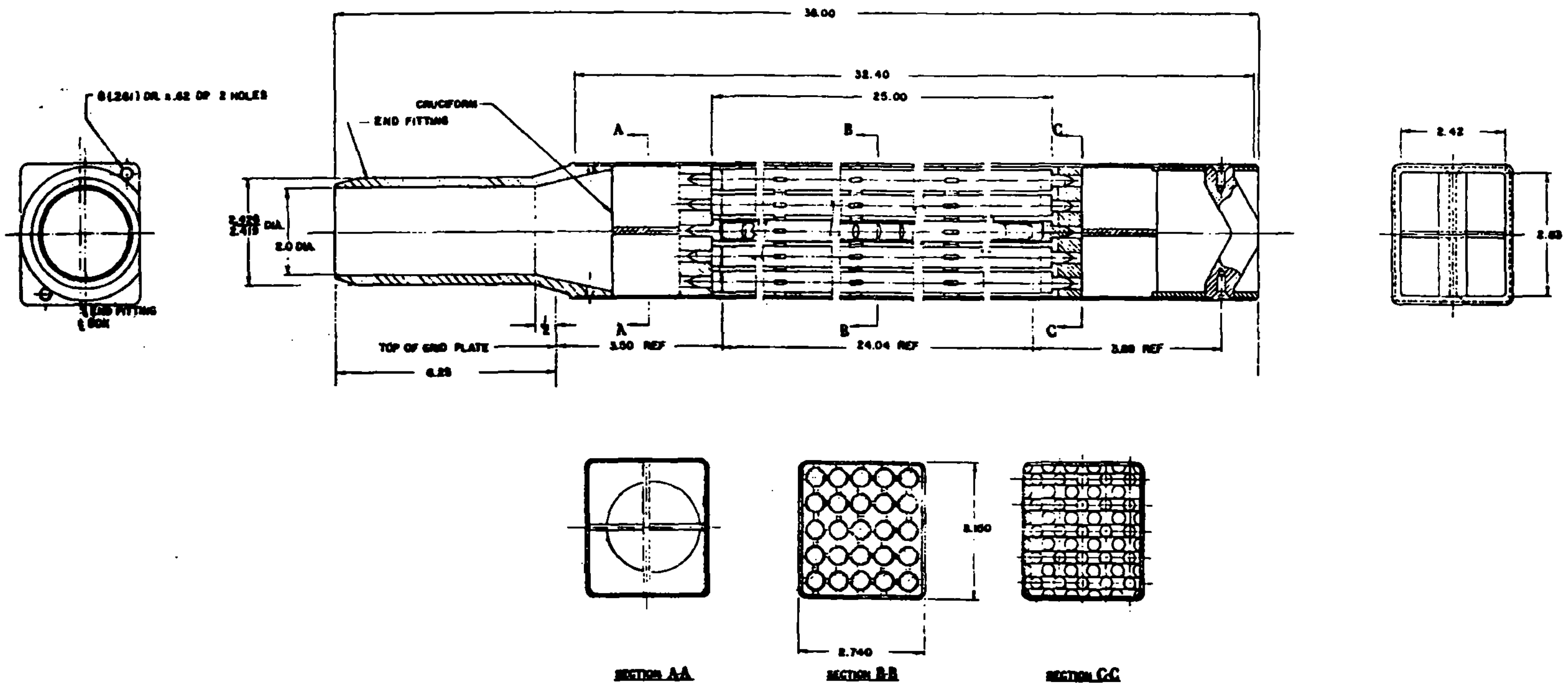

Figure 4.5 Fuel assembly

Table 4.1 Calculated maximum power levels and temperatures corresponding to critical heat fluxes (DNB)

		Temperature °C			
No. of fuel assemblies	Power MW	Fuel Centerline	Fuel Surface	Inner Cladding	Outer Cladding
16	3.1	2,090	1,001	141	93
25	4.39	1,795	868	134	94
35	5.57	1,636	796	131	94

Table 4.2 Licensee's safety limits for forced convection cooling

Parameter	Limit
Core power (maximum)	3.3 MW
Pool level above core (minimum)	5.18 m
Coolant core inlet temperature (maximum)	60°C
Coolant flow (minimum)	63 l per second

5 REACTOR COOLING SYSTEM

The cooling system for the NSTF reactor is composed of three subsystems: (1) primary coolant, (2) secondary coolant, and (3) purification and makeup system.

The cooling system instrumentation and controls are described in Section 7, and the emergency pool filling system is described in Section 6. Schematic drawings of the systems are shown in Figures 5.1 and 5.2.

5.1 Primary Cooling Subsystem

The reactor core is submerged in an aluminum-lined concrete pool filled with demineralized water. In the forced convection mode, coolant water is drawn down through the core into an outlet plenum and out of the pool to a holdup tank. The purpose of the holdup tank is to increase the transit time of the water to allow ^{16}N radioactivity to decay to an acceptable level before the coolant is returned to the pool. The primary coolant pump forces the coolant through the shell side of the heat exchanger and back into the pool. There is a counterbalanced flapper valve in the reactor outlet plenum that is held closed by the primary coolant pump suction. If the primary pump or the water flow stops, gravity opens the flapper valve allowing heat transfer from the core to the pool water by natural convection.

Initially, the reactor outlet plenum and thus the reactor core was supported on the discharge pipe flange at the pool floor. When pipe leakage caused the abandonment of this discharge line and the original coolant return line, the reactor plenum support was provided by four aluminum pipes. Reactor coolant flow now is routed from the discharge plenum through an 8-in. pipe located in what was originally the 12-by-12-in. beam-tube penetration and to the holdup tank located immediately outside the containment building in an underground vault. Coolant return flow from the heat exchanger enters the pool through one of the original 6-in. beam-tube penetrations. At the discharge end of this pipe, an elbow directs the return flow downward towards the pool floor to prevent flow-induced movement of the control blades.

Valves in the pool coolant discharge line (P-8) and return line (P-3) can be closed to isolate the pool in case of primary coolant system component failure. These isolation valves are manually operated from a station immediately outside the control room.

The primary coolant pump has a nominal capacity of 1,200 gpm. The primary flow is normally maintained at 1,150 gpm by adjusting manual valves in the pump room.

The reactor pool temperature is maintained at the desired value by adjusting the secondary coolant flow with a pneumatically operated bypass valve. The valve is controlled by a temperature sensor on the primary coolant outlet on the heat exchanger.

5.2 Secondary Cooling System

The secondary coolant subsystem consists of the tube side of the heat exchanger, the secondary coolant pump, the cooling tower, and associated piping and valves.

The secondary cooling water removes heat from the primary coolant in the heat exchanger and dissipates it through the cooling tower to the outside atmosphere. The coolant is drawn from a sump in the cooling tower basin, passed through the secondary pump, the heat exchanger, and to the spray trays at the top of the cooling tower.

Frequently the primary system pressure is higher than the secondary system, providing the potential for leakage of radioactive primary coolant into the secondary coolant. To guard against this possibility, weekly tests of the secondary coolant are conducted. In addition, the heat exchanger is cleaned and carefully inspected annually. In response to the staff's questions, the licensee has analyzed possible events for which the heat-exchanger tubes develop leaks and primary coolant enters the secondary side of the system. Because the concentration of radioactivity in the primary water is normally below the levels of 10 CFR 20, Appendix B, this pathway for release of radioactive liquid effluents does not cause significant potential radiation exposure to the public.

5.3 Coolant Purification and Makeup Systems

The quality of the primary coolant is maintained by circulating a portion of the coolant flow through a prefilter, a demineralizer, and an afterfilter. The inlet to this purification loop is downstream of the heat exchanger, and the purified coolant is returned at the inlet to the primary circulation pump. The quality of the secondary coolant is maintained by periodic bleeding of the system and replacing some of the chemical treatment solution.

Water to replace coolant lost from the reactor pool by evaporation is taken from the city water system, passed through a filter and demineralizer, and stored in a tank. This water is introduced into the coolant system at the inlet to the primary purification loop.

5.4 Conclusion

The staff concludes that the NSTF reactor cooling system is adequate to remove heat from the fuel and prevent melting under all normal and reasonable offnormal operating conditions. Potential coolant leakage between the primary and the secondary systems in the heat exchanger would not lead to significant radiation exposure to the public. There is reasonable assurance that the system can continue to function adequately for the proposed duration of the license.

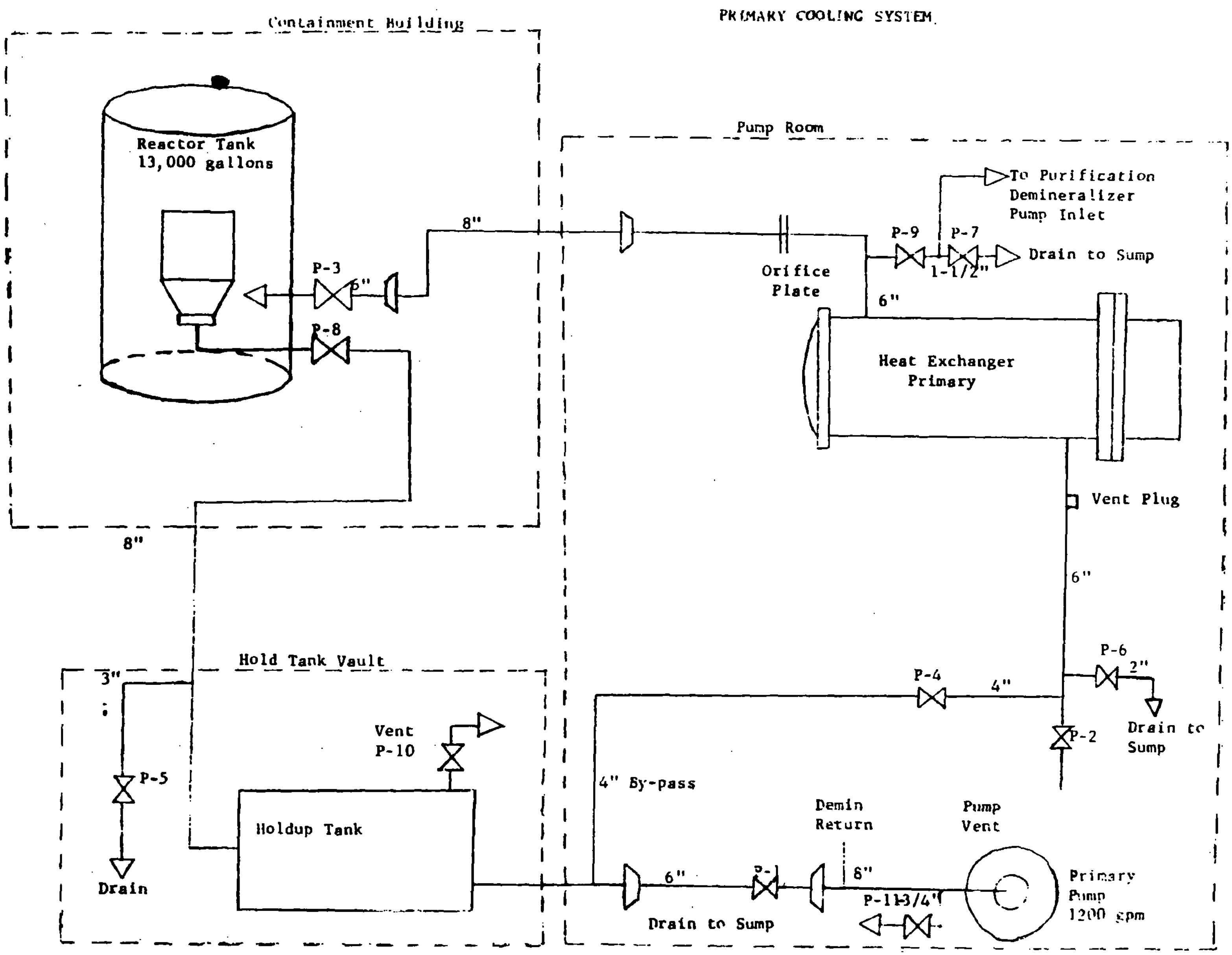

Figure 5.1 Primary cooling system

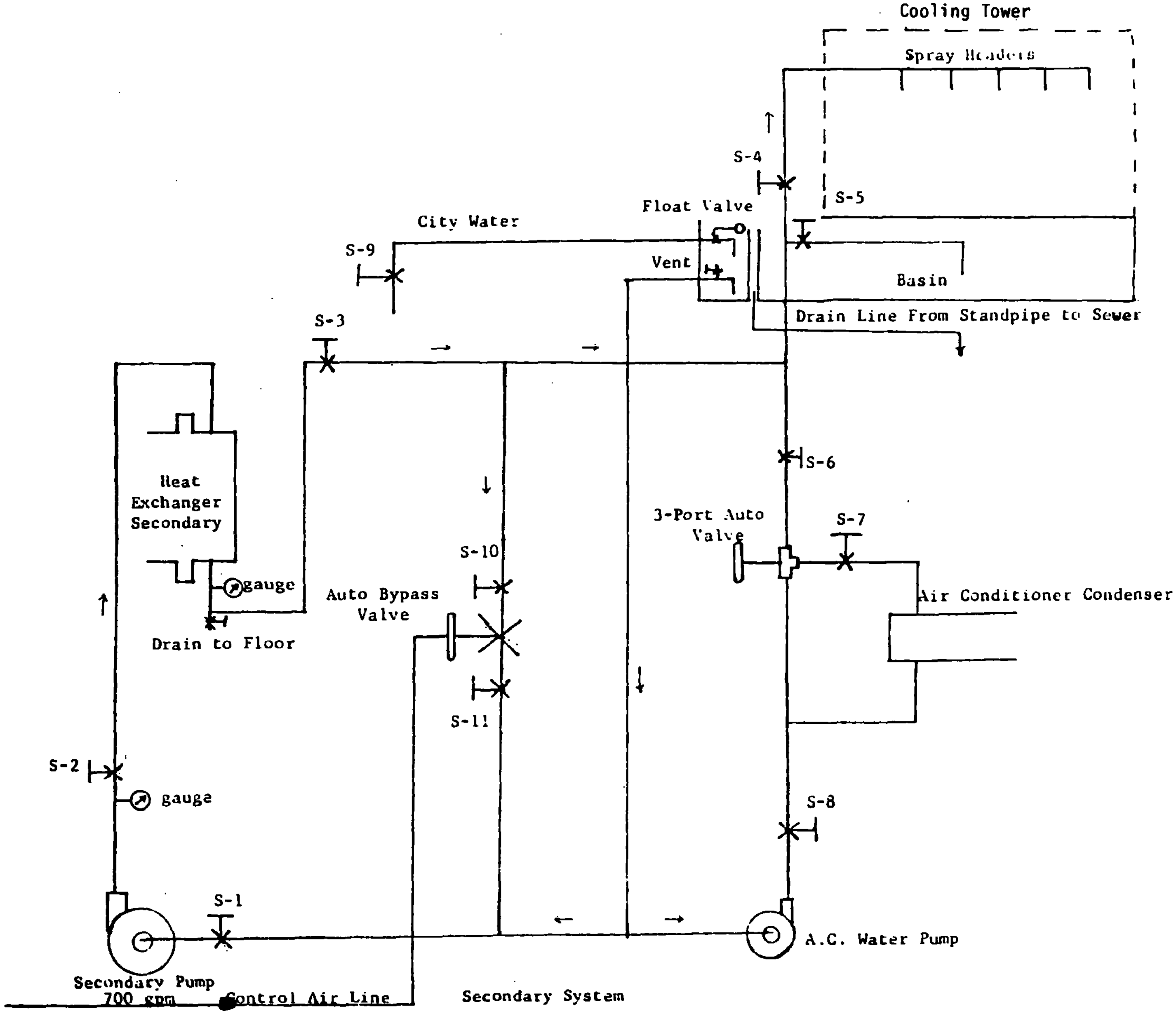

Figure 5.2 Secondary cooling system

6 ENGINEERED SAFETY FEATURES

The engineered safety features associated with the SUNYAB NSTF reactor are the emergency coolant replacement (pool-filling) system and the building ventilation system. The ventilation system is shown in Figure 6.1.

6.1 Emergency Coolant Replacement System

An emergency pool-filling system is available for adding city water directly to the pool. A normally closed solenoid valve is installed in a line between the city water system and the pool. In the event of a large leak from the reactor pool or a primary cooling system component, low pool water level is sensed by a transducer that automatically causes the valve to open and raw water to flow into the pool. This valve remains open and flow continues until it is manually reset. Thus it would be possible, under certain conditions, for the pool to overflow and cause containment building flooding. Such an event would have no significant radiological consequence, but could cause considerable damage and expense to experimenter's equipment. As discussed in Section 14.2, loss of all pool water will not cause fuel assembly damage. For these two reasons, a change in the Technical Specifications has been proposed that would eliminate the solenoid valve and allow for manual operation of the emergency pool-filling system.

The evaluation of loss of coolant in Section 14.2 indicates that in the event of coolant loss, the temperature of the fuel would not reach its maximum for at least an hour, and this temperature would not exceed the melting point of either the cladding or the uranium dioxide pellets. Further, if the loss of pool water occurred, the loss of shielding would increase the radiation level in the reactor room and radiation detectors would alarm and alert the operator to initiate emergency action. For the above response, there is no need to have a fast-acting automatic pool-filling system that could cause damage by malfunctioning. On the basis of the above considerations, the staff concurs in the substitution of manually operated valve for the automatic valve.

6.2 Ventilation System

The reactor building air conditioning system provides sufficient fresh filtered air for four air changes per hour. A blower pulls air from the occupied spaces of the building and some fume hoods with low radioactive release potential through a bank of prefilters and high efficiency particulate air (HEPA) filters and exhausts it through a 36-in. duct at the building roof level. While the reactor is operating, the building is maintained at a 1/2-in. water gage negative pressure by means of an automatically adjusted damper in the building air fan inlet. Air locks in personnel access passages and inflatable seals on the rollup door allow this negative pressure to be maintained.

A separate supply system can provide air for the fume hoods. Exhaust from the hoods and from the experimental areas with the potential for radioactive release is pulled through individual HEPA filters into a common exhaust duct connected to a blower in the mechanical equipment room. From there the air flows through an underground duct to a booster fan located at the base of the power plant stack and then up a duct mounted inside the 50-m-high power plant stack, exhausting at the top. Although power plant stack discharge helps to dilute reactor facility effluent, no credit is taken for it in radioactive material dispersion calculations.

If both the radiation monitor mounted on the bridge above the reactor pool and the one monitoring the 36-in. building exhaust duct sense an abnormally high radiation level, the following events occur automatically.

(1) The two inlet ducts and the two exhaust ducts are closed by hydraulically operated dampers.

(2) The two exhaust blowers in the reactor facility shut down. The 6,000-ft^3-per-minute blower at the base of the stack remains in operation.

(3) The damper in a 6-in. emergency exhaust duct opens, which allows a controlled release of building air into the exhaust duct in the mechanical equipment room and from there up the stack.

The flow through the 6-in. emergency exhaust duct is controlled to maintain the reactor building at 0.018 psig negative pressure. Cleanup of radioactive contamination is accomplished by filtering the emergency exhaust through a bank of HEPA filters bracketing a charcoal filter.

6.3 Conclusions

The staff concludes that the emergency pool fill system to be modified as discussed above is adequate to mitigate the consequences of a pool or cooling system leak.

The staff also concludes that the design and operating features of the reactor building air control system give assurance that airborne radioactivity in the building will be adequately confined, or diluted and delayed in release to unrestricted areas.

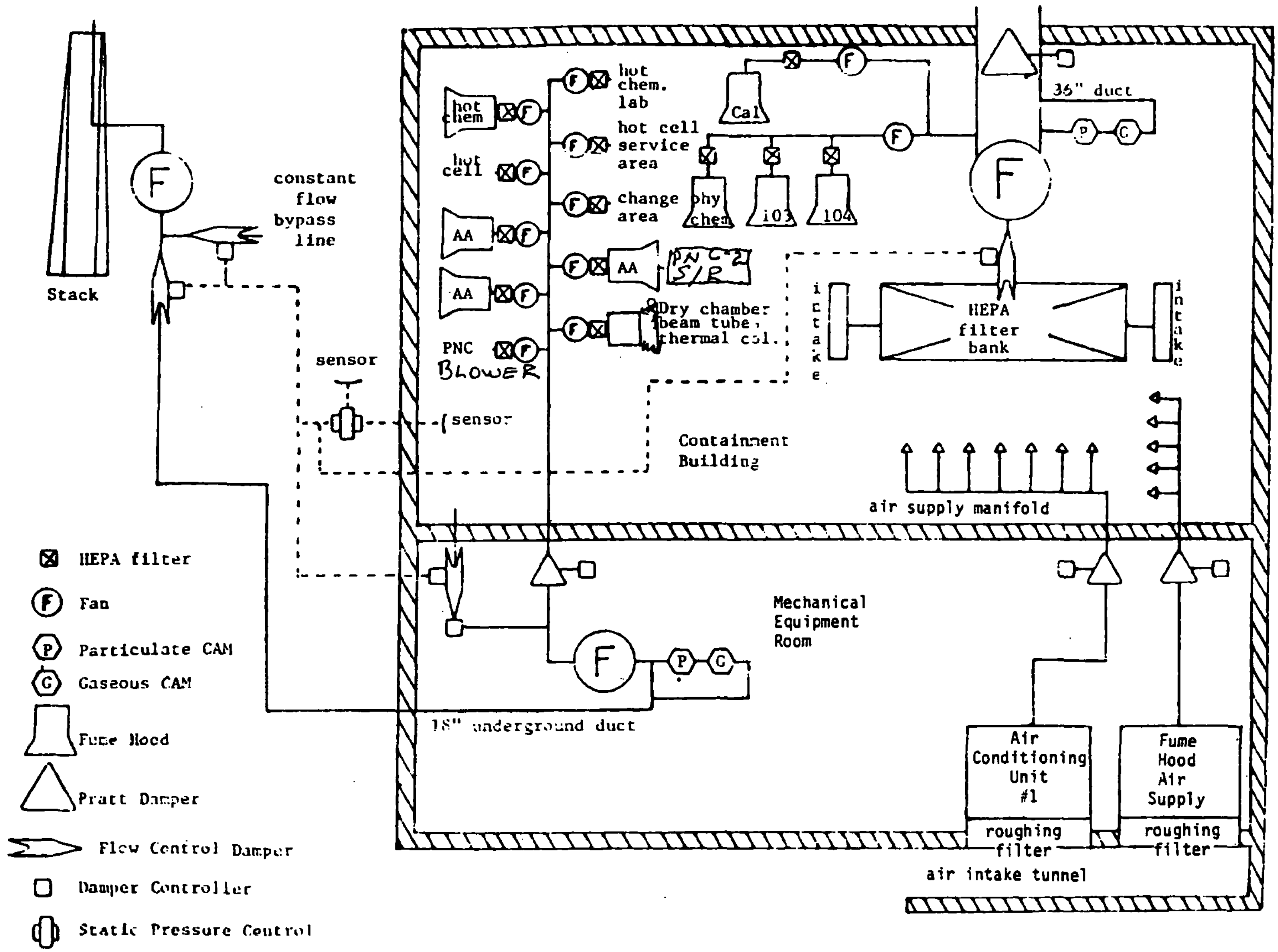

Figure 6.1 Containment building air supply and exhaust system

7 CONTROL AND INSTRUMENTATION

The control and instrumentation systems at the NSTF reactor are similar to those in wide use for research reactors in the United States. Control of the nuclear fission process is achieved by using five control-safety (scrammable) blades and one control (nonscrammable) blade. The instrumentation system, which is interlocked with the control system, is composed of both nuclear and process instrumentation and is generally characterized by modern components. The NSTF has a program in operation to replace older instruments with state-of-the-art systems, providing the same functions more reliably. The control and instrumentation systems are summarized in Table 7.1.

7.1 Control System

The control system is composed of both nuclear and process control equipment in which safety-related components are designed for redundant operation in case of single failure or malfunction of components essential to the safe operation or shutdown of the reactor.

7.1.1 Nuclear Control Systems

Control of the reactor is achieved in the standard way by inserting and withdrawing neutron absorbing control blades by the use of control drive units mounted on the bridge structure over the pool. One control blade has a solid coupling and cannot be scrammed. Five control elements are supported by electromagnets so that any electrical power interruption will result in the elements falling by gravity into slots in the core, causing a reactor scram. The control element drives are controlled from the control room by the reactor operator. The control blade systems are discussed in more detail in Section 4.2.2.

7.1.2 Supplementary Control Systems

These control systems, also designated as process control systems, are designed to control the various processes involved in reactor operation but do not directly relate to safety. Included in this category are circuits and devices that energize and/or monitor coolant pumps, flapper valve, and coolant parameters such as flow rate, temperature, and conductivity.

7.2 Instrumentation System

The instrumentation system is composed of both nuclear control 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. Automatic scram function is provided through the safety amplifier, discussed below.

7.2.1 Nuclear Instrumentation

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

(1) Log count rate or startup channel - This channel receives data from a movable fission chamber. Its primary purpose is to monitor the reactor power during startup.

(2) Linear-N power or linear power channel - This channel receives data from an electrically compensated ion chamber. This channel monitors the reactor power level in the range of 1 W to greater than 2 MW and provides the signal for automatic servo-control of reactor power.

(3) Log-N power channel - This channel also receives data from a compensated ion chamber and monitors the reactor power level from a few watts to greater than 2 MW. This channel also provides the signal to the period amplifier for indication of the reactor period.

(4) Safety channels - Two uncompensated ion chambers provide signals for two independent channels, which give the redundancy to scram the reactor in response to abnormally high power. These chambers share a common power supply, failure of which causes a reactor scram.

(5) ^{16}N power channel - This channel uses a gamma-ray sensitive ion chamber to monitor the ^{16}N level in the primary coolant. This channel monitors the reactor power level from about 50 kW to more than 2 MW.

All neutron-sensing ion chambers are located in the pool outside of the core and are independently adjustable over a limited distance to allow calibration of their respective channels to the reactor thermal power derived from primary coolant flow rate and core differential temperature (ΔT) measurements.

7.2.2 Reactor Safety System

The control and instrumentation systems are interconnected through a safety amplifier. This unit provides current for the electromagnets that support the control-safety blades, as well as current for the ion chambers. Each ion chamber is provided with an independent amplifier circuit, which will cause a fast scram upon receipt of an appropriate trip signal or upon failure. The safety circuit provides for either a fast scram by decreasing the dc current in the holding magnets, or a slow scram by turning off the ac power supply for the magnets.

7.2.3 Inhibits and Annunciation

Inhibit signals that will prevent control blade removal (reactor startup) are provided by low neutron countrate in the startup channel, and if the chart recorders are inoperable on the log count rate, linear-N, or log-N instruments.

A control console-mounted annunciator panel of lights provides the operator with information on conditions of important variables related to reactor operation. The annunciator is energized continuously through the main power disconnect switch. Following annunciation of an event the condition must be corrected and the operator must reset to restore the annunciator to normal operating condition. Table 7.1 summarizes the functions of the various instruments.

7.3 Supplementary Instrumentation

Additional process instrumentation consists of the facility fixed radiation monitoring systems.

The fixed area monitors include three located on the neutron deck, one located under the reactor bridge, and one in the hot cell. These monitors provide exposure rate indication and alarms both locally and in the control room. Alarm set-points are specified in a facility operating procedure. The hot cell monitor controls an interlock on the hot cell door. Coincident alarms from the reactor bridge and building air exhaust monitor will activate the containment building ventilation damper closure system.

The effluent monitors comprise two separate air monitoring systems: the building and exhaust stack systems. The building system consists of the building air gaseous continuous β-γ monitor and an in-line fixed filter. The fixed filter is removed for laboratory evaluation of particulate releases. The stack system consists of the stack air (gaseous) continuous β-γ monitor and the stack particulate (fixed filter) continuous β-γ monitor. The alarm points for the two gaseous monitors and the stack particulate monitor are specified in a facility operating procedure and are posted in the control room. The monitors provide both local and control room indication and alarm. The outputs of the two gaseous monitors and the stack particulate monitor are recorded on a strip chart in the control room.

The primary coolant β-γ monitor is located on the core coolant exit line downstream from the holdup tank, thus eliminating the effect of the ^{16}N radioactivity on the monitor. The monitor provides both local and control room indication and alarm. The alarm set point is variable and based on ambient conditions.

7.4 Conclusions

The control and instrumentation systems at the NSTF reactor are well designed and maintained. The quality of workmanship of individual components is high. Redundancy in the important ranges of power measurements is ensured by overlapping ranges of the log-N and linear power channels.

The licensee's performance specifications for the individual components used throughout the system exceed the minimum required. This helps to ensure system reliability and decreases the chances of serious simultaneous multicomponent failures.

The control system is designed so that the reactor is automatically and safely shut down if electrical power is lost. However, emergency power is provided to functions required to provide information on facility status, (see Section 8).

On the basis of its review of the control and instrumentation systems, the staff has concluded that these systems are adequate to ensure safe operation of the reactor within the context of the revised Technical Specifications and the license conditions for the duration of the proposed renewal period.

Table 7.1 Required instrumentation

Instrument channel	Minimum number operating	Function	Set point	Modes in which required
Log count rate (a, b)	1	Indication/ inhibit	<2 cps; 9800 cps	Startup
Linear power (a, b)*	1	Indication	-	All
Log power (a, b)	1	Indication	-	All
Period (a, b)	1	Indication		All
Power safety (a, b)	2	Indication/scram	120%	All
Power safety (a, b)	1	Reverse	110%	All
Manual scram (a, b)	5	Scram	-	All
Dry chamber door open (a, b)	1	Scram	Door < full closed	All
Flow (a, b)	1	Indication/scram	68 lps	Forced convection
Flapper open (a, b)	1	Scram	> 250 kW	Forced convection
Water level low (b)	1	Scram	6.13 m over fuel	All
Water level low (b)	1	Annunciation	6.43 m over fuel	All
Water level high (b)	1	Annunciation	6.74 m over fuel	All
Pool temperature (a)	1	Scram	52°C	Forced convection
Core outlet temperature (a)	1	Annunciation	52°C +ΔT	Forced convection
Recorders inoperative (b)	3	Inhibit	-	Startup
Conductivity (a)	0	Annunciation	200 K ohms	None
EPF valve open (a)**	0	Annunciation	Valve open	None
Demineralizer temperature (a)	0	Annunciation	43°C	None
Suction valve closed (a)	1	Disables primary pump	Valve < full open	Forced convection

See footnotes at end of table.

Table 7.1 (Continued)

Instrument channel	Minimum number operating	Function	Set point	Modes in which required
Servo deviation (a)	1	Annunciation/ transfer to manual	±10%	Servo control
Blade position - analog (a)	1 of 2	Indication	-	All
Blade position - digital (b)	1 of 2	Indication	-	All
Nitrogen-16 (a)	2 of 3	Indication	-	Forced convection
Primary temperature (a)	2 of 3	Indication	-	All
Core Δ T	2 of 3	Indication	-	Forced convection

(a) - Operability check required prior to operation
(b) - Test and/or calibration required four times/year
* - Linear power channel and any recorder may be inoperable for short periods while operating
** - Emergency pool filling valve

8 ELECTRICAL POWER

8.1 Main Power

Three-phase power is supplied to the reactor facility at 5,000 V by commercial utility. Transformers within the facility supply distribution panels with 440 V, 220 V and 110 V as required.

8.2 Emergency Power

Emergency power for the facility is provided by a 5-kVA natural gas-fueled motor generator. An electric power failure causes this generator to start automatically and assume the load when up to speed. Emergency power is provided to the following.

(1) fire alarm system

(2) building exit signs

(3) building evacuation alarm horn

(4) radiation monitors for effluent streams, the hot cell, and the bridge monitors

(5) reactor building intercommunication system

There is an emergency generator located in the power plant. Although not directly related to the reactor facility, the exhaust blower located at the base of the stack described in Section 6.2 may be operated from this generator.

8.3 Conclusion

On the basis of the staff's review and the above information, the staff concludes that the electrical power provisions at the SUNYAB NSTF provide reasonable assurance of adequate operation and that loss of offsite power will lead to safe shutdown of the reactor (Section 7.4), with adequate monitoring functions operable on emergency power.

9 AUXILIARY SYSTEMS

The auxiliary systems discussed in this section are the fuel-handling and storage system, the compressed air system, the hydraulic system, and the fire protection system. The building ventilation system is discussed in Section 6 as an engineered safety system.

9.1 Fuel Handling and Storage

Unirradiated fuel pins and fuel assemblies are stored in locked vaults inside the containment building. Each vault has no more than 15 storage cylinders, each capable of holding 1 fuel assembly or a maximum of 25 fuel pins. The fuel spacing is such that a critical assembly is not possible even with a vault completely filled with water.

Irradiated fuel normally is stored in the reactor pool either on the grid plate or in cylindrical storage racks. When it is necessary to perform work in the lower tank, fuel may be transferred in a transfer cask or through the pass-through tube to the hot cell, or to a shielded facility designed for the purpose within the containment building, in accordance with Section 5.7.2 of the revised Technical Specifications. Transfer of fuel assemblies in the pool as well as work on the fuel assemblies is accomplished under water using long-handled tools.

9.2 Compressed Air System

There are three compressed air subsystems.

(1) A large compressor supplies air to the air lock and truck door inflatable seals, the demineralizers, the fume hoods, and laboratory benches.

(2) A second subsystem supplies air to the reactor building ventilation controls. Normally one of two compressors supplies this requirement. When a single unit cannot maintain pressure, the second compressor automatically starts.

(3) A third subsystem consists of a small compressor dedicated to supply air to the primary coolant flow transmitter (orifice plate). All three compressed air susbystems are connected through normally closed valves to allow cross connection.

9.3 Hydraulic System

Electrically driven hydraulic pumps supply high-pressure hydraulic fluid to open and close the ventilation system dampers. In the event of a power failure, the system pressure is maintained by air bladders in the accumulators for 10 to 15 min, allowing operation of the dampers during this period.

9.4 Fire Protection System

Fire fighting equipment includes several types of portable fire extinguishers located throughout the facility. A hydrant located outside the facility is supplied with city water.

Fire alarm boxes are installed in several locations in the facility. Campus security police, who control traffic and protect the public, and the City of Buffalo Fire Department personnel, who combat fire, respond to fire alarms.

In addition to fire drills, Fire Department personnel are given periodic orientation tours and lectures about the reactor facility and potential radiological hazards.

9.5 Conclusion

The fuel-handling and storage system design is adequate to ensure that reactor fuel can be moved, serviced, and stored without danger to operating personnel or the public because of radioactivity of the fuel or a possible accidental criticality event.

The facility compressed air and hydraulic systems are designed to adequately service the facility under normal and emergency conditions that might occur. On the basis of the above, the staff concludes that the NSTF auxiliary systems will provide the necessary service to the reactor facility for the requested license period.

10 EXPERIMENTAL PROGRAMS

The NSTF reactor serves as a source of ionizing and neutron radiation for research and radionuclide production. In addition to in-pool irradiation capabilities, the experimental facilities include two pneumatic transfer systems, a dry irradiation chamber, a thermal column, several beam tubes, and a hot-cell/gamma-ray facility. The effect of any experiment or sample on excess reactivity is limited by Sections 3.1 and 3.8 of the revised Technical Specifications.

10.1 Experimental Facilities

10.1.1 Pool Irradiation

The open pool of the reactor permits bulk irradiations and provides storage space for irradiated fuel and activated equipment. The decision to perform experiments in the reactor pool is dictated by specimen size and the type and intensity of radiation fields required. The actual placement of experiments or samples in the core region is controlled by their effect on excess reactivity.

10.1.2 Pneumatic Transfer Systems

Two 2-in. pneumatic transfer tubes are provided for the rapid transport of samples to and from the face of the reactor core. These sample holders 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. These systems have individual automatic timing controls and shielded containers for receiving the irradiated specimens.

10.1.3 Dry Irradiation Chamber

The dry irradiation chamber has a column of about 2.1 by 2.1 by 2.1 m. Access is obtained by removing a 2-m-thick high-density concrete door that is manually operated on rails. This chamber is separated from the reactor core by a voidable nosepiece used in conjunction with a 0.25-m-thick lead shutter to reduce radiation levels in the chamber when entry is required.

During reactor operations, both high neutron and gamma-ray fluxes are available. When the reactor is shut down only a gamma-ray field is available.

10.1.4 Thermal Column

A horizontal thermal column occupies the west face of the reactor core. Immediately adjacent to the core is a space for a 0.15-m-thick lead shield which can be used on the thermal column side and which causes a small increase in neutron reflection and attenuates gamma radiation. Between this shield space and the reactor tank wall is a graphite-filled aluminum can that is 0.69 m thick. Beyond the tank wall in place of the usual high-density concrete shield is a 1.2- by 1.2- by 1.5-m cavity stacked with graphite stringers. The working chamber is 1.2 by 1.2 by 0.61 m deep.

This facility provides a source of a relatively high ratio of thermal-to-fast neutrons. 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. This controls the release of activated gases, primarily ^{41}Ar.

A fission plate is available for use in selected experiments in the thermal column. It is made of uranium-aluminum alloy clad with aluminum, and measures 0.33 m by 0.33 m by 1.3 cm. The Technical Specifications limit the use of the fission plate to the outer face of the thermal column. The thermal power developed in the plate is approximately 1 W at the maximum neutron flux density in this position.

10.1.5 Beam Tubes

Five round beam tubes, 6 in. in diameter, radiate horizontally outward from the reactor core around the lower tank section and extend through the shield wall. The beam tubes can be used as dry irradiation chambers for samples placed within them at the face of the core or as neutron paths for irradiation of samples near the ports on the outside of the biological shield. The beam tubes also can be filled with demineralized water to provide neutron attenuation and to eliminate voids near the reactor core. They are normally filled with water when not in use. When air filled, the beam tubes are vented continuously to prevent buildup of ^{41}Ar by the same filtered ventilation system as that for the thermal column and the dry irradiation chamber.

The basic tube assembly consists of an embedded aluminum sleeve, a retractable aluminum liner, and a set of interior shielding plugs made of canned borated barite concrete and lead. When the beam tubes are used, external shield walls or beam catchers can be installed to control radiation levels in the experimental work areas.

10.1.6 Hot Cell

The hot cell is adjacent to the west wall of the reactor tank. Its construction is integral with that of the biological shield, with 0.91 m of concrete between the pool liner and the cell. It is connected to the upper portion of the reactor pool by a water-tight passthrough for sample transfer. The passthrough itself may be drained by a suitable arrangement of interlocked valves, permitting the introduction or removal of a sample from the reactor tank. The other walls of the cell are also of 0.91 m-thick high-density concrete; a 0.91 m-thick lead glass window is used for viewing. Samples are moved by using remote manipulators. A remotely operated 1-ton travelling crane inside the hot cell services this facility. Access to the cell is either by a 1.2 m by 2.1 m stepped access door or by a 0.91 m stepped plug in the roof of the cell. The hot-cell source strength capacity is limited to the equivalent of 25 kCi of ^{60}Co.

10.1.7 Gamma Facility

A 0.61 by 0.61 m cavity that is open to the hot cell but not to the reactor pool is in the concrete shield between the hot cell and the reactor tank. Irradiated fuel assemblies placed along the wall in special holders in the reactor tank can provide a source of gamma radiation for experiments in the hot cell. A total of eight irradiated elements can be positioned in a row to supply the source. A 4-in.-thick lead shield can be lowered by a winch into the cavity for shielding purposes.

10.1.8 Isotope Production Facilities, Standpipes, and Thimbles

Several fuel assembly positions may be filled by vertical tubes. Sets of tubes known as "isotope facilities," located near the center of the core to maximize its neutron flux density, terminate below the pool surface to maintain water shielding while irradiation samples are manipulated.

Other tubes extend above the surface of the pool and provide dry chambers in the core or reflector accessible from their tops. These tubes allow flexibility because experimental samples can be inserted and withdrawn easily and electrical leads are not obstructed.

10.2 Experimental Review

A Nuclear Safety Committee is established to provide an independent review of changes in operating procedures and all new experiments affecting reactor operation. This committee is composed of six members collectively having broad expertise in reactor-related technology. This committee is discussed in Section 6 of the revised Technical Specifications, and in Chapter II, Volume II of the SUNYAB Radiation Protection Manual.

All experiments involving reactor operation are reviewed by the NSTF Operating Committee, which is currently comprised of the Director, the Operations Manager, and the Radiation Protection Department Manager. Each request for an experiment must be approved separately on an Irradiation Service Request form before the experiment is started. Irradiations that are routine and present no significant safety considerations will be approved by the Operating Committee. Nonroutine requests are submitted to the Nuclear Safety Committee.

In addition to ensuring safe reactor use in compliance with the license, 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 unlikely to fail, (2) are unlikely to release significant radioactivity to the environment, and (3) are unlikely 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, released as liquid to the SUNYAB sanitary sewer system, or packaged as solids and transferred to the Radiation Protection Department for disposal, all in accordance with applicable regulations.

Further, the administration and the staff of the NSTF follow closely the principles of the ALARA concept in handling radioactive materials and in considering their release to the unrestricted environment.

11.1 Waste Generation and Handling Procedures

11.1.1 Airborne Waste

The potential airborne waste includes gaseous ^{16}N, ^{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 water and the air dissolved in the pool water and of the air and airborne particulates in the thermal column, dry irradiation chamber, pneumatic transfer system, and beam tubes.

Exposure of occupational personnel is limited by constantly sweeping the air from the reactor room and experimental area. A separate ventilation system is provided for the thermal column, dry irradiation chamber, pneumatic transfer system, and beam tubes. The thermal column, dry irradiation chamber, and beam tube discharge stream have an exhaust fan separate from the pneumatic transfer system discharge stream exhaust fan. During operation the exhausted air is passed through HEPA filters to remove most of the airborne particulates and then monitored for radioactivity. Data provided by the licensee demonstrate that airborne radioactive ^{41}Ar released to unrestricted areas in recent years has never exceeded 10% of 10 CFR 20 guidelines, when averaged over a year. Both the licensee's and the staff's computations indicate that potential annual dose rates from ^{41}Ar are also well within 10 CFR 20 guidelines for unrestricted areas.

11.1.1.1 Loss of Experimental Facility Ventilation

If the ventilation system fails, radioactive gases from the beam tubes, the thermal column, and the dry chamber could escape into the reactor room air. The major radiation source is ^{41}Ar resulting from the ^{40}Ar (η,γ) ^{41}Ar reaction.

In the analysis of the effect of loss of beam tube ventilation, the licensee has assumed that

(1) the experimental facility ventilation system is not operating

(2) a beam port cover is removed

(3) a thermal neutron flux density of 1×10^{13} $n/cm^2 \cdot sec$ at the core end of the 6-in. beam tubes

(4) all but 50 cm of the tubes is solidly plugged

(5) the ^{41}Ar activity had reached the saturation level

(6) complete instantaneous mixing of the beam tube air with the reactor room air

The staff has calculated independently the ^{41}Ar concentrations using the above assumptions and current nuclear data. The average reactor room initial concentration of ^{41}Ar resulting from opening one 6-in. beam tube was calculated to be 3.8×10^{-11} Ci/ml based on a reactor room volume of 6.3×10^{9} ml. This value is not significantly different from the licensee's value of 5×10^{-11} Ci/ml. The staff considers the probability of inadvertently and simultaneously opening more than one beam port to be negligible and, therefore, not a credible event.

The computed initial concentration is above that of Table I of Appendix B of 10 CFR 20 for occupational exposure (2×10^{-12} Ci/ml). The potential employee exposure dose, neglecting radioactive decay and building air exchange, is less than 40 mrem. The total loss of beam tube ventilation because of the series fan arrangement would require failure of both the fixed experimental facility exhaust fan and stack exhaust fans. Even if this unlikely set of failures occurs and the ^{41}Ar content of a beam port is released to the reactor room, the building air exhaust system would remove the ^{41}Ar rapidly because of 3 to 4 building air exchanges per hour. In addition to these facility features that limit the potential consequences of the event, the reactor is equipped with a visual alarm that indicates failure of the stack exhaust fans, and administrative procedures prohibit opening a beam port unless the reactor is in a shutdown condition and the beam port is monitored during opening.

When the large dry chamber is in use the chamber nosepiece is voided and the ^{41}Ar production rate is approximately 1.25 $mCi \cdot min^{-1}$. The chamber is ventilated by the fixed experimental facility fan, and no significant ^{41}Ar accumulation is expected. The nosepiece is flooded when the dry chamber is not in use and the ^{41}Ar production rate is substantially lower. Complete loss of dry-chamber ventilation requires failure of the fixed experimental facility fan and the exhaust system. The dry-chamber door is provided with a reactor scram in addition to the administrative prohibition against opening the door when the reactor is operating. The visual alarm on the stack exhaust fans provides an indication of potential loss of dry-chamber ventilation, allowing the operator to take appropriate action.

Regarding the loss of ventilation event, the revised Technical Specifications require that the stack exhaust fan in the SUNYAB steam plant and the building air exhaust must be operable if the reactor is operating.

The ^{41}Ar concentration in the reactor room resulting from the opening of the thermal column door also was calculated by the staff. The staff assumed (1) a thermal neutron flux density of 1.5×10^{7} $n/cm^2 \cdot sec$ at the outer face of the graphite (licensee's measured value), (2) an experimental air space of 1.22 m by 1.22 m by 0.61 m, (3) that the ^{41}Ar had reached saturation activity, and (4) complete mixing with the reactor room air.

The instantaneous calculated ^{41}Ar concentration is 2.3 x 10^{-15} Ci/ml, which is approximately a factor of 800 below that of Table I of Appendix B of 10 CFR 20. The consequences of this event are, therefore, insignificant.

The forced convective coolant flow down through the core to the holdup tank and heat exchanger precludes the release of ^{16}N from the pool water into the reactor room air. This isotope ($T_{1/2}$ = 7 sec) has essentially decayed within the piping system by the time the water returns to the open pool, which is approximately 6 min after it exits the core.

11.1.2 Liquid Waste

Several activities conducted within the reactor facility are capable of generating radioactive liquid waste. The largest volume of contaminated water from the reactor systems is produced by the regeneration of the demineralizer and from pump cooling.

All potentially radioactive liquid waste from the hot cell and from the hot laboratories is collected in two 250-gal stainless-steel tanks buried underground. Lower level radioactive waste, or liquid that could contain a small amount of radioactivity, is collected in two 600-gal stainless-steel tanks, also buried underground. If the activity in the liquid of either the hot-or the low-level system is low enough, the liquid is pumped to a 10,000-gal mild-steel tank. All three of the tank systems can be stirred to provide a representative sample for analysis. If activity is low enough in the 10,000-gal tank to be in compliance with applicable state and Federal regulations for release of radioactive materials, the liquid is pumped to the SUNYAB sanitary sewer system. As necessary, dilution of the stored waste with the sewer liquid with which it combines is calculated so that only enough is released to be within applicable limits. A total of 1 Ci per year is the maximum acceptable activity to be released, according to regulations as well as to the revised Technical Specifications.

This system, which is operated on a collect-hold-sample-analyze-release philosophy, provides a positive method of preventing accidental discharge of radioactive liquids to the unrestricted environment.

11.1.3 Solid Waste

Low-level solid waste generated as a result of reactor operations consists primarily of ion exchange resins, filters, potentially contaminated paper and gloves, and occasional small, activated components. These are packaged in accordance with applicable NRC and U.S. Department of Transportation (DOT) regulations and are transferred to the Radiation Protection Department for disposal in accordance with applicable regulations.

High-level solid radioactive material generated by routine reactor operations consists of 20 to 24 spent fuel assemblies about every 12 years. Spent assemblies are stored in the reactor pool until the accumulation justifies shipment for reprocessing.

11.2 Conclusion

The staff concludes that the waste management activities of the NSTF have been conducted and can be expected to continue to be conducted in a manner consistent both with 10 CFR 20 and with ALARA principles. Among other guidance, the staff review has followed the methods of ANSI/ANS 15.11, 1977, "Radiological Control at Research Reactor Facilities."

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 operational releases. The staff concludes that the maximum potential doses in unrestricted areas as a result of actual releases of ^{41}Ar have never exceeded or even approached the limits specified in 10 CFR 20 for unrestricted areas when averaged over a year. Furthermore, the staff's conservative computations of the exposure rates beyond the limits of the reactor facility give reasonable assurance that potential doses to the public as a result of ^{41}Ar would not be significant, even if there were a major change in the operating schedule of the reactor.

12 RADIATION PROTECTION PROGRAM

The Radiation Protection Department at SUNYAB 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 all university facilities. The NSTF and its reactor are provided with health physics personnel from the Radiation Protection Department. The NSTF monitors both liquid and airborne effluents at the points of release to comply with applicable regulations. The Radiation Protection Department also has an environmental monitoring procedure to verify that radiation exposures in the unrestricted areas surrounding the facility are well within regulations and guidelines.

12.1 ALARA Commitment

The SUNYAB Radiation Protection Department has implemented the policy that all operations are to be conducted in a manner to keep all radiation exposures as low as reasonably achievable (ALARA). This policy is implemented by a set of specific guidelines and procedures. 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 current university health physics staff consists of at least four professionals plus several part-time student assistants. The current health physics staff at the NSTF reactor is one professional with additional support as needed. 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, the authority, and adequate lines of communication to provide an effective radiation safety program. The radiation safety organization is shown in Figure 12.1.

The Health Physics staff provides radiation safety support to the entire NSTF research complex, including an accelerator. However, the staff has determined that the university health physics staff is adequate for the proper support of all research efforts within this facility.

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 NSTF. 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 contained in the SUNYAB radiation protection manuals are readily available to the operational and research staffs and to the health physics and administrative personnel.

12.2.3 Instrumentation

The NSTF has a variety of detecting and measuring instruments available for monitoring potentially hazardous radiation. The instrument 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 annually. This program is described in the approved reactor operator requalification program.

12.3 Radiation Sources

12.3.1 Reactor

Sources of radiation directly related to reactor operations include radiation from the reactor core, ion exchange columns, and radioactive gases (primarily ^{41}Ar and small quantities of fission products from tramp uranium).

The reactor fuel is contained in Zircaloy-2 cladding. Radiation exposures from the reactor core are reduced to acceptable levels by water and concrete shielding.

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 are associated with reactor use include radioactive isotopes produced for research, activated components of experiments, and activated samples or specimens.

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

12.4 Routine Monitoring

12.4.1 Fixed-Position Monitors

The NSTF uses several fixed-position radiation monitors. Area radiation monitors are placed at strategic locations throughout the building in regions where radiation levels might increase and reflect an abnormality or hazard in operations. These include three area monitors on the neutron deck, a monitor on the bridge above the reactor, and a monitor in the hot cell. The area monitor in the hot cell is interlocked with the access door to the cell.

When the reactor is operating, additional monitors are required by the Technical Specifications. These include a building air continuous monitor located in the fan room, a stack gaseous continuous monitor, a stack particulate continuous monitor, and a primary water monitor. The two stack continuous monitors sample from a side stream.

All fixed-position monitors have adjustable alarm set points and read out in the control room.

12.4.2 Experimental Support

The health physics staff participates in experiment planning 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, operating procedures require that changes in experimental setups include a survey by health physics personnel using portable instrumentation, and all items removed from the containment must be surveyed and tagged if radioactive. Low-level activity items can be surveyed and tagged by reactor personnel and experimenters.

12.4.3 Nonroutine Tasks

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 a detailed staff review. The work is then performed with health physics coverage.

12.5 Occupational Radiation Exposures

12.5.1 Personnel Monitoring Program

The NSTF reactor personnel monitoring program is described in a SUNYAB radiation protection manual. To summarize the program, personnel exposures are measured by the use of film badges assigned to individuals who might be exposed to radiation. In addition, TLDs and non-self-reading pocket ion chambers are used, 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 the limits specified in 10 CFR 20.

12.5.2 Personnel Exposures

The NSTF reactor personnel annual exposure history for the last several years is given in Table 12.1.

12.6 Effluent Monitoring

12.6.1 Airborne Effluents

As discussed in Section 11, airborne radioactive effluents from the reactor facility consist principally of activated gases. In the normal operating mode, the two gaseous effluent streams are filtered to remove most particulate materials before discharge to the environment through the steam plant stack and through the building roof exhaust.

The two airborne effluent streams are continuously monitored to provide prompt indication of any abnormal concentrations being discharged by withdrawing a representative side stream from the main discharge duct, passing this through particulate and gaseous monitors, and returning it to the main discharge stream. The particulate monitor used on the stack exhaust is a fixed paper filter in front of an end-window Geiger-Mueller detector. The gas monitors for both the stack and the building air roof exhausts are shielded volumes, each containing two Geiger-Mueller tubes. The two monitors for the stack exhaust, located in the mechanical equipment room on the neutron deck level, draw the samples from the discharge duct immediately before the blower on the stack. The gas monitor for the building air, located in the fan room on the control deck level, draws the sample just downstream of the HEPA filter. The output of these monitors is indicated on meters having adjustable alarm set-points. These outputs are repeated on chart recorders and meters in the reactor control room. In addition, each monitor has a remote alarm, sounding in the control room.

12.6.2 Liquid Effluent

The reactor generates very limited radioactive liquid waste during routine operations. However, leaks in the primary coolant system do not have the potential for being released, and experimental activities associated with reactor usage also generate radioactive liquids. All of the latter potentially radioactive liquids are collected in two 250-gal stainless-steel holdup tanks or in two 600-gal stainless-steel tanks. These tanks are pumped to the 10,000-gal tank, the contents of which can be recirculated to obtain a representative sample, sampled and analyzed for the amount of radioactivity. Taking into account the dilution of the NSTF effluent stream by the campus effluent stream, the level of contained radioactivity is maintained below the levels specified in 10 CFR 20.303. The SUNYAB campus sanitary sewer system is in turn diluted by the campus storm drain system, for which no credit is taken in the NSTF calculations. The campus system is finally released into the Buffalo sewer system, and it also is diluted by storm drains.

12.7 Environmental Monitoring

The current environmental monitoring procedure conducted by the Radiation Protection Department is directed toward measuring direct radiation and toward detecting leaks resulting from the liquid waste storage tank system.

The potential direct radiation is measured by film badges located near the outside shipping area and on the side of the cooling tower. Infrequent positive readings that can be attributed to the twice weekly shipment of radioisotopes have been measured by the badge in the shipping area. No significant radiation levels above background are found near the cooling tower.

No significant activity has ever been found in the groundwater samples from the liquid waste storage tank areas.

12.8 Potential Dose Assessments

Natural background radiation levels in the Buffalo area result in an exposure of about 120 mrems/yr to each individual residing there. At least an additional 7% (approximately 8-9 mrems/yr) will be received by those living in a brick or masonry structure. Any medical diagnosis and x-ray examinations will add to these natural background radiations, increasing the total accumulative annual exposure.

The measured ^{41}Ar annual releases from the NSTF reactor have varied in recent years at between 110 Ci and <500 Ci. Conservative calculations by the NRC staff based on this higher value (a 500 Ci annual ^{41}Ar release) predict a maximum exposure of less than 10 mrem/yr to an individual in the unrestricted area.

12.9 Conclusion

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

The staff also concludes that the effluent monitoring programs and the environmental monitoring procedure conducted by Radiation Protection Department personnel are adequate to promptly identify significant releases of radioactivity and to predict maximum exposures to individuals in the unrestricted area. These predicted maximum levels are a small fraction of applicable regulations and guidelines specified in 10 CFR 20.

Additionally, the staff concludes that the NSTF reactor radiation protection program is acceptable because the staff has found no instances of reactor-related exposures of personnel above applicable regulations and no unidentified significant releases of radioactivity to the environment. Furthermore, the staff considers that there is reasonable assurance that the personnel and procedures will continue to protect the health and safety of the public from significant radiation exposures related to routine operations.

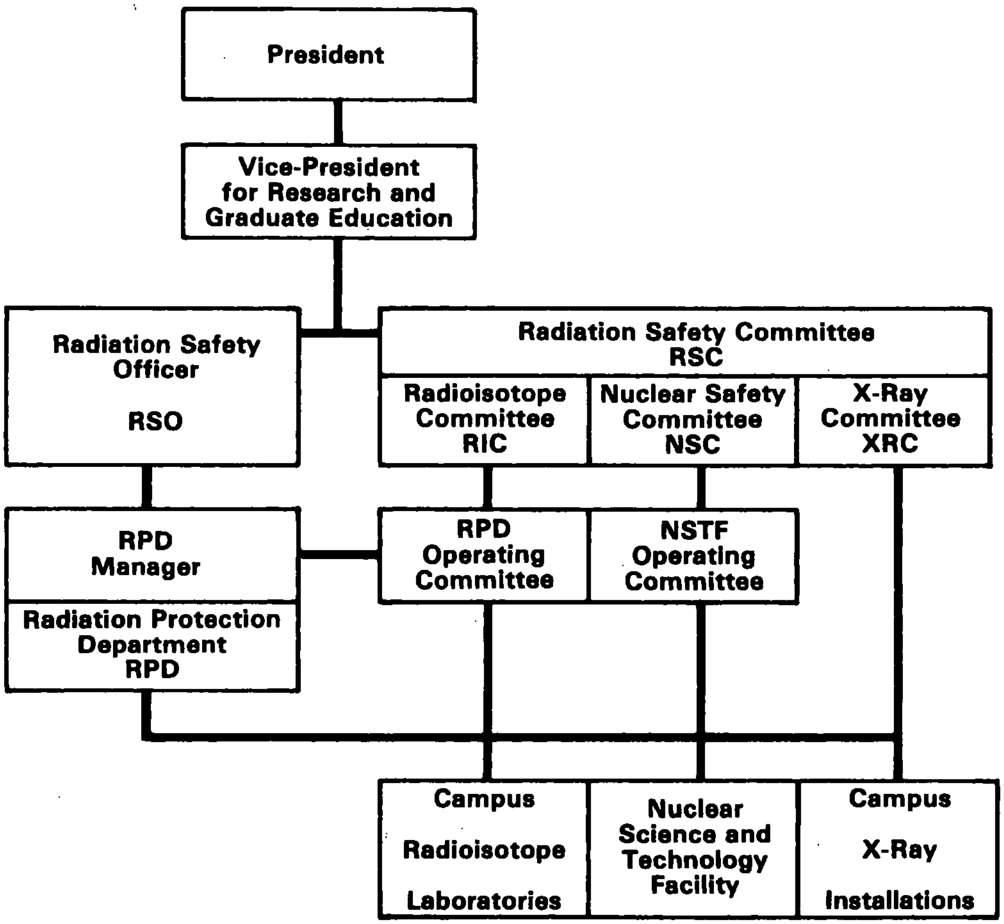

Figure 12.1 Administrative chain of command and responsibility for radiation safety

Table 12.1 Number of individuals in exposure interval

	Number of individuals in each range				
Whole-body exposure range (rems)	1978*	1979**	1980	1981	1982
No measurable exposure	168	65	10	5	1
Measurable exposure:					
< 0.1	128	56	5	4	2
0.1 to 0.25	5	4	8	5	3
0.25 to 0.5	6	7	4	3	7
0.5 to 0.75	3	0	1	2	3
0.75 to 1.0	3	1	1	1	1
1.0 to 2.0	3	0	1	0	0
Over 2.0	2	0	0	0	0
Number of individuals monitored	*318**	*133***	*30*	*20*	*17*

*In 1978, in addition to Special Projects and Public Safety personnel these figures include students in classes utilizing the counting room. Also, during 1978, the reactor was refueled, and many refurbishing and maintenance activities were performed.

**In 1979, Special Projects personnel (~15 individuals) and Public Safety (Night watchmen) personnel (~75 individuals) are included.

13 CONDUCT OF OPERATIONS

13.1 Overall Organization

Responsibility for the safe operation of the reactor facility is vested in the chain of command shown in Figure 13.1.

13.2 Training

The operators and senior operators for the reactor are trained in-house by the facility staff. The licensee's Operator Requalification Program has been reviewed, and the NRC staff has concluded that it meets applicable regulations (10 CFR 50.34b).

13.3 Emergency Planning

10 CFR 50.54(q) and (r) require that a licensee authorized to possess and/or operate a research reactor shall follow and maintain in effect an emergency plan that meets the requirements of Appendix E to 10 CFR 50. In 1979 the guidance available to licensees was contained in Regulatory Guide 2.6 (1978 For Comment Issue) and in ANS 15.16 (1978 Draft). In 1980, new regulations were promulgated, and licensees were advised that revised guidance would be forthcoming. Thus, revised ANS 15.16 (November 29, 1981 Draft) and Regulatory Guide 2.6 (March 1982 For Comment) were issued. On May 6, 1982, an amendment to 10 CFR 50.54 was published in the Federal Register (47 FR 19512, May 6, 1982) recommending these guides to licensees and establishing new submittal dates for Emergency Plans from all research reactor licensees. The deadline for submittal from a licensee in the NSTF class ($\geq$2 MW) was September 7, 1982. The licensee made a timely transmittal of an Emergency Plan, thereby complying with existing applicable regulations.

13.4 Operational Review and Audits

In addition to the line personnel responsible for reactor operations and for radiological protection, a Nuclear Safety Committee (NSC), reporting to the Vice President for Research and Graduate Education, reviews and oversees facility operation. The NSC consists of a minimum of six persons with expertise in the physical sciences and preferably some nuclear experience. Permanent members of the NSC are the NSTF Director, the Campus Radiation Safety Officer, the Radiation Protection Department Manager, and the Operations Department Manager. A quorum of this committee must have at least 5 members present, a majority of whom shall not be direct members of the NSTF line staff. There also is an Operating Committee that is a subgroup of the NSC. The operating committee consists of the NSTF Director, the Radiation Protection Department Manager, and the Operations Department Manager. The Operating Committee reviews and approves routine activities. Experiments that might involve significant hazards considerations, proposed amendments to the license, and reportable occurrences are reviewed and subject to approval of the NSC. The university engages an independent consultant to perform an annual audit of the NSTF operations and to submit a report.

13.5 Physical Security Plan

The NSTF has established and maintains a program designed to protect the reactor and its fuel and to ensure its security. The staff has visited the site and has reviewed the Physical Security Plan to compare it with the requirements of 10 CFR 73.67 for special nuclear material of low strategic significance. The staff has concluded that the licensee's Physical Security Plan, as amended, submitted by letter dated March 11, 1983 meets the requirements of the regulations and has been incorporated as a condition of the operating license.

Both the Physical Security Plan and the staff's evaluation are withheld from public disclosure under 10 CFR 2.790(d)(1) and 10 CFR 9.5(a)(4).

13.6 Conclusions

Based on the above considerations, the staff concludes that the licensee has sufficient experience, management structure, and procedures to provide reasonable assurance that the reactor will be managed in a way that will cause no significant risk to the environment or to the health and safety of the public.

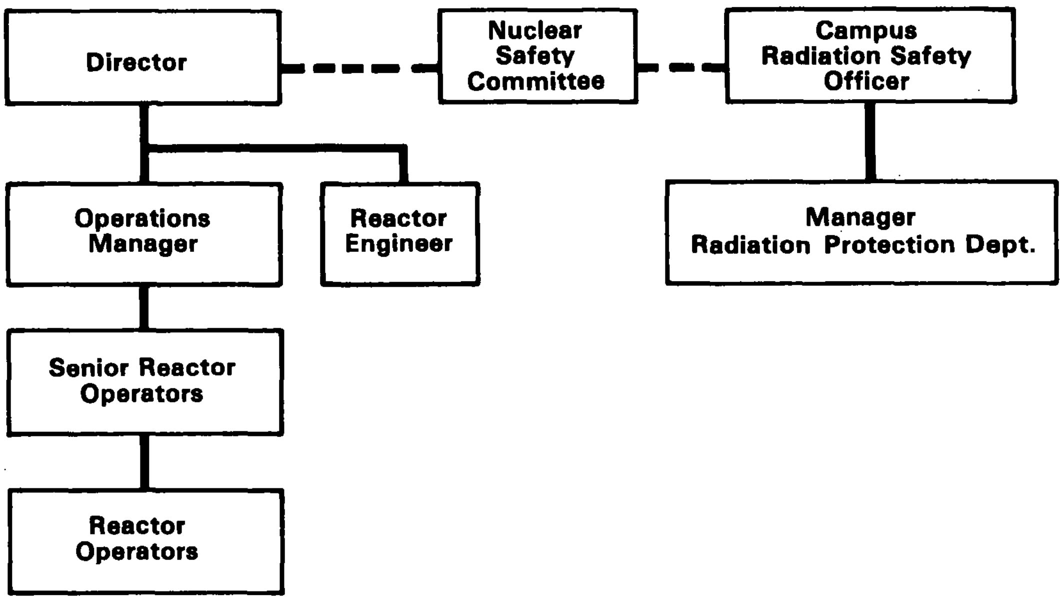

Figure 13.1 Overall organization

14 ACCIDENT ANALYSIS

To establish limiting safety system settings and the limiting conditions for operation for the NSTF reactor, the licensee analyzed potential reactor transients (Western New York (WNY) Nuclear Research Center, SAR Rev. II, 1963; NSTF SAR update, 1981). Other hypothetical accidents and their potential effects on the core and the health and safety of the public also were analyzed. In addition, the licensee analyzed potential effects of natural hazards on the reactor.

Among the accidents postulated, the one with the greatest potential effect on the environment in the unrestricted area is partial meltdown of fuel assemblies as a result of a fuel-handling accident. Occurrence of the accident would require violation of administrative procedures and of the revised Technical Specifications. This accident is designated by the licensee as the design-basis accident (DBA). A DBA is defined as a postulated accident for which the potential risk to the public is greater than that from any credible accident and for which the engineered safety systems are intended to mitigate the effects of the accident. The DBA does not have to be credible; the staff assumes that the accident occurs but does not try to describe or evaluate the mechanistic details of the accident or the probability of its occurrence. Only the hypothetical consequences are evaluated.

Several postulated transients and accidents have been evaluated (rapid insertion of reactivity, loss of coolant, loss of coolant flow, maximum startup, fuel cladding rupture, and experimental facility accidents) and are discussed in the following sections.

14.1 Rapid Insertion of Reactivity

The licensee has examined a hypothetical accident that has been called both a design-basis accident and a maximum credible accident in different parts of the documentation on the NSTF reactor. Because activation of the reactor room ventilation system is called on to mitigate the potential consequences, the staff will also refer to it as a DBA. This hypothetical accident results from an accidental rapid insertion of a fuel assembly into an already critical core. Such an event would violate both the Technical Specifications and related administrative procedures. The licensee has made the following assumptions for the analysis of the accident:

(1) The core has been loaded in an optimum configuration in which a water-filled fuel position has an unusually high worth when occupied by fuel.

(2) The reactor is critical.

(3) A fuel assembly is dropped from above directly into the water-filled high-worth fuel position.

(4) The high level power trip drops the control-safety blades.

The assumed core configuration is a U shape, in which the fuel assembly in a corner of the base is calculated to have a worth of 3.8% $\Delta k/k$. The licensee further postulates that the insertion of excess reactivity is sufficient to cause a self-terminating fast transient producing 180 MW·sec of energy. This is consistent with an extrapolation of data taken during the pulse-testing program at NSTF in 1964. The licensee's results also indicate that a production of 180 MW·sec total energy in a transient could lead to melting of the UO_2 in the hottest regions of the core where energy densities are 2.3 times the average. The licensee has calculated that 2% of the fuel would melt, but in order to be conservative in the analysis, he has assumed that 5% melted. It was also assumed that cladding integrity was lost during the event and that 100% of the noble gases, 50% of the halogens, and 1% of the nonvolatile fission products in the molten fuel were released. Further, it was assumed that the transient event caused the simultaneous release of 50% of the inventory of fission products accumulated in the gap between cladding and fuel meat in the entire core.

The licensee considered the scenario in which it is postulated that all of the fission products itemized above are released from the pool and uniformly dispersed in the reactor room air volume, with no reduction (as a result of plate-out or washout) in the amounts, including the halogens. The licensee's scenario also assumed that some (3.5%) of the contaminated air was exhausted in a 5 min "puff" from the reactor room through the air cleanup system. Subsequently, the room air was confined and released through the emergency cleanup system at a rate of 5% per day. The licensee evaluated the potential doses and body-burdens accumulated by inhalation at various locations and under various assumed conditions. It was found that the largest potential doses in unrestricted areas were caused by ^{89}Sr, ^{90}Sr, and ^{131}I, so the licensee considered only these radionuclides in the detailed evaluation of offsite radiation exposures. For both the initial 3.5% "puff" and the slower, controlled release of room air, the potential accumulated body burdens in unrestricted areas were found to fall well within 10 CFR 20 guidelines for all three of the radionuclides considered.

To ensure that the computations were conservative, very unfavorable (and unlikely) meteorological conditions were assumed, namely very unstable (inversion) conditions with a wind speed of 1 m per second. It also was assumed that the initial 3.5% puff of air was released without being cleaned in passing through the filter system that is in the air-exhaust path. An additional computation was performed for which it was assumed that the air cleanup system functioned normally.

For occupational personnel within the reactor/control room area at the time of the postulated accident, potential whole-body dose was computed, making reasonable assumptions for stay-time in the room. The computed dose was less than 2 rems. The licensee also computed hypothetical whole-body dose in the unrestricted areas resulting from direct exposure to the gamma rays shining out from the source contained in the air in the reactor room. The initial exposure rate at the nearest point off campus was estimated to be approximately 1 mrem per hour. Both radioactive decay and cleanup by the filter systems would lead to a continuous decrease of this exposure rate with time.

The staff and its contractor have reviewed the licensee's assumptions, methods of computations, and results related to the design-basis accident. The staff accepts that designation, and also notes that many of the licensee's assumptions lead to computed potential doses and body burdens that are much larger than could be reasonably expected. Furthermore, the initial assumption of a 3.8% $\Delta k/k$ reactivity insertion is based on a calculated value that is higher than any fuel assembly worth ever measured at the NSTF. Hence, it is unlikely that the initial conditions for this event could be achieved. Among the assumptions that are very likely to yield computed doses higher than those that would realistically occur are the following:

(1) Unfavorable atmospheric conditions were assumed (inversion, low wind speed).

(2) No credit was taken for dissolution, chemical combination, washout or plateout of radionuclides in the pool or reactor building.

(3) No decrease in source strength resulting from radioactive decay was assumed for the puff release.

(4) Low efficiencies were assumed for the charcoal trap and particulate filters.

(5) It was assumed that the exposed individual remain at the point of maximum offsite concentration continuously for 1 to 10 days.

On the basis of the above considerations, the staff concludes that the DBA for the NSTF reactor, while not a credible accident, does demonstrate that the NSTF reactor's engineered safety systems would limit the consequences of the DBA so that its occurrence would not result in undue risk to the health and safety of the general public. The analysis demonstrated that the only significant radiation doses from the DBA would result from inhalation of ^{131}I, ^{89}Sr, and ^{90}Sr and that the probable quantities of these radionuclides inhaled during the puff or in 1-to-10-days continuous exposure by individuals in unrestricted areas would be orders of magnitude less than those forming the basis for 10 CFR 20, Appendix B.

14.2 Loss of Coolant

The licensee considers that the rapid loss of all pool water is the worst credible accident (NSTF SAR update, 1981). The licensee's analysis of the accident is presented in the WNY Research Center's SAR (Rev. II, 1963) and updated in the licensee's SAR update, responses to staff's questions, (March 10, 1983) and Technical Specifications. The analysis assumes complete instantaneous loss of pool water, thereby uncovering the core while a significant amount of power is still being generated in the fuel pins through fission product decay heat. It was further assumed that the reactor had been operating at 2 MW for a long period of time and, therefore, that the fission product inventory had attained equilibrium.

The licensee calculated the maximum surface temperature of the fuel cladding based on experiments at the low intensity training reactor (LITR) in which the loss of water was used as a shutdown mechanism. The LITR fuel temperature data measured at 1.0 and 1.5 MW were extrapolated to 2 MW and then adjusted to reflect differences in hydraulic and thermodynamic characteristics. The resulting calculated maximum fuel pin temperature, not reached for at least 1 hour, is 707°C compared with the melting point of Zircaloy-2, which is approximately 1,815°C. The melting temperature of the UO_2 fuel is approximately 2,760°C. In response to staff questions (April 20, 1983), the licensee also reevaluated the loss of pool water through a ruptured beam port, basing the analysis on methods developed for the research reactor at the National Bureau of Standards. This conservative analysis confirmed the previous computations that fuel temperatures would not reach the melting points of either UO_2 or zircaloy. Therefore, the licensee concludes that no fuel or cladding melting will occur as a result of a gross pool water loss and that the major hazard would be from extremely high radiation levels above the pool. The licensee indicates the hazard could be short lived because the manually operated emergency-pool-fill system can be activated in response to either the low level pool water alarm or low level pool water scram. As a last resort the core could be recovered by flooding the lower level of the containment vessel.

The staff has reviewed the licensee's analysis, and concurs with the assumptions and methods. Therefore, the staff concludes that there is reasonable assurance that rapid loss of all pool water would not cause fuel cladding failure that could cause a risk of release of radioactivity from the NSTF reactor fuel.

14.3 Loss-of-Coolant-Flow Accident

In the event that the forced convective coolant flow stops while the reactor is operating, the low-flow condition would scram the reactor, and gravity would open the flapper valve on the plenum below the core, allowing natural convective cooling of the fuel. Approximately 1 sec after the onset of loss of flow the scram signal is activated, and flow continues to decrease until the flapper valve opens at 0 flow, approximately 10 sec later.

The licensee analyzed the loss-of-flow accident assuming that the flapper valve fails to open so that no natural convection is available. It was further assumed that (1) the reactor had been operated continuously for about one calendar quarter at 2 MW, (2) upon scram the control blade of highest worth was full out and failed to insert, (3) the net available reactivity for shutdown was 3% $\Delta k/k$, (4) the power remains at 2 MW for 2 sec after scram, and (5) the reactor power decays with an 80-sec period. The heat release because of fission product decay and neutron power was calculated using methods of Shure (1961) and Lamarsh (1966), respectively. The calculations were carried out for the total core and per assembly for a 16-assembly core for time periods up to 600 sec after scram. The total heat developed per assembly in the first 600 sec was calculated, and it was assumed that the hottest assembly led the core average by a factor of 2.5. It was further assumed that no heat would be transferred from the fuel into the ambient water and that the specific heat of UO_2 remains constant rather than increasing with temperature. On the basis of these assumptions, the licensee obtained a maximum fuel temperature of ~560°C and an average fuel temperature of ~494°C in the hottest assembly of the 16-assembly core. Because the melting points of UO_2 and Zircaloy-2 are approximately 2,760°C and 1,815°C, respectively, the licensee concludes that no loss of fuel or cladding integrity is anticipated in a loss-of-flow accident in which the flapper valve fails to open.

The staff has reviewed the licensee's assumptions and the methods used in the analysis and has found that the analysis is appropriate for the NSTF reactor and that the assumptions lead to an ultimate fuel/cladding temperature that is unrealistically high. For example, it is assumed that even though coolant is not lost and the fuel is totally immersed, there is no heat transfer to the water. On the basis of these considerations, the staff concludes that a loss-of-coolant flow will not lead to fuel temperatures that would cause damage to the fuel or the cladding.

14.4 Maximum Startup Accident

In this accident analysis the licensee has assumed that all control safety blades (five) and the control blade are withdrawn simultaneously because of circuit malfunction. It is further assumed that (1) no protective action is taken until the high power scram is automatically tripped at 2.4 MW (120% of rated power), (2) the maximum blade withdrawal rate is 7.6 cm (3 in.) per minute, and (3) the blades are at their point of maximum differential worth, which gives the maximum reactivity insertion rate (0.038% $\Delta k/k$ per second). These assumptions make the analysis conservative. The licensee's detailed analysis is presented in Appendix C of the licensee's SAR update (1981). The minimum stable reactor period calculated on the basis of these assumptions is 47.5 msec. Two pulses performed during the pulse test program produced periods that bracketed this calculated value. The periods were 57.9 and 37.6 msec with energy releases of 2.5 MW·sec and 2.8 MW·sec and test pin maximum surface temperatures were measured to be 128°C and 133°C, respectively. The energy releases in these pulses were well below the 35 MW·sec energy release evaluated and previously authorized for routine pulsing. The licensee concludes that the energy released and temperatures associated with the maximum startup accident are modest and pose no hazard to NSTF personnel or the reactor core. The staff concurs with the licensee's evaluation and conclusion.

14.5 Fuel Cladding Rupture Accident

The possiblity has been considered that loss of integrity of the cladding on fuel pins might occur because of events other than the catastrophic reactivity excursion reviewed in Section 14.1 of this report. Possible causes of cladding failure are material fatigue, which could develop from initial defects, or waterlogging, which results from water leaking slowly into the fuel meat and being rapidly turned to steam during a reactivity transient. Even though the licensee intends not to pulse the reactor in the future, inadvertent transients are still possible, so these hypothetical events or accidents were evaluated. Furthermore, one waterlogging event did occur in 1971. Both the licensee and the staff evaluated this event at the time, and the staff's conclusions are given in D. J. Skovholt's letter to WNY Nuclear Research Center, Inc. (September 15, 1971) and Burger's Report (September 15, 1971). Those conclusions were that (1) the fuel-pin cladding failure was an isolated incident that was probably caused by a random pinhole defect in the cladding material, (2) established margins for operation in the pulse mode had not been exceeded, and (3) airborne radioactivity inside the containment building remained well within the guideline values of 10 CFR 20 for unrestricted areas.

The staff has reviewed both the licensee's analysis and the previous staff's analysis and evaluation and reconfirms those evaluations and conclusions.

Hence, the staff concludes that even though the inventory of fission products in the NSTF fuel is larger now than it was at the time of the previous fuel-cladding failure, there is reasonable assurance that a future failure would not lead to potential doses in the unrestricted environment that exceed the guidelines of 10 CFR 20, averaged over a year.

14.6 Experimental Facility Accidents

Experimental facility accidents may be considered in the context of the demonstrated pulse-mode operation of and the pulse-test program results on the PULSTAR reactor (WNY-017, 1964; WNY Summary Report, 1966). Reactivity changes in excess of 1.7% $\Delta k/k$ would be required to damage fuel pins. Therefore, the staff has required that the combined worth of movable and unsecured experiments that could result in a positive reactivity change because of a simultaneous removal of experiments be limited to 1.7% $\Delta k/k$. The analysis of potential transients associated with reactor experiments are discussed in Section 4.4 of this report.

The inadvertent misuse of an experimental facility releasing significant quantities of ^{41}Ar to the reactor room and to unrestricted areas is discussed in Section 11.1.1.1 of this report.

14.7 Conclusion

The staff has reviewed the credible accidents and transients from the NSTF reactor. On the basis of this review, the staff concludes that no credible accidents or transients are postulated that can result in the release of significant quantities of fission products to the unrestricted environment. The design-basis accident is extremely unlikely if not totally incredible. However, it does demonstrate the ability of the NSTF reactor safety systems to mitigate the consequences of an accident in which fission products are released to the environment to such an extent that the resultant doses to the public would be below the limits that form the basis of 10 CFR 20. Therefore, the staff concludes that the design of the facility together with the revised Technical Specifications provides reasonable assurance that the NSTF reactor can be operated at 2 MW without significant risk to the health and safety of the general public or the NSTF staff for the requested license period.

15 TECHNICAL SPECIFICATIONS

The licensee's Technical Specifications evaluated in this licensing action define certain features, characteristics, and conditions governing the continued operation of this facility. These Technical Specifications will be explicitly included in the renewal license 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 as guidance the ANSI/ANS 15.1-1982 consensus standard "The Development of Technical Specifications for Research Reactors."

On the basis of its review, the staff concludes that normal reactor 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, surveillance requirements, and engineered safety features will limit the likelihood of malfunctions and mitigate the consequences to the public and the environment of offnormal or accident events.

16 FINANCIAL QUALIFICATIONS

The NSTF reactor is owned and operated by a state university in support of its role in education and research. Therefore, the staff concludes that funds will be made available, as necessary, to support continued operations and eventually to shut down the facility and maintain it in a condition that would constitute no risk to the public. The licensee's financial status was reviewed and found to be acceptable in accordance with the requirements of 10 CFR 50.33(f).

17 OTHER LICENSE CONDITIONS

17.1 Prior Reactor Utilization

Previous sections of this SER concluded that normal operation of the reactor causes no significant risk of radiation exposure to the public, and that only an off-normal or accident event could cause some measurable exposure. Even a design-basis accident would not lead to a dose to the most exposed individual greater than applicable guideline values of 10 CFR 20.

The staff concluded in earlier safety evaluations that the reactor was initially designed and constructed and modified in 1963/64 in such a way to operate safely, with additional engineered safety features, and also considered whether operation would cause significant degradation in the capability of components and systems to perform their safety function. Since fuel cladding is the component most responsible for preventing release of fission products to the environment, possible mechanisms that could lead to detrimental changes in integrity were considered. Prominent among the considerations were the following: (1) radiation degradation of cladding integrity, (2) high fuel temperature or temperature cycling leading to changes in the mechanical properties of the cladding, (3) corrosion or erosion of the cladding leading to thinning or other weakening, (4) mechanical damage resulting from handling or experimental use, and (5) degradation of safety components or systems.

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

(1) Very similar fuel has been laboratory tested both at NSTF and elsewhere, and has been exposed in similar irradiation conditions to much higher total radiation doses in most commercial operating power reactors. No significant degradation of cladding has resulted from normal operation.

(2) The power density and maximum temperatures reached in the NSTF fuel are well below these parameters in the power reactors using similar fuel. No damaged has occurred during normal operations.

(3) The coolant flow rate at NSTF is much lower than used at several higher powered research reactors and at commercial power reactors using zircaloy-clad UO_2 fuel. No significant erosion problems have been observed. At NSTF, corrosion is kept to a reasonable minimum by careful control of the conductivity and pH of the primary coolant water.

(4) The fuel is handled as infrequently as possible, consistent with required surveillance. Any indications of possible damage or degradation are investigated immediately, and damaged fuel would be removed from service, in accordance with Technical Specifications. All experiments placed near the core are isolated from the fuel cladding by a water gap and at least one barrier or encapsulation.

(5) NSTF 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 has determined that the preventive maintenance program would lead to adequate identification and replacement before significant degradation occurred. Therefore, the staff concludes that there has been no apparent significant degradation of safety equipment, and because there is strong evidence that any future degradation will lead to prompt remedial action by NSTF, 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 discussion, the staff concludes that the NSTF reactor is operated under conditions that are conservatively below the safety limits of its components, and that surveillance and maintenance procedures give reasonable assurance that continued operation will pose no significant radiological risk to the health and safety of the public.

18 CONCLUSIONS

Based on its evaluation of the application as set forth above, the staff has determined that

(1) The application for renewal of Operating License R-77 for its research reactor filed by the State University of New York at Buffalo, dated June 14, 1979, as amended, 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 1.

(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 1.

(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 1.

(5) The renewal of this license will not be inimical to the common defense and security or to the health and safety of the public.

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16. ABSTRACT (200 words or less)

The Safety Evaluation Report for the application filed by the State University of New York at Buffalo for a renewal of Operating License R-77 to continue to operate a research reactor has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is owned by the State University of New York and is located on the campus in Buffalo, New York. Based on its technical review, the staff concludes that the reactor facility can continue to be operated by the university without endangering the health and safety of the public or endangering the environment.

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