NUREG-0882

Safety Evaluation Report

related to the renewal of the operating license for the research reactor at the Armed Forces Radiobiology Research Institute

Docket No. 50-170

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

January 1982

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ABSTRACT

This Safety Evaluation Report for the application filed by the Armed Forces Radiobiology Research Institute (AFRRI), Defense Nuclear Agency, for a renewal of license R-84 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 an agency of the U. S. Department of Defense, and is located on the grounds of the National Naval Medical Center, Bethesda, Montgomery County, Maryland. Subject to favorable resolution of one outstanding item discussed in this report, the staff concludes that the facility can continue to be operated by AFRRI without endangering the health and safety of the public.

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1 INTRODUCTION

The Armed Forces Radiobiology Research Institute (AFRRI) submitted a timely application to the U. S. Nuclear Regulatory Commission (NRC) for renewal of the Class 104 Operating License (R-84) for its TRIGA Mark F research reactor by letter (with supporting documentation) dated October 3, 1980. The letter requests renewal of the Operating License for 20 years to permit continued operation at thermal steady state power levels up to and including 1 MW and pulsed operation with step insertions of reactivity up to 2.8% $\Delta k/k$ ($\Delta k/k_{\beta eff}$ =4.00$). AFRRI currently is permitted to operate the reactor within the conditions authorized in past amendments in accordance with Title 10 of the Code of Federal Regulations, Paragraph 2.109 until NRC action on the renewal request is completed.

The renewal application is supported by information provided in various documents: Physical Security Plan, as supplemented on June 6, 1980; Technical Specifications, as supplemented on September 26, 1980; Environmental Impact Appraisal Data, as supplemented on July 13, 1981; Safety Analysis Report, as supplemented on October 9, 1981; Reactor Operator Requalification Program; and Emergency Plan.*

The renewal application contains the information regarding the original design of the 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 NRC staff technical safety review with respect to issuing a renewal operating license to AFRRI has been based on the information contained in the renewal application and supporting supplements, plus responses to requests for additional information. This material is available for review at the Commission's Public Document Room at 1717 H Street N.W., Washington, D. C. This Safety Evaluation Report was prepared by Robert E. Carter, Project Manager, Division of Licensing, Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission.

The purpose of this Safety Evaluation Report (SER) is to summarize the results of the safety review of the AFRRI-TRIGA Mark F reactor and to delineate the scope of the technical details considered in evaluating the radiological safety aspects of continued operation. This SER will serve as the basis for renewal of the license for operation of the AFRRI facility at steady-state thermal power levels up to 2.8% $\Delta k/k$. The facility was reviewed against the Federal regulations (10 CFR Parts 20, 30, 50, 51, 55, 70 and 73), applicable Regulatory Guides (Division 2, Research and Test Reactors) and appropriate accepted industry standards (American National Standards Institute/American

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*The Environmental Impact Appraisal Data and Safety Analysis Report (SAR) were used as basic review documentation, and are referenced throughout this report. They are also listed as reference 1-1.

Nuclear Society (ANSI/ANS 15 series)). Because there are no accident-related regulations for research reactors, the staff has at times compared calculated dose values with related standards in 10 CFR Part 20, the standards for protection against radiation, both for employees and the public. Additionally, a qualified intervenor is contesting the renewal of the AFRRI reactor Operating License, so this SER addresses the contentions accepted by the Atomic Safety Licensing Board.

The initial AFRRI reactor Operating License was issued on June 26, 1962, with the provision that an initial 6 months of preliminary testing be performed before routine operation at authorized license conditions be permitted. These preliminary tests were conducted successfully.

The AFRRI-TRIGA Mark F reactor was initially authorized routine operation in three modes: Mode I, up to 100 kW thermal steady state; Mode II, between 100 kW and 1 MW for periods not to exceed 10 minutes for a maximum of 1 MWhr per day (Square Wave); Mode III, pulse, with step excess reactivity insertions up to 1.9% Δ k/k (2.71$).

After operating the facility for several years, AFRRI: (1) substituted an improved version of fuel for the original core, (2) modified some of the original control instrumentation, (3) increased the reactor primary cooling capacity, and (4) applied for a license amendment for authorization to operate the reactor at steady state thermal power levels up to and including 1 MW. This license amendment, No. 13, was issued by NRC on August 29, 1968.

By letter dated August 14, 1970, AFRRI requested an extension of the expiration date of the Operating License which was due to expire on November 8, 1970. On November 4, 1970, NRC amended license R-84, extending the expiration date to November 8, 1980. The AFRRI reactor has operated for more than 19 years with an average annual use in the experimental programs of about 30 MWhr per year. In terms of radiation exposure of reactor components or production of radioactive material, this amount of operational use corresponds to about four working days per year at maximum authorized steady-state power. On the other hand, the reactor has provided the principal support to a major component of the laboratory's research program, being in use approximately 2000 hours per year. AFRRI is the major research facility of the Defense Nuclear Agency, which is responsible for and committed to studying the radiobiological and biomedical effects of nuclear weapons' radiations. The pulsing research reactor is a key facility at the laboratory, but there are other inhouse radiation sources, used both separately and in conjunction with the reactor in the research programs.

TRIGA reactors--utilizing essentially the same kind of fuel, control rods and drive systems, and safety circuitry as at AFRRI--have been constructed and operated in many countries of the world. Among the approximately

58 such reactors in operation, some since 1958, there have been no reported events that caused significant radiation risk to the public health and safety. Other TRIGA reactors have annual MW hours of operation at least a factor of 10 greater than the AFRRI reactor, primarily because of different types of research programs.

1.1 Summary and Conclusions of Principal Safety Considerations

The staff evaluation considered the information submitted by the applicant, past operating history recorded in annual reports submitted to the Commission by the applicant, reports by the Commission's Office of Inspection and Enforcement, and onsite observations. In addition, as part of the licensing review, the staff obtained laboratory studies and analyses of several accidents postulated for the TRIGA-type reactor.

The principal matters reviewed for the AFRRI 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 the most serious credible accidents and determined that the calculated potential radiation doses outside of the reactor room are not likely to exceed 10 CFR Part 20 doses in unrestricted areas.

(3) The applicant'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 applicant'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 applicant are such that the staff has determined that the applicant has sufficient revenues to cover operating costs and eventually to decommission the reactor facility.

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

(8) The applicant'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.

1.1.1 Outstanding Item

One matter, Emergency Planning, is incomplete at the time of publication of this safety evaluation. This item is discussed further in Section 13.3.

1.2 Reactor Desciption

The AFRRI-TRIGA Mark F is a heterogeneous pool-type reactor. The core is cooled by natural convection of light water, moderated by zirconium-hydride and water, and reflected by water and graphite. The reactor core is located near the bottom of a water-filled clover-leaf-shaped aluminum tank which has an inner diameter of approximately 13 ft and a depth of 19.5 ft. The core and control systems are suspended from a carriage which rides on rails above the reactor tank; this arrangement permits controlled movement of the reactor system across the pool to provide radiation fields in exposure rooms located on opposite sides of the tank. An interlock system prohibits operation of the reactor except in limited positions within the pool.

The reactor core has approximately 85 to 90 cylindrical fuel rods containing uranium enriched to less than 20%, homogeneously mixed with a zirconium-hydride moderator. Each element is approximately 28 in. long and 1.5 in. in diameter. The fuel section is 15 in. long and 1.4 in. in diameter. The element includes, at each end, a thin wafer of burnable poison to partially counteract reactivity changes due to fuel burnup and sections of graphite about 3.5 in. long to provide neutron reflection. The fuel elements are clad with .020-in-thick stainless steel, and all closures are welded.

Reactivity of the reactor core is changed by the operator by four borated-graphite control rods that are approximately the same size and shape as a fuel element, and that are suspended from fail-safe electromagnets located on the support bridge. The ionization chambers used for sensing neutron and gamma-ray flux densities are suspended above the core. The control console is located in a small room adjacent to the reactor room, from which the operator can observe the reactor room and the top structures of the reactor through a large window. The control console consists of typical read-out and control instrumentation.

The reactor tank is embedded in a monolithic reinforced-concrete biological shield. Additional details of the reactor facility and auxiliary systems are contained in the documentation submitted by the applicant.

1.3 Reactor Location

The AFRRI building complex is located toward the south side of the National Naval Medical Center (NNMC) preserve, in southern Montgomery County, Maryland, about 3 mi north of the District of Columbia boundary. The reactor building is constructed of reinforced concrete in gently rolling terrain partially below grade.

1.4 Shared Facilities and Equipment and Special Location Features

The reactor building is attached to a complex of laboratory and support buildings dedicated primarily to radiation biology and biomedical research. Some of these facilities 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 has its own ventilation control system, capable of isolation, which exhausts air through an elevated stack located on the roof of one of the buildings. This stack also exhausts air from other buildings in the complex, for a typical total flow of about 40,000 cfm. The nearest occupied buildings not part of the AFRRI complex, yet still on the NNMC grounds, are about 500 ft from the location of the reactor exhaust stack.

1.5 Comparison with Similar Facilities

The reactor core and control system are similar to those of most of the 58 TRIGA reactors operating throughout the world, 27 of which are in the United States (24 are licensed by NRC).

1.6 Modifications

Other than additions to the total building complex that did not affect the reactor building, the one major change to the reactor facility since the last license renewal in 1970 has been the replacement of the previous control console and its instrumentation. In 1978, AFRRI acquired and installed one of the TRIGA vendor's (General Atomic Company) standard modern control consoles. This control system had been used for a few years, which primarily means that the random malfunctions of components often found in newly assembled instrumentation had probably all occurred, and the failed components were replaced or repaired.

1.7 Operations Summary

The total annual usage of the AFRRI reactor has varied over its history, depending primarily on the requirements of the research programs. Since 1967, the annual thermal energy produced has averaged less than 30 MWhr, with more than 90% accrued during steady-state operation. Since installation of the present stainless-steel-clad core, the reactor has been pulsed more than 6800 times, with nearly 3500 of these pulses corresponding to reactivity insertions of 2.0$ (1.4% $\Delta k/k$) or more (Reference (1-1)).

2 SITE CHARACTERISTICS

Chapter 2 of the Safety Analysis Report provides information pertaining to the site of the AFRRI complex.

2.1 Geography and Demography

2.1.1 Geography

The AFRRI complex is on the grounds of the National Naval Medical Center (NNMC), Bethesda, Maryland. The coordinates of AFRRI are 39°00'10" north latitude and 77°05'21" west longitude. The NNMC is approximately 3 mi north of the Washington, D. C.-Maryland line. The AFRRI site is on a moderate slope that declines northward toward a narrow creek valley. The terrain surrounding AFRRI is rolling, with elevations above sea level ranging from 230 to 320 ft.

The location of AFRRI within the NNMC is shown on the site plan, Figure 2.1. This figure also shows the location of buildings at the NNMC complex with respect to the AFRRI exhaust stack.

2.1.2 Demography

The NNMC has a peak daytime population of approximately 6100 persons, which includes an estimated 1600 visitors.

The nearest occupied building on the grounds of the medical center is about 500 ft from the AFRRI exhaust stack. The main hospital buildings housing most of the hospitalized patients are between about 1000 to 1500 ft from the stack. The nearest residential buildings outside of the medical center site are about 1000 ft from the stack. From that point outward, the population distribution is typical surburban, in high-density single-family homes, but with localized multiple-family dwellings, as well as open and park lands. Schools, churches, nursing homes, and other localized centers of high density occupancy are scattered throughout the community. Within about 2500 ft to the south-southwest, the central business and shopping region of Bethesda starts and extends for about a mile. Directly west of the NNMC lies the campus of the National Institutes of Health, comprised of many laboratory, office, and clinical facilities.

The area around the NNMC is heavily utilized and heavily traveled by commuters in private vehicles. Approximately 2100 ft from the AFRRI stack the metropolitan Washington Beltway is used by motor vehicular traffic of all sorts.

2.2 Nearby Industrial Transportation and Military Facilities

2.2.1 Transportation Routes

There are no heavily used commercial airports within 10 mi of NNMC, and no scheduled airline routes over the complex. There are no heavily used railroads within several miles.

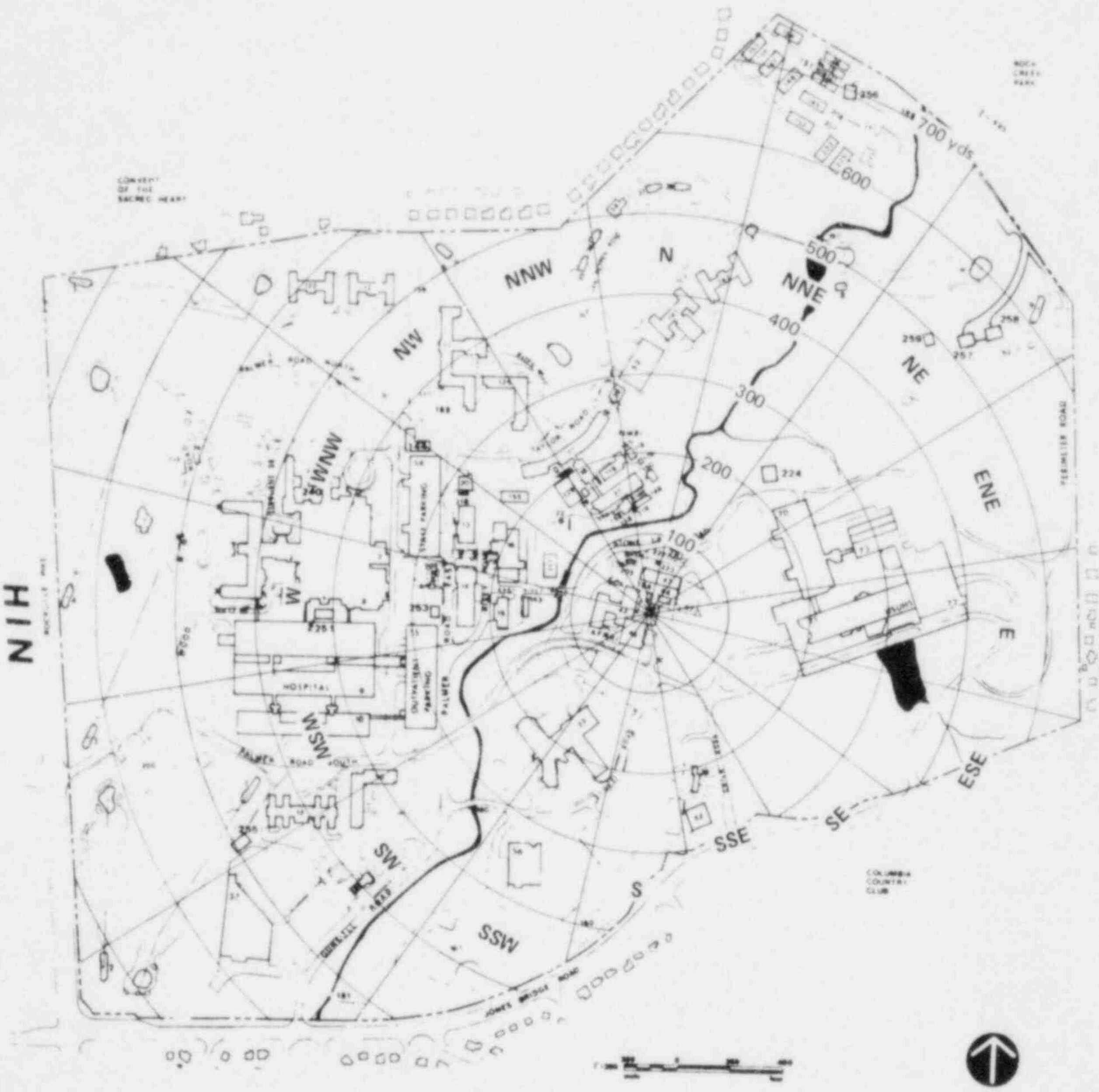

Figure 2.1 NNMC site plan (annuli and sectors with respect to AFRRI stack)

2.2.2 Nearby Facilities

There are no heavy industries or major military establishments other than medical facilities nearby.

2.2.3 Conclusion

Because there are no industrial or military facilities near the reactor that could possibly cause accidental damage to the reactor facility, the staff concludes that the only accidents which need to be evaluated in detail in considering the safety of the public are those which might originate from within the AFRRI facility. These are discussed in Chapter 14 of this SER.

2.3 Meteorology

2.3.1 Description

The AFRRI lies at the western edge of the Middle Atlantic coastal plain, approximately 50 mi east of the Blue Ridge Mountains and 35 mi west of the Chesapeake Bay. The site has a continental type of climate, moderated by the proximity of the Atlantic Ocean. The site lies near the principal track of storms that originate during the winter and spring over the Gulf of Mexico and move northeast through the region. From October through June, the prevailing winds are from the northwest because of the preponderance of anticyclonic circulation over the northern portion of the country. Hence, continental polar air of Canadian origin is the predominant air mass throughout the winter. However, the Appalachian Mountains to the west act as a partial barrier to delay the advance of the cold air.

During the summer, as the mean storm track moves far north of the Washington area, the semipermanent Atlantic High moves northward and eastward and dominates the circulation of air over the eastern United States. Consequently, southerly winds prevail for much of the summer and transfer moist, tropical air from the Gulf of Mexico into the region.

The joint wind frequency distributions presented in Figure 2.2 and Table 2.1 were prepared using hourly National Weather Service observations taken at National Airport, Washington, D. C., for the period from January 1960 through December 1964. This period was utilized because 1964 was the last year that hourly observations were recorded and archived by the National Weather Service The wind speed and direction were measured at a height of 25 ft. The stability is based on observations of wind speed, insolation, and cloud cover. Because AFRRI is about 10 mi from the National Airport, wind conditions are assumed to be similar.

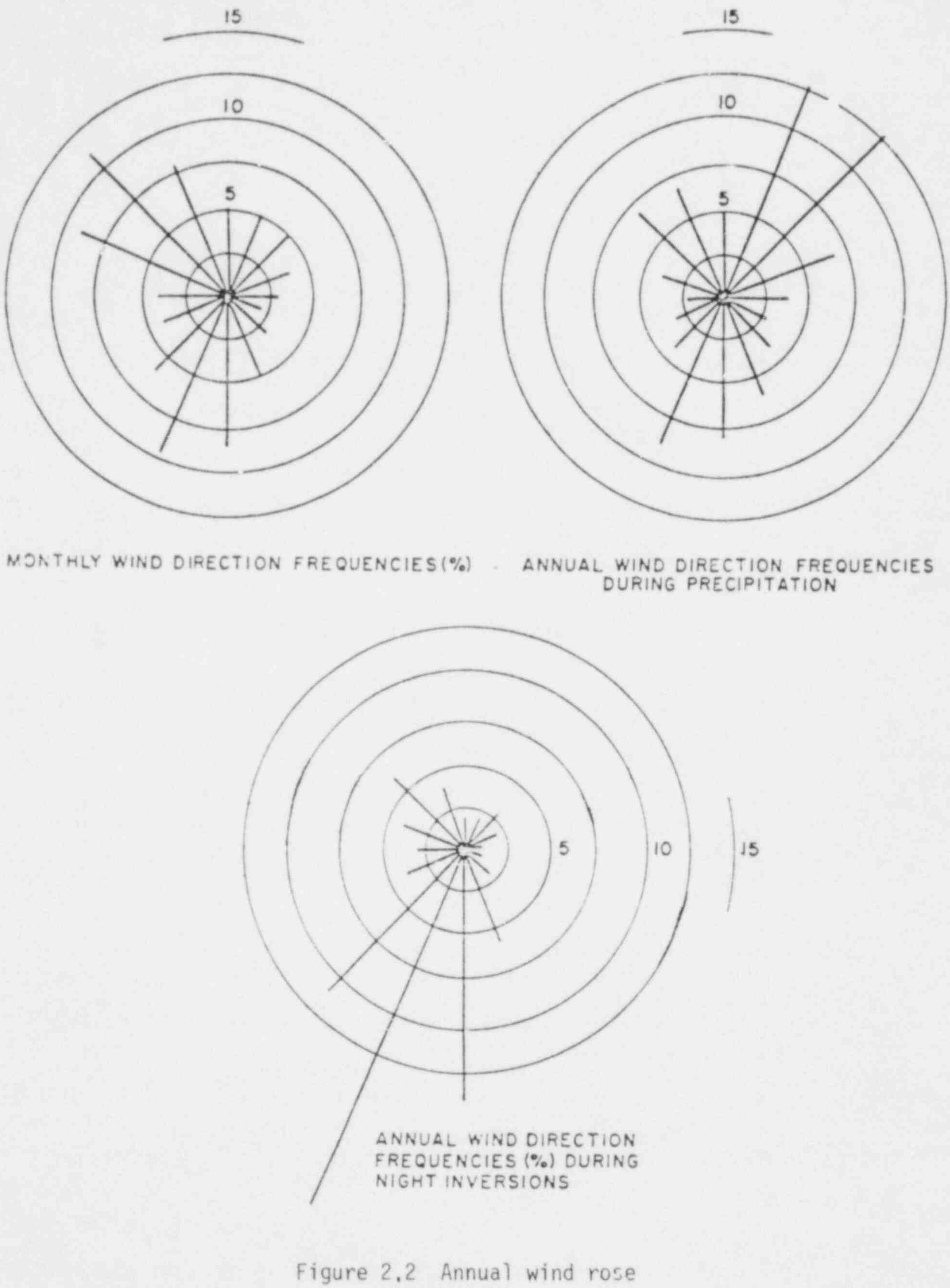

Figure 2.2 Annual wind rose

Table 2.1 Joint wind frequency distribution by stability class
Data period: 1960 through 1964

WIND SECTOR	0.0-1.5	1.5-3.0	3.0-5.0	5.0-7.5	7.5-10.0	>10.0	TOTAL	MEAN SPEED
	WIND SPEED CATEGORIES(METERS PER SECOND)							
NNE	17 .19	59 .67	121 1.38	134 1.53	14 .16	0 .00	345 3.93	4.43
NE	22 .25	75 .85	174 1.98	145 1.65	26 .30	0 0.00	442 5.03	4.41
ENE	27 .31	84 .96	172 1.96	121 1.38	12 .14	4 .05	420 4.78	4.13
E	20 .23	85 .97	105 1.20	53 .60	4 .05	0 0.00	267 3.04	3.58
ESE	27 .31	58 .66	50 .57	17 .19	6 .07	0 0.00	158 1.80	3.20
SE	15 .17	75 .85	99 1.13	31 .35	6 .07	0 0.00	226 2.57	3.50
SSE	23 .26	91 1.04	186 2.12	53 .60	2 .02	0 0.00	355 4.04	3.57
S	41 .47	230 2.62	716 8.15	418 4.76	28 .32	0 0.00	1433 16.31	4.16
SSW	37 .42	179 2.04	451 5.13	231 2.63	39 .44	4 .05	941 10.71	4.16
SW	66 .75	184 2.09	178 2.03	99 1.13	18 .20	6 .07	551 6.27	3.60
WSW	81 .92	175 1.99	109 1.24	38 .43	9 .10	9 .10	421 4.79	3.15
W	40 .46	100 1.14	122 1.39	78 .89	17 .19	4 .05	361 4.11	3.85
WNW	11 .13	46 .52	140 1.59	135 1.54	38 .43	7 .08	377 4.29	4.96
NW	14 .16	42 .48	196 2.23	306 3.48	133 1.51	44 .50	735 8.37	5.97
NNW	17 .19	73 .83	267 3.04	258 2.94	116 1.32	38 .43	769 8.75	5.47
N	17 .19	93 1.06	212 2.41	220 2.50	53 .60	9 .10	604 6.88	4.81
CALM	379 4.31						379 4.31	CALM
TOTAL	854 9.72	1649 18.77	3298 37.55	2337 26.61	521 5.93	125 1.42	8784 100.00	4.18

NUMBER OF VALID OBSERVATIONS 8784 100.00 PCT.
NUMBER OF INVALID OBSERVATIONS 0 0.00 PCT.
TOTAL NUMBER OF OBSERVATIONS 8784 100.00 PCT.

KEY XXX NUMBER OF OCCURRENCES
XXX PERCENT OCCURRENCES

ALL WINDS
DATA SOURCE: NATIONAL AIRPORT, WASH., D.C
WIND SENSOR HEIGHT: 7.00 METERS
TABLE GENERATED: 11/19/80. 20.00.32.

AFFRI-TRIGA REACTOR
BETHESDA, MD.
AFFRI

The wind data show that the predominant wind direction is from the south with a frequency of 15.6%; winds from the northwest occur 9.4% of the time. This reflects the seasonal prevailing wind patterns described earlier. The most frequent stability is Pasquill type D stability, which represents neutral conditions.

2.3.2 Severe Weather

Three types of severe weather conditions may occur in the Washington-AFRRI area: thunderstorms, tornadoes, and tropical disturbances.

Thunderstorms, which occur on the average of 29 days per year in the metropolitan area, often bring sudden heavy rains and may be accompanied by damaging winds, hail, or lightning. The frequency of occurrence at any one site, such as at AFRRI, would be much less than in the Washington metropolitan area as a whole.

Tornadoes are relatively rare. Three rather destructive tornadoes have been recorded in the Washington area. Eleven tornadoes were reported within the 1°-latitude-longitude square containing the AFRRI site during the period 1955 to 1967, giving a mean annual tornado frequency of 0.85, and a recurrence interval of 1500 years for a tornado at the specific location of AFRRI.

Tropical disturbances, during their northwest passage, occasionally influence Washington's weather, mainly with high winds and heavy rainfall. However, of 83 such disturbances between 1901 and 1963, only three made landfall in the Middle Atlantic coast. In addition, Hurricane Agnes, in June 1972, caused extensive damage in the Middle Atlantic states and the District of Columbia. However, there was only insignificant damage at the AFRRI complex.

2.4 Hydrology

The main surface water feature at the NNMC is a second order stream which serves as storm drainage. This stream traverses the complex from southwest to northeast and flows directly into Rock Creek. The watershed which drains into the unnamed stream has an area of 652 acres and includes most of the NNMC grounds and nearly all its buildings. About three-fourths of the watershed is upstream of NNMC and includes commercial areas and suburban residential neighborhoods in Bethesda, Maryland.

The 100-year flood plain on site, calculated for the level of urbanization in the area during 1975, is small because of the high slopes in close proximity to the stream. The flood plain encroaches on no buildings at NNMC, but it does cover portions of two nearby parking lots.

The water table near the AFRRI reactor site is a subdued replica of the surface topography. The depth of the water table measured by test drillings varied from 38.8 to 41.4 ft. Seasonal variation of the water table in the Washington area is typicaly 5 to 7 ft. Shallower water tables are associated with the channels of the onsite stream.

The movement of water in the ground above the water table is through pores and openings along partings in the weathered rock and is generally in a near-vertical direction. After entering the zone of saturation, most of the groundwater probably moves horizontally along a permeable zone in the saprolite near the top of the water table. It is discharged to the onsite stream near the foot of the slope or to small wet weather seeps to the east or west of the site. Because the movement of groundwater in the bedrock is confined to fracture zones, the direction of groundwater movement may vary from the direction perpendicular to the general water table contours.

No determination of rate of movement of groundwater has been made. In general, the velocity of groundwater movement through fine-grained materials, such as clay or silt, is very low (on the order of a few feet per year), whereas the velocity of groundwater movement in the saturation zones near the water table, where velocities are at a maximum, may range from less than a foot to several feet per day.

2.4.1 Conclusion

From the information provided by the licensee in the application and from site visits, the staff concludes that there is low risk of flooding of the reactor as a result of precipitation, runoff, or rising groundwater.

2.5 Geology and Seismology

The NNMC site is in an area where silts, clay, and fine-grained sand are interspersed with pods of unweathered bedrock, down to depths of 35 to 45 ft. Below this, generally, lies a massive gneissic bedrock. The reactor site is situated in a zone where the probability of seismic activity is very small. The nearest known fault line to AFRRI is approximately 19 mi away.

2.5.1 Conclusion

The staff concludes that seismic activity in the area of the AFRRI reactor does not pose a significant risk of damage to the facility.

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

Chapter 3 of the licensee's Safety Analysis Report provides information on the design, construction, and functions of the reactor building, reactor systems, and auxiliary systems.

3.1 Wind Damage

Meteorological data indicate a low frequency of tornadoes and effects of tropical disturbances, but a moderately high frequency of summer thunderstorms. However, the reactor tank is embedded in a monolithic reinforced-concrete shield, integrally constructed in a reinforced-poured-concrete building located partially below grade. Therefore, the staff concludes that wind or other storm damage to the AFRRI reactor facility is unlikely.

3.2 Water Damage

The reactor building is situated on the side of a well-drained hill, above the flood plain. Therefore, the staff concludes that there is reasonable assurance that damage to the reactor by flood or groundwater is small.

3.3 Seismic-Induced Reactor Damage

The nearest seismic fault is some 19 mi away, and the incidence of seismic activity has been infrequent. Further, AFRRI is situated in an area of low probability of seismic activity. These considerations, in addition to the construction features of the reactor building, lead the staff to conclude that the risk of 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 rods suspended from the superstructure, which also supports the reactor core. 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 AFRRI 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. These malfunctions have almost exclusively been one of a kind (that is, no repeats) and/or of components that were fail safe or self annunciating (see I&E Inspection Reports and reports of Reportable Occurrences from the licensee, Docket No. 50-170). 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 for the requested period of renewal will not increase the risks to the public.

4 REACTOR

4.1 Summary Description

The AFRRI reactor is a TRIGA Mark-F unit designed and fabricated by the General Atomic Company. This reactor first achieved criticality in 1962. It is an open-pool-type, light-water-moderated reactor that is currently authorized to operate in either the steady-state mode up to 1 MW thermal, or in the pulse mode with a step reactivity insertion of up to 2.3% $\Delta k/k$.

This reactor is used as a source of intense flux densities of ionizing and neutron radiation for research in radiobiology and related matters, as well as for training AFRRI reactor operators and for radioisotope production.

4.2 Reactor Core

The reactor core consists of a relatively compact array of approximately 87 standard TRIGA fuel elements, four control rods and control rod guides, and a startup neutron source and source guide tube. The fuel elements are held in concentric cylindrical arrays by an upper and a lower grid plate. The single central position, "ring A," contains the transient rod. The fueled rings are identified as B, C, D, E, and F, from the center outward. The active (or fueled) region of the reactor core forms a right circular cylinder approximately 17 in. in diameter and 15 in. high. The fully loaded operational core currently contains about 3.3 kg of U-235.

4.2.1 Fuel Elements

The AFRRI reactor uses standard TRIGA stainless-steel-clad cylindrical fuel elements in which enriched uranium is homogeneously mixed with a zirconium hydride moderator. The fuel part of each element consists of a cylindrical rod of uranium-zirconium hydride containing 8.5 weight-percent uranium with U-235 enriched to less than 20%. The hydrogen-to-zirconium atom ratio of the fuel moderator material is approximately 1.7 to 1. The nominal weight of U-235 in each fuel element is 38 g. The fuel section of each element is approximately 15 in. in length and 1.43 in. in diameter. Graphite end plugs 3.44 in. long are located above and below the fuel section and function as neutron reflectors. Burnable poison (samarium) is included in each fuel element to compensate partially for reactivity changes caused by fission-product buildup and uranium burnup. The samarium is mixed with aluminum to form thin wafers that are placed between the fuel moderator material and the graphite reflectors. At least two fuel positions contain special instrumented fuel elements into which thermocouples were fitted during fabrication. In all other respects, these elements are identical to standard fuel elements. The thermocouples monitor the axial temperatures in the instrumented elements.

The fueled section, the burnable poison wafers, and the graphite reflectors are contained in a 0.020-in.-thick Type 304 stainless-steel walled can. The can is sealed by welds with stainless-steel fittings at the top and bottom. Each element is about 28.3 in. in length and weighs about 3.4 kg.

4.2.2 Control Rods

Power levels in the AFRRI-TRIGA reactor are regulated by three standard control rods and one transient control rod. All four rods contain boron as the neutron-absorbing material.

The control rods are clad in aluminum tubes approximately 31 in. long and 1.25 in. diameter. The upper 15.25 in. of the tube contains a compacted borated-graphite rod as the neutron absorber or poison. The lower end of the tube (the follower) contains a solid aluminum rod 15.5 in. long and 1.125 in. in diameter.

4.3 Reactor Tank

The reactor core is positioned in the reactor tank under approximately 16 ft of light, demineralized water (Figure 4.1). This water serves as radiation shielding, neutron moderator and reflector, and reactor coolant. The reactor tank is constructed of aluminum and embedded in ordinary concrete. The reactor pool is cloverleaf-shaped, approximately 19.5 ft deep and 13 ft wide across the lobes (Figure 4.2). The nominal wall thickness of the aluminum tank is 3/8 in., except for the two cloverleaf projections that extend into the exposure rooms, where the wall is 1/4 in. thick. The tank bottom is 1/2 in. thick.

The reactor tank contains approximately 15,000 gal of water. The natural thermal convection of this water adequately disperses the heat generated in the core by the normal operations of the reactor, both steady-state and pulsed. The pool water is pumped through an external heat exchanger system that ultimately disposes of the heat to the atmosphere. In the event of the loss of all coolant, the natural convection of air through the core will maintain its temperature below the cladding failure level, and all fission products will be retained within the individual elements (4-1).

4.4 Support Structure

A four-wheeled carriage that travels on two tracks spanning the reactor tank supports the reactor core and is used to move it laterally from one operating position within the tank to another. In addition, this carriage serves as a support for the four control-rod drives and the core power-level monitors.

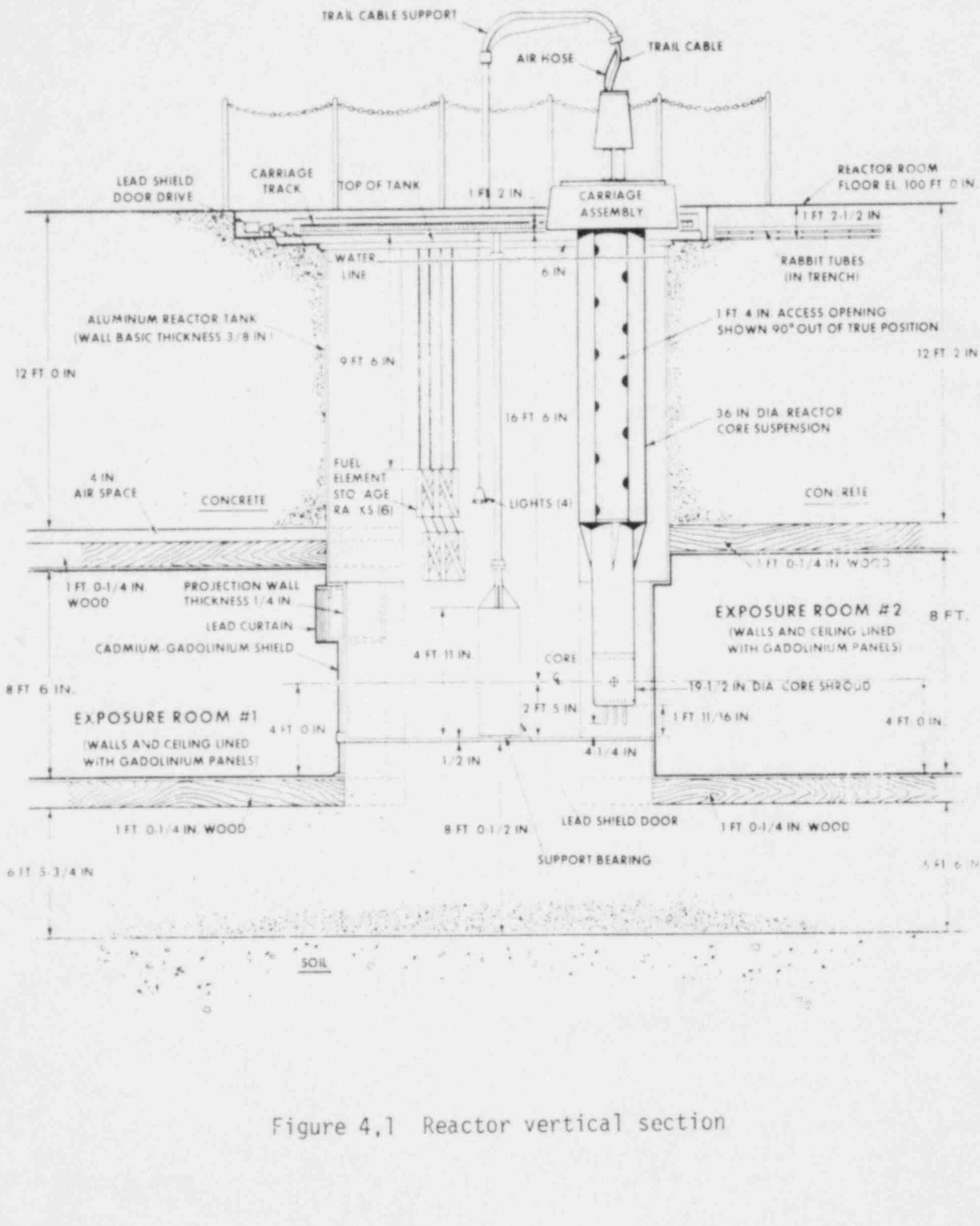

Figure 4.1 Reactor vertical section

A core support structure is attached to the underside of the carriage. This structure consists of an aluminum cylinder 12 ft high, connected at its bottom to an aluminum adapter 19.5 in. in diameter and 5 ft high. Both the cylinder and adapter are formed from 5/16-in.-thick aluminum plate. A vertical slot about 16 in. wide, extending the full height of the aluminum cylinder on one side, provides access to the inside of the support structure and allows core components to be installed and removed without raising them above the pool water level.

The reactor core is enclosed in a 3/16-in.-thick aluminum shroud attached to the bottom of the core-support adapter. Grid plates bolted to the top and bottom of the shroud hold the fuel elements the control-rod guides, and the neutron-source guide tube in place.

4.5 Reactor Instrumentation

The operation of the reactor core is monitored by six separate detector channels. Two of these detectors are the thermocouples in two instrumented fuel elements positioned in different regions of the core. A fission ionization chamber, two boron-lined ionization chambers, and a gamma-ray detector complete the system. Five of these detectors are used to provide independent channels that monitor the neutron-flux density or fuel temperature of the core and provide trip signals to the safety circuits. The gamma-ray channel is used to measure parameters during pulsing of the reactor.

4.6 Biological Shield

The reactor core is shielded in the radial direction by the reactor pool water and a minimum of about 9 ft of ordinary concrete (except for the protrusions into the two exposure rooms). The vertical shielding consists of about 16 ft of water above the core and about 1.5 ft of water and 8 ft of ordinary concrete below it, which separates the reactor tank from the subsoil underlying the reactor building.

Two lead shielding doors are located in the reactor pool, dividing it into two equal sections. When the shielding doors are fully closed, authorized personnel may enter one exposure room without significant radiation exposures if the reactor core is positioned at the opposite side of the pool.

Access to each exposure room is by way of separate reinforced-concrete rolling-plug shield doors. Both doors have steps on all four sides to prevent radiation streaming from the exposure room (ER), and both are about 7 ft wide by 9 ft high. The ER #1 door is about 12 ft thick, and the ER #2 door is 9 ft thick; both provide adequate protective shielding.

The exposure room surface and the inner surface of the plug doors are covered with 1 ft of wood to limit neutron irradiation of the concrete. In turn, the wood is lined with gadolinium panels that absorb thermal neutrons and thereby decrease formation of both fixed and airborne radionuclides.

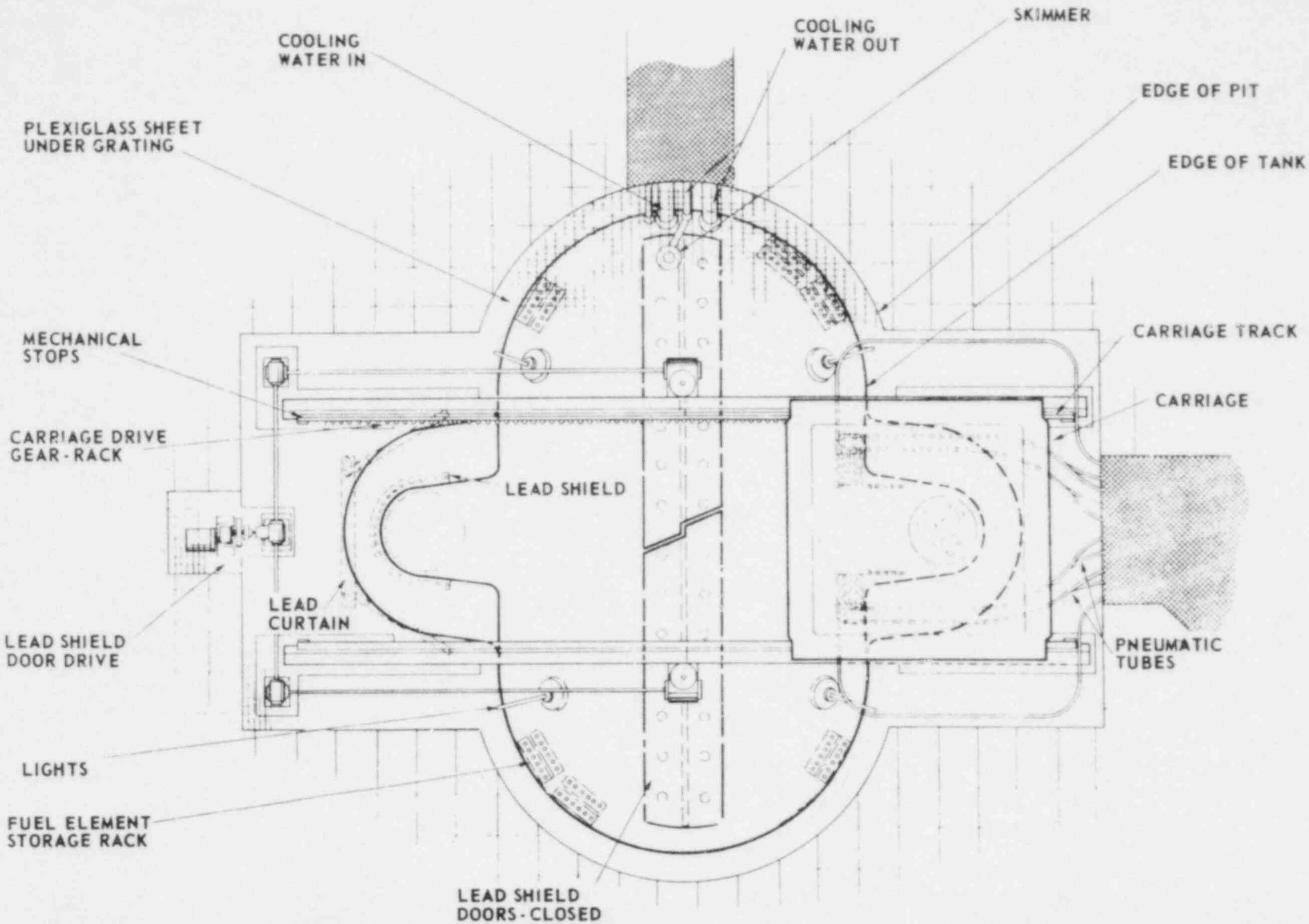

Figure 4.2 AFRRI-TRIGA reactor tank plan

4.6.1 Conclusion

The staff concludes that the AFRRI-TRIGA reactor is designed and built according to good industrial practices. It consists of standardized components representing hundreds of reactor years of operation, and includes redundancy of safety-related systems.

4.7 Dynamic Design Evaluation

The safe operation of a TRIGA reactor during normal operations is accomplished by the control rods, and is monitored accurately by the core power-level detectors. A backup safety feature resides in the reactor core's inherent large negative temperature coefficient of reactivity resulting from an intrinsic molecular characteristic of the zirconium-hydride alloy at elevated temperatures. Because of the large prompt negative temperature coefficient, step insertions of excess reactivity resulting in an increasing fuel temperature rapidly and automatically will be compensated for by the fuel matrix. This will terminate the resulting excursion without any dependence on (1) the electronic or mechanical reactor safety systems or (2) actions of the reactor operator. This inherent characteristic of the uranium-zirconium-hydride fuel has been the basis for designing these reactors with a pulsing capability as one normal mode of operation. Similarly, because of the large negative temperature coefficient of reactivity, changes of reactivity resulting in a change in fuel temperature during steady-state operation will be rapidly compensated for by this special fuel mixture, thus limiting the reactor steady-state power level (4-2). Nonnormal operations (accidents) are discussed in Chapter 14.

4.7.1 Excess Reactivity

The maximum power excursion (transient) that could occur would be one resulting from the rapid insertion of the total available excess reactivity, whether intended or inadvertent. The AFRRI-TRIGA Mark-F fuel loading is limited by the Technical Specifications to 3.5% $\Delta k/k$ (5.0\$) excess reactivity above xenon-free cold critical, with or without all experiments in place.

General Atomic has conducted numerous experiments to evaluate TRIGA fuel performance involving rapid reactivity insertions up to 3.5% $\Delta k/k$ (5.0\$) (these have included thousands of pulses with peak power levels of 2000 MW or greater) with fuel element temperatures reaching up to 1100°C. The experiments revealed no apparent fuel damage, in the type of fuel currently used in the AFRRI-TRIGA reactor (4-2, 4-3).

AFRRI has applied for a license amendment for a change in its Technical Specifications that would increase its authorized step reactivity insertion from 2.3% $\Delta k/k$ (3.28\$) to a maximum of 2.8% $\Delta k/k$ (4.00\$) in the pulse mode. The Technical Specifications will continue to limit the measured fuel temperature to not more than 600°C. The licensee has committed to approaching the larger insertion limit by small increases to ensure that no Technical Specification limits will be exceeded. The General Atomic work has indicated that a measured fuel temperature of 600°C in the B or C ring as a result of a pulse implies that local transient peaks in fuel temperature may approach 950°C (4-3). This temperature peaking is in the periphery of the fuel element; it decreases within seconds after the pulse is complete as the zirconium-hydride alloy redistributes the heat by thermal conduction. However, because the ambient water is required to achieve the excess reactivity for pulsing, the cladding will necessarily be immersed, and therefore its maximum temperature will always be lower than that of the fuel itself.

4.7.2 Shutdown Margin

The Technical Specifications limit the total excess reactivity of 3.5% $\Delta k/k$ (5.0\$) and the minimum shutdown margin is 1.00\$. The sum of the reactivity worths of all experiments in the reactor and the associated experimental facilities is limited by Technical Specifications to 2.1% $\Delta k/k$ (3.0\$). All of these limits are applicable for any and all fuel loadings and reactor operating conditions.

The change in reactivity resulting from full operational withdrawal of a standard control rod is approximately 1.4% $\Delta k/k$ (2.0\$). The change in reactivity caused by complete operational withdrawal of the transient control rod is approximately 2.8% $\Delta k/k$ (4.0\$). The change in reactivity resulting from the complete physical removal of the transient control rod from the core is about 3.0% $\Delta k/k$ (4.3\$).

The shutdown margin of the AFRRI reactor with all control rods in place in a core having maximum authorized excess reactivity is approximately 5\$= (3x2\$+4\$-5\$). Therefore, to comply with the minimum shutdown margin limit and to be able to perform experiments of positive reactivity worth, the normal loading must be less than the maximum authorized.

4.7.3 Conclusion

The staff concludes that the inherent large, prompt, negative temperature coefficient of reactivity of the uranium-zirconium-hydride fuel moderator provides a basis for safe operation of the AFRRI reactor in the steady-state mode, and is the essential characteristic supporting the capability of operation of the reactor in a pulse mode.

Furthermore, the staff concludes that with an excess reactivity of less than 3.5% $\Delta k/k$ (5.0\$), the worth of the AFRRI control rods will ensure a shutdown margin within Technical Specifications even if the most reactive control rod (transient rod) were totally removed from the core. In addition, TRIGA cores have been operated in pulse mode with reactivity insertions as large as 3.5% $\Delta k/k$ (5.0\$) with no significant mechanical changes and no loss of integrity of the cladding (4-2,4-3). Therefore, the staff concludes that it is reasonable to authorize routine operation at pulse sizes significantly lower than 3.5% $\Delta k/k$ (5.0\$) namely up to 4.00\$ (2.8% $\Delta k/k$).

Accordingly, the staff is including in the revised Technical Specifications the authorization to operate with larger pulses, not to exceed an excess reactivity insertion of 2.8% $\Delta k/k$, at any location in the pool, with the stipulation that the measured temperature in the B ring not exceed 600°C during or following the pulse.

4.8 Functional Design of Reactivity Control System

Power level in the AFRRI reactor is regulated by use of three standard control rods spaced 120° apart in the reactor core D ring (the third ring from the center) and one transient control rod positioned in the center of the core. All of these rods contain boron as the neutron-absorbing material. The control rods are moved in and out of the core, vertically, by electromechanical systems. Each control rod drive system is energized from the control console through its own independent electrical cables and circuits, which tends to minimize the probability of multiple malfunctions of the drives. Any or all of the four control rods can be released to fall by gravity on the receipt of a "scram" signal.

4.8.1 Standard Control Rod Drive

The standard control rod drive mechanism consists of an electric motor coupled to a rack-and-pinion gear system. There is an electromagnet on the bottom of the rack that normally engages an iron armature on the upper end of a connecting rod; the control rod is attached to the lower end of this connecting rod. Under normal operations, the motorized system slowly withdraws and inserts a control rod. If power to the electromagnet is interrupted for any reason, the connecting rod is released and the control rod falls by gravity into the core, rapidly shutting the reactor down (scramming).

The reactor interlock system prevents the simultaneous withdrawal of two or more standard control rods during steady-state operation and prevents the withdrawal of any standard control rod during pulse operation.

4.8.2 Transient Control Rod Drive

The transient control rod is operated by a pneumatic-electromechanical system. The pneumatic part of the system consists of a single-acting piston in a pneumatic cylinder. The piston is attached to the transient control rod by a connecting rod. During pulse reactor operation, compressed air is admitted to the bottom of the cylinder through a solenoid valve, which drives the piston and transient rod upward. When the piston strikes the anvil of the shock absorber, it is stopped, ending the insertion of reactivity. The solenoid valve is then deenergized, venting the air in the piston, causing the transient rod to drop by gravity into the reactor core. Changing the anvil position determines where the piston is stopped during pulse operation, and, thereby, the amount of reactivity insertion.

If air is admitted to the pneumatic cylinder through the solenoid and is not subsequently vented, the air pressure holds the piston against the anvil. In this configuration, the electromechanical operation of the system allows the transient control rod to operate like a standard control rod.

4.8.3 Scram Logic Circuitry

The scram circuitry ensures that several reactor core and operational conditions must be satisfied for reactor operation to occur or continue (in accordance with the Technical Specifications). The scram logic circuitry involves a set of open-on-failure logic relay switches in series. Any scram signal, or component failure in the scram logic will result in the loss of standard control-rod magnet power and loss of air to the transient rod cylinder, causing a reactor shutdown. (Details of the individual core sensors can be found in Section 7.5.) The time between activation of the scram logic and the total insertion of each control rod is limited to less than 1 second by the Technical Specifications, to ensure adequate safety for the reactor and fuel elements for the range of anticipated operations at AFRRI.

4.8.4 Conclusion

The AFRRI reactor is equipped with safety and control systems typical of most nonpower reactors. Therefore, the staff concludes that there is sufficient redundancy of control rods so that the reactor can be brought to safe shutdown even if the most reactive control rod fails to insert upon receiving a "scram" signal. The power level sensors are firmly attached and move with the core. Furthermore, more than one monitor channel detects each of the two indications of power level--fuel temperature and neutron flux density--providing redundancy to mitigate consequences of single malfunctions.

In addition to the active electromechanical safety controls for normal and abnormal operation, the large, prompt, negative temperature coefficient of reactivity inherent in the uranium-zirconium hydride fuel moderator discussed in Section 4.7 provides a unique backup safety feature. The reactor shutdown mechanism of this fuel terminates reactor transients that produce large increases in temperature, and will limit steady-state power level. Because this inherent shutdown mechanism acts to limit the magnitude of a possible transient accident, it would mitigate the consequences of such accidents, and can be considered to be equivalent to a failsafe engineered safety feature.

In accordance with the above discussion, the staff concludes that the reactivity control systems of the AFRRI reactor are designed and function adequately to ensure safe operation and safe shutdown of the reactor under all normal operating conditions.

4.9 Operational Procedures

AFRRI has implemented a thorough preventive maintenance program that is supplemented by a detailed preoperational checklist to ensure that the reactor is not operated at power without all of the safety-related components fully operational.

The reactor is operated by trained 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 this reactor are reviewed by the AFRRI Reactor and Radiation Facility Safety Committee for potential effects on the reactivity of or damage to the core, as well as for possible effects on the health and safety of employees and the general public.

4.10 Conclusion

The staff review of the AFRRI reactor facility has included studying its specific design and installation, its control and safety instrumentation, and its specific pre-operational and operating procedures. As noted earlier, these features are similar to those typical of the research reactors of the TRIGA type operating in many countries of the world, more than 20 of which are licensed by NRC. Based on the review of the AFRRI reactor and experience with these other facilities, the staff concludes that there is reasonable assurance that the AFRRI reactor is capable of safe operation, as limited by its Technical Specifications, for the period of the license renewal.

5 REACTOR COOLANT AND ASSOCIATED SYSTEMS

5.1 Systems Summary

The systems associated with reactor cooling are the primary cooling system, the secondary cooling system, the primary water purification system, the primary coolant makeup water system, and the reactor pool nitrogen-16 (N-16) diffuser.

5.2 Primary Cooling System

The reactor core is cooled by natural convection in a nonpressurized pool containing approximately 15,000 gal of light, demineralized water. This primary coolant is drawn from the pool by the primary coolant pump at a rate of approximately 350 gpm, pumped through the tube side of a shell and tube heat exchanger, and returned to the reactor pool.

There are no beamports in the tank wall. There is no piping through the tank wall near or below the level of the reactor core; the primary coolant water pipes enter the pool across the top edge of the tank. The bottoms of the exit pipe and return pipe are 4 ft and 8 ft, respectively, below the pool surface.

To prevent siphoning of coolant from the pool in case of a primary coolant system leak or rupture, holes are drilled in both the exit and return lines about 4 in. below the pool surface to act as siphon breaks.

With the exception of the tank protrusions into the exposure rooms which are 1/4-in.-thick aluminum, the tank wall and bottom are supported by poured concrete. There is an emergency water supply system whereby raw municipal water can be used to maintain the pool level above the reactor core in the event of a large pool leak or rupture.

If the circulation pump in the primary system or any part of the secondary system should fail, the 15,000 gal of pool water provide an adequate heat sink to allow continued reactor operation for several hours at 1 MW before pool water temperature limits or other limiting Technical Specifications are approached.

5.3 Secondary Cooling System

The secondary coolant pump draws water from the cooling tower sump, passes it through the shell side of the heat exchanger at a rate of about 700 gpm, and returns it to the top of the cooling tower. In the heat exchanger, the pressure of the secondary system is higher than that of the primary, so that if any tube leaks occur, potentially radioactive primary water will not have uncontrolled access to the sanitary sewer system by way of the cooling tower.

5.4 Primary Water Purification System

The primary coolant system is maintained at high purity to minimize corrosion of reactor components and to minimize formation of neutron-activated impurities.

The primary coolant purification pump draws about 20 gpm from the return line of the primary system, passes it through a water monitor box, a 5- to 25- micron prefilter, and two parallel mixed-bed demineralizers. This removes chemical species which might cause corrosion, particulates, and radioactive materials produced in the water by operation of the reactor. This purified water is returned to the primary coolant line, which returns to the pool. The measurements made in this loop include:

(1) water temperature, conductivity, and gross radioactivity

(2) pressure drop across the prefilters

(3) the flow rate at the outlet of each demineralizer

5.5 Primary Coolant Makeup Water System

The primary water makeup system functions to replace coolant water lost through evaporation from the pool. During normal operation, raw city water is filtered and demineralized and, after the conductivity is measured, introduced into the pool. In the event of the failure of this system, there is an 80-gal tank supplied by a distillation unit that can be gravity fed into the pool. Eighty gallons approximates the pool water lost by evaporation in 1 week.

5.6 Nitrogen-16 Diffuser System

The N-16 diffuser system imparts a turbulent motion to the coolant that increases the time for radioactive decay of the N-16 before it escapes from the water surface into the reactor room. The diffuser system consists of a pump installed on the core-support dolly structure with the intake and discharge lines located inside the reactor core-support structure. When the reactor is in operation, about 70 gpm of pool water is discharged above the reactor core inside the support structure in a tangential direction.

5.7 Conclusion

The staff concludes that the reactor coolant systems are adequate to maintain fuel temperatures within safe limits during normal operations and to ensure that no component failure or combinaton of component failures will cause a significant radioactive release to the environment.

6 ENGINEERED SAFETY FEATURES

6.1 Summary

The only engineered safety system associated with the AFRRI facility that is not directly associated with reactor control is the ventilation system. The only significant airborne radioactive materials formed as a result of normal reactor operations are argon-41 (Ar-41) and nitrogen-16(N-16).

6.2 Ventilation System

The reactor building has a ventilation system that can be isolated from that of the rest of the AFRRI complex. Air enters the building through two supply fans. The building is maintained at a slight negative pressure by the exhaust fan. This ensures that air flows from clean areas (such as offices) to potentially contaminated areas (such as exposure rooms and hot cells), through roughing and absolute filters, through the exhaust fan, and up the AFRRI stack.

The reactor room/control room area is connected to the rest of the AFRRI building complex by a double door system that serves as an air lock. These doors are fitted with flexible gasket systems that impede air exchange between the reactor building and the other occupied areas of the complex.

The door between the reactor room and the hallway past the control room/office area is sealed with a compressible gasket. Hatches in the roof and floor of the reactor room are sealed by flexible gaskets.

There are three air supply dampers to the reactor room and one exhaust damper from the room. In the event of the release of airborne radioactivity within the reactor room, the continuous air monitor alarm triggers a signal that automatically closes all four dampers, isolating the reactor room. Manual control of the dampers is also provided.

The dampers on the air ducts are spring loaded and held open by pneumatic units, so that interruption of either the air supply or electrical power causes automatic fail-safe closure to seal the room from the external environment.

Visual and audible alarms indicate failure of the exhaust fan motor or loss of flow of the reactor building air to the stack. Additionally, if this motor becomes uncoupled from the fan (that is, the "V" belt breaks), the resulting drop in the total AFRRI stack flow rate would be indicated on a strip chart recorder.

6.3 Conclusion

The reactor building ventilation system equipment and procedures are adequate to control the release of airborne radioactive effluents in compliance with regulations and to minimize releases of airborne radioactivity in the event of off-normal or accident conditions. Therefore, the staff concludes that the public will be adequately protected from airborne radioactive hazards related to reactor operations.

7 CONTROL AND INSTRUMENTATION

7.1 Systems Summary

The control and instrumentation systems provide the means for operating the various components of the reactor and the experimental facilities in a manner consistent with design objectives.

7.2 Primary Control Systems

There are several different control systems associated with the reactor facility, each of which is used to control specific components of the overall installation. Control and power cables are carried in cable trays from various parts of the facility. This ensures that the cables are relatively safe from physical damage and are readily accessible for maintenance, repair, and inspection. A voltage regulator is provided to prevent external electrical power surges from overloading the circuits in the reactor console. Specific details of the various control systems are described in the following sections.

7.2.1 Reactor Control

The reactor is controlled from the operator's console by adjusting four control rods containing a neutron-absorbing material. Individual rod positions are indicated by potentiometer and limit-switch circuitry with digital voltmeter read-out at the control console. The read-out indicators are conveniently placed for direct operator observation. The rods are physically adjusted by use of three rack-and-pinion drives and one pneumatic-electromechanical drive. Because a loss of system power results in the control rods immediately dropping into the reactor core by gravity, causing automatic safe reactor shutdown, and because the pool water can dissipate reactor core heat, emergency electrical power is not needed for any of the reactor control systems.

7.2.2 Core Support Carriage Control

The four-wheeled carriage that is used to move the reactor core from one operating position to another within the reactor pool is propelled by an electric motor and a rack-and-pinion gear system. The unit is operated from the control room and is provided with microswitches that control the limits of carriage travel. These microswitches are connected to the facility interlock system described in Section 7.4. The specific safety features of this system include mechanical stops to automatically limit carriage travel in the event of a microswitch malfunction. The carriage position is indicated on an indicator at the reactor console and is verifiable by direct observation from the console by the reactor operator. A clutch system at the motor is provided to prevent damage to the reactor core or to the drive system in case of electrical malfunction.

7.2.3 Shield Doors Control

The two large shield doors located in the reactor tank that divide it into equal sections provide shielding during reactor operations. They also provide a positive physical barrier that could prevent movement of the reactor core from one end of the pool to the other. The doors are controlled from the control console, and two limit switches that indicate fully open and fully closed positions are provided. The limit switches are connected to the facility interlock system in such a manner as to deny electrical power to the control rod magnets and the exposure room shield doors. A clutch in the drive train prevents damage to the reactor core, the shield doors, and the drive system in case of electrical malfunction.

7.2.4 Transient Rod Drive Control

The transient rod is equipped with two independent drive mechanisms, both controlled from the control room. The primary drive is a pneumatic system composed of an accumulator, a three-way solenoid valve, a remotely located air compressor, and a pneumatic piston. In this drive system, gravity holds the transient rod fully inserted into the core unless air is supplied to the cylinder and the anvil is off the bottom. The position of the transient rod is indicated by a system of microswitches that show the transient rod anvil position on a digital voltmeter display located on the control console. Adjustment of the anvil's position controls the piston stroke and hence the amount of reactivity inserted into the reactor to initiate a pulse.

7.2.5 Exposure Room Door Control

The control systems for access to both exposure rooms are similar and are both part of the interlock system discussed in Section 7.4. Electric power to open either exposure room door is provided by the control power box located in the preparation area. Two microswitches are associated with opening and closing the doors. When either door is fully open, a microswitch is activated and the electric power is interrupted. A second microswitch is activated and interrupts power when either door is fully closed. In addition, circuit breakers are installed in the preparation area to allow manual power interruption in the event of failure or malfunction of either the control power box or the microswitches.

7.2.6 Control Rod Withdrawal Prevent (RWP)

This portion of the interlock system is designed primarily to prevent the control rods from being withdrawn further unless specific operational conditions are satisfied. This interlock is preventive in nature and differs from the scram interlocks by the fact that a scram will drop all control rods. The six RWPs are as follows:

(1) Air cannot be applied to the transient rod drive unless the reactor power is below 1 kW.

(2) Source-level neutrons, as indicated on the operational channel, must be measured and be above a minimum before power can be supplied to the control rod "up drive."

(3) The rate of reactor power increase must have a period of at least 3 seconds for electrical power to be supplied to the control-rod "up drive."

(4) The temperature of the reactor tank water must be below 60°C for electrical power to be supplied to the control-rod "up drive."

(5) The operational channel must be supplied with high voltage for electrical power to be supplied to the control-rod "up drive."

(6) The "up drive" cannot be energized while the operational channel is in any mode other than "operate."

The control rods can be driven down or dropped, thereby causing a safe reactor shutdown, independent of any or all of the above RWP interlocks.

7.3 Supplementary Control Systems

Several systems control specific functions and/or components of the facility other than the safety-related units, and increase the overall safety of operations at AFRRI. These supplementary systems are discussed below.

7.3.1 Time-Delay Circuits

There are two such units that cause a warning horn to sound in either or both of the exposure rooms under certain conditions. During the 3-min interval, the control rods cannot be withdrawn. One of the circuits responds to the position of the lead shield doors in the reactor tank, and the second is connected directly to the control console. The entire purpose of the 3-min horn is to warn personnel that an operation is about to begin that may result in radiation in the exposure rooms.

7.3.2 Emergency Stop Circuit

This circuit provides a means by which an individual trapped in one of the exposure rooms can prevent reactor startup independent of the reactor operator. This circuit consists of three emergency stop buttons and emergency stop relays. An emergency stop button is located in each of the two exposure rooms, and one is located on the reactor console. When depressed, each emergency stop button will scram the reactor if it is operating, or interrupt the 3-min time-delay circuits and prevent reactor startup, and annunciate at the control console.

7.3.3 Exposure Room Door Controls

These controls consist of the electric power control boxes specifically dedicated to the exposure rooms. They are interlocked with the lead shield door and the core-support carriage controls, and they control the drive motor of the exposure room shield plug door. Each exposure room shield door has a hand-crank control that is normally padlocked.

7.3.4 Core Support Carriage Position

This control system uses carriage position (as indicated by a system of microswitches and lights) and a linear potentiometer to provide information to the interlock system to permit or deny operation of the reactor. As discussed in Section 7.2, this system is operated from the control room.

7.4 Control Interlock System

Operation of this system encompasses the position of the core-support carriage, the lead shield-door position, the positions of the exposure room doors, and the time-delay circuit. The resulting logic establishes a set of conditions that must all be satisfied before the reactor can be operated, with specific conditions that must be met for reactor operation in each of the three permissible pool positions. Position 1 is within 13 in. of the ER #1 tank wall, Position 2 is identified as between Position 1 and Position 3, and Position 3 in within 13 in. of the ER #2 tank wall.

7.5 Instrumentation System

The instrumentation system consists of both nuclear and nonnuclear components, annunciators, read-out devices, digital indicators, chart recorders, meters, and gauges. In addition, there are several radiation monitors with associated alarms for health physics purposes.

The nuclear instrumentation system consists of six channels for nuclear operation, control, and safety. The four safety channels are independent of the two operational channels and are provided with independent electrical power and circuitry. A brief description and an analysis of all six channels are provided below.

(1) The multirange linear operation channel detects, displays, and records the reactor power level from 1 mW to 1 MW. The channel measures the output of a fission detector positioned just above the reactor core. The output from the channel is recorded on the console chart recorder in the control room during steady-state operation.

(2) The wide-range log operation channel detects, displays, and records the reactor power level from 1 mW to 1 MW by monitoring the output of the same fission detector used by the multirange unit. This wide-range log channel circuitry is provided with a number of set-point bistable trips, including source count-level interlock, power-level set point for mode switching, loss of high voltage to the detector, and "period rod withdrawal prevent." In addition, the wide-range log channel is provided with an output indicating the rate of change of power level (period meter).

(3) High-flux safety channel #1 is a linear power channel measuring the reactor power level as indicated by an ion chamber detector placed in the neutron field. A builtin scram provision is included in the circuit which trips at a power level of 1.1 MW.

(4) High-flux safety channel #2 is an independent power channel which operates identically to high-flux safety channel #1. This channel is also provided with scram logic if the 1.1 MW signal is initiated. Moreover, channel #2 is also used to measure and record peak output power and energy generation under pulse conditions. This is accomplished by placing a separate detector on the channel input.

(5) Fuel temperature safety channel #1 consists of three chromel-alumel thermocouples, an amplifier module board, a bistable trip, and a panel read-out meter. An automatic scram signal is provided by high fuel temperature or when the front panel switch is in any position other than "operate." Moreover, this temperature channel output is recorded when the reactor is operated in the pulse mode.

(6) Fuel temperature safety channel #2 is identical to fuel temperature channel #1 except that the temperature is not recorded on the console chart recorder during pulse operations.

7.6 Additional Nuclear Instrumentation

In addition to the nuclear instrumentation associated directly with reactor operation, the AFRRI facility is equipped with a wide variety of radiation monitoring systems to help ensure the safety of occupational personnel and the public. These monitoring systems are discussed below.

7.6.1 Remote Area Monitoring System

This system is designed to monitor the radiation levels in various areas within AFRRI where potential radiation hazards may exist during reactor operation. There are nine such units, and all are gamma detectors.

The units can detect gamma radiation with energy greater than 20 keV with a response time of less than 2 seconds. The meter and alarm response time is less than 1 second. The units are rugged and require a minimum of preventive maintenance. Seven of the detectors alarm at high radiation levels and upon failure. The deployment of these monitors is shown in Table 7.1.

The remaining two remote area monitors (RAMs) differ from those in Table 7.1 in that each is equipped with three visual level indicators and one audible alarm, and all associated circuitry and read-out are contained in a single unit. Their range is 0.05 to 50 mrads/hr, and they may be used to monitor gamma radiation background anywhere within AFRRI. Their designations are RAM E-4 and RAM E-5. Their current locations are in the preparation area on the walls adjacent to the ER #1 and ER #2 shield doors, respectively.

Table 7.1 Deployment of Remote Area Monitors (RAMs)

Monitor No.	Range (m rad/hr)	Location
RAM R-1	1 to 10^6	Reactor room, east side of pool, approximately 2 ft above the water level.
RAM R-2	1 to 10^5	Reactor room, west wall, approximately 7 ft above the floor.
RAM R-3	1 to 10^5	Reactor room, west wall, approximately 7 ft above the floor, adjacent to the reactor room door.
RAM R-4	1 to 10^5	Hallway 3105, east wall, 6 ft above the floor.
RAM E-3	1 to 10^5	Preparation area, west wall, approximately 6 ft above the floor, opposite the ER #1 shield door.
RAM E-6	1 to 10^5	Preparation area, west wall, approximately 6 ft above the floor, opposite the ER #2 shield door.
RAM-STACK	1 to 10^5	Room 3152, approximately 6 ft above the floor, on the outside of the air exhaust stack from the reactor building.

7.6.2 Continuous Air Monitors

A total of five continuous air monitors (CAMs) are used to sample and monitor the gross gamma-beta activity of the airborne radioactive particulate matter in various areas within AFRRI. All the units are equipped with audible and visual alarms, adjustable alarm settings, and a range of 50 cpm to 50 x 10^3 cpm. The reactor room CAM also provides readout and alarm in the control room and a trip signal to close the room dampers.

7.6.3 Stack Effluent Monitoring System

This system consists of three subsystems. Because air from all parts of the AFRRI complex is exhausted through the one stack, the monitors also apply to total complex.

(1) Stack Flow Monitoring System

This system consists of two flow-measuring Pitot tubes, two pressure gauges, two linear variable differential transformers, and a dual pen recorder. The sensing unit and transformers are mounted near the top of the stack. The Pitot tubes are positioned to give an average air flow. There are alarms associated with this system, as noted in Section 6.2.

(2) Stack Particulate Monitoring System

This system consists of an air filtration (TRAP) system coupled with a Geiger-Mueller detector used to monitor gamma-beta radiation emitted by radioactive particles trapped in the filter. Two alarm set points are provided: high level and detector system failure.

(3) Stack Gas Monitoring System

This system is used to measure the activity of gaseous radioactive nuclides that are exhausted from the AFRRI complex. The system has several related alarms including high-radiation level, system failure, effluent-sample low flow, propane gas low pressure, "pump off," and "high voltage off."

7.6.4 Criticality Monitors

AFRRI has operating two types of units which would be used to measure radiation emitted in the event of a criticality accident. However, as discussed in Sections 4.7 and 14, there is no credible reactivity excursion which would lead to a release of radioactivity to the environment. Therefore, these monitors are intended to assess occupational exposures in case of an unforeseen event.

7.7 Experiment Instrumentation

Additional instrumentation at AFRRI consists of electrical patch panels from which instrumentation signals can be transmitted between ER #1 and ER #2, ER #1 and ER #2 to the preparation area, and ER #1 and ER #2 to the control room and other areas of AFRRI.

7.8 Conclusion

The control and instrumentation systems at AFRRI are well designed and maintained. The various monitoring units and electromechanical interlocks provide operations personnel with timely information about the facility, and have a wide variety of builtin safety options. The staff considers that the control and instrumentation systems, coupled with administrative devices such as the facility component check list (which is used to verify the operability of key facility components before bringing the reactor to power), are adequate for safe operation of the facility. They also can reasonably be expected to be maintained so that operations comply with the Technical Specifications.

8 ELECTRICAL POWER SYSTEM

Electrical power is supplied to AFRRI from the NNMC power systems. Two transformers are located inside the reactor building. One transformer supplies the low-voltage circuitry such as lighting, wall receptacles, and the reactor console. The second transformer supplies higher voltage power to such components as the compressor, fan, and pump motors.

The staff concludes that no emergency backup power system is needed for reactor control systems because any electrical interruption results in a safe reactor shutdown and natural convection of the reactor pool water through the core prevents damage to the reactor fuel cladding (see Chapter 14).

9 AUXILIARY SYSTEMS

9.1 Systems Summary

The auxiliary systems considered are the fuel-handling and storage system, the compressed air system, as well as the provisions for fire protection system.

9.2 Fuel Element Handling and Storage

Fuel element handling is done with a long-handled tool that grips the fuel elements and enables the operator to relocate elements within the core, to inspect elements, or place them in or remove them from the storage racks. The inspection tool measures fuel-element bow and change in length. These measurements are performed in the pool approximately 9 ft under the surface. Six storage racks are located within the pool, at the side walls. Each rack can store up to 12 fuel elements at a minimum of 9 ft below the pool surface.

9.3 Compressed Air Systems

Two compressed air systems are associated with the AFRRI reactor facility. Both systems consist of a compressor, pressure regulators, valves, associated piping, and air-drying provisions. The first system supplies air to the transient control-rod system and the reactor pool shield-door bearings. The air supply to these bearings impedes the leakage of water into the bearing housing if the bearing seals fail.

The second air system supplies compressed air to the pneumatically operated reactor room ventilation system dampers discussed in Section 6.2. When the reactor room is isolated either manually or by a high-radiation alarm, the damper-actuating cylinders are vented and the dampers are closed by spring action.

9.4 Fire Protection System

The fire protection system consists of portable fire extinguishers located throughout the reactor building and three 6-in. hydrants with a capacity of up to 1000 gpm located outside the building. Combustible material loading within the reactor areas is minimized. An AFRRI-complex fire alarm system warns personnel for evacuation, and automatically summons the NNMC fire department. Personnel of the fire department are trained to anticipate and respond to radiation hazards.

9.5 Conclusion

The staff concludes that these auxiliary systems are adequate to support the AFRRI reactor complex in a safe and reliable manner.

10 EXPERIMENTAL PROGRAMS

10.1 Summary

The AFRRI reactor serves as a source of ionizing and neutron radiation for research and isotope production. Increased flexibility of this facility is achieved by the horizontally moveable reactor core that can travel from one irradiation position to another and be positioned adjacent to either of two large exposure rooms.

10.2 Experimental Facilities

10.2.1 Exposure Rooms

The exposure rooms are located on the first level of the reactor building at opposite sides of the reactor tank (see Figures 4.1 and 4.2). A semi-cylindrical section of the reactor tank protrudes into each exposure room. With the reactor core in an extreme position (adjacent to either exposure room) only about 1 in. of water is between the core shroud and the inside surface of the reactor tank.

ER #1 has a volume of slightly more than 3000 cu.ft. This room has a cadmium-gadolinium (Cd-Gd) shield attached to the tank projection to reduce the thermal neutron leakage from the reactor core into the exposure room, which helps to reduce Ar-41 production. This room also has lead curtains and movable shields to reduce gamma radiation, thereby enhancing the fast neutron-to-gamma ratio desirable for many experiments.

ER #2 has a volume of about 1800 cu.ft. and provides space for experiments requiring higher levels of gamma and thermal neutron radiation. The principal disadvantage of the use of ER #2 is its greater Ar-41 production rate, as the lack of a Cd-Gd shield on the tank results in a more intense thermal neutron flux density.

These exposure rooms allow the irradiation of large experimental and test equipment. Safety features include heavy shield doors interlocked into the reactor operate/ prevent system, automatic alarms, and manual emergency stop buttons.

10.2.2 In-Core Experiment Tube

The In-Core Experiment Tube (CET) provides an exposure facility with a characteristic high thermal neutron flux density that is primarily used for the production of radioisotopes and the activation of small samples for subsequent analysis. The CET may be positioned in any core lattice location. It is an air-filled aluminum guide tube with a 1-5/16 in. inner diameter. A nipple sealed to the lower end fits into the lower grid plate in a core fuel element location. The tube extends through the upper grid plate in the same grid array position. The CET has an S-bend above the upper grid plate to prevent radiation streaming and terminates at the reactor carriage.

10.2.3 Pneumatic Transfer System and Hot Cell

The pneumatic transfer system allows sealed samples to be rapidly transported between the reactor pool and the radiochemistry laboratory. Some of the transfer tubes have the additional capability of diverting irradiated samples from the normal sender-receiver station in the radiochemistry laboratory to the adjacent hot cell. The system consists of two banks of four tubes each. The pneumatic transfer system is made up of a blower, an absolute filter, eight send-receive stations, eight transfer tubes, a common air line, eight solenoid valves, and two four-way control valves. The irradiation termini are located just inside the reactor tank near ER #2.

10.2.4 Pool Irradiations

The open tank of the reactor permits the irradiation of experiments submerged in the vicinity of the core. Eighteen holes in the upper grid plate also allow small samples to be inserted into the core region for irradiation. The decision to perform experiments in the reactor pool--as opposed to the pneumatic transfer system or the CET--is dictated by specimen size and the desired type and intensity of radiation fields. The actual placement of experiments or samples in the core region or the reactor pool is limited by the Technical Specifications.

10.3. Experimental Review

10.3.1 Reactor Safety Review

Before any new experiment can be conducted using the reactor or experimental facilities, it is reviewed by the AFRRI Reactor and Radiation Facility Safety Committee. If it is anticipated that an experiment might result in a reactivity change of $\pm$ 0.5% $\Delta k/k$, actual k-excess measurements must be made at the core position where the experiment is to be located, both with and without the experimental device inserted, to determine its reactivity worth.

10.3.2 Radiation Safety Review

The review and approval process for experiments allows personnel specifically trained in radiological safety and reactor operations to consider and recommend alternative operational conditions--such as different core positions, power levels, and irradiation times--that will minimize personnel exposure and/or the release of radioactive materials to the environment.

10.4 Conclusion

The staff concludes that the design of the experimental facilities, combined with the detailed review and administrative procedures applied to all research activities, is adequate to ensure that experiments (1) are not likely to fail, (2) are unlikely to release significant radioactivity to the environment directly, 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

11.1 Summary

The major radioactive waste generated by reactor operations is activated gases, principally Ar-41 (1-1). A limited volume of radioactive solid waste, primarily resins, is generated by reactor operations, and some additional solid waste is produced by the associated research program. Very small amounts of radioactive liquid waste are generated by normal reactor operations. In addition, small amounts of radioactive liquid waste are developed as a result of several of the institute's research activities which do not use the reactor.

11.2 ALARA Commitment

The AFRRI operates with the philosophy of minimizing the release of radioactive material to the environment. Its administration instructs all research personnel to develop procedures to limit the generation and subsequent release of radioactive waste materials.

11.3 Waste Generation and Handling Procedures

11.3.1 Solid Waste

Solid waste generated as a result of reactor operations consists primarily of ion exchange resins and filters, potentially contaminated paper and gloves, and occasional small activated components.

Some of the reactor-based research results in the generation of solid low-level radioactive waste in the form of contaminated paper gloves, glassware, and animal bedding. This solid waste generation has typically contained a few millicuries of radionuclides per year.

During the 1960s, the NNMC operated an incinerator system, and occasionally provided service to AFRRI in disposing of biomedical preparations. NNMC no longer provides this service.

The solid waste is collected by the health physics staff and held temporarily before being packaged and shipped to an NRC-approved disposal site in accordance with applicable NRC and Department of Transportation regulations.

11.3.2 Liquid Waste

Normal reactor operations produce no radioactive liquid waste. However, many of the research activities conducted within the AFRRI complex are capable of generating such waste. Liquid waste drains in the reactor room and equipment areas drain into holdup tanks; thus, there is no direct flow into the NNMC sanitary sewer system. Other laboratories and experimental areas in the AFRRI complex where radioactivity may be used are also provided with waste lines that flow into these holdup tanks. All potentially contaminated liquids are collected in these holdup tanks. When nearly full, the individual tanks are isolated, mixed, and sampled. The sample is dried, and the residue is analyzed for radioactive content by standard techniques. If the concentrations of radioactive material in the tank are less than the levels specified by 10 CFR Part 20, the contents are discharged to the sanitary sewer system. If the concentrations are initially above 10 CFR Part 20 levels, the contents of the tank are diluted to below those levels before discharge.

The facility also has the capability to solidify small volumes of highly contaminated liquid for shipment offsite as solid waste.

11.3.3 Airborne Waste

The potential airborne wastes are gaseous Ar-41 and neutron-activated particulate matter. No fission products escape from the fuel cladding during normal operations. The radioactive airborne waste is principally produced by the neutron irradiation of air and airborne particulate materials in the exposure rooms. This air is constantly swept from the exposure rooms and discharged to the environment through the AFRRI stack. The reactor building exhaust system, which includes the exposure room discharge, is equipped with a filter system that collects more than 99.9% of the particulate matter. These filters are eventually disposed of as solid potentially radioactive waste. A stack monitoring system measures the stack air flow rate and the radioactive particulate and gaseous concentrations in the effluent. During normal operations no measurable radioactive particulates have been released in the air effluents from the AFRRI stack.

AFRRI has measured the release of Ar-41 over the years with gas sampling instruments calibrated with known quantities of Ar-41. The staff has examined the AFRRI techniques and finds them acceptable (11-1). Therefore, the staff considers that the releases reported to NRC by AFRRI are reliable data which can be used to evaluate the potential impact in the local environment. During the years since the reactor was first licensed, AFRRI has reported an annual release of more than 40 curies of Ar-41 only three times, all in the 1960s. In the past 10 years, in accordance with the ALARA principles, AFRRI has modified its exposure rooms (gadolinium liners) and its choice of experimental facilities. Accordingly, the amount of Ar-41 discharged over the last 10 years has averaged less than 12 Ci/yr, and about 7.5 Ci/yr over the last 5 years. Furthermore, since the major building addition in about 1970, when the stack system and exhaust air flow were stablilized, the annual average concentration of Ar-41 in the air stream at the exit of the stack has never been significantly above the maximum permissible concentration (MPC) stipulated in 10 CFR Part 20 for unrestricted areas (1-1).

Using an assumed 40 curies of Ar-41 per year, AFRRI has employed conservative methods and assumptions accepted by NRC to compute the potential maximum whole-body-immersion dose at a distance of about 300 ft from the stack during a whole year (8736 hours). The computed dose is 2 mrems per year. According to the International Commission on Radiological Protection (ICRP), the dose due to inhalation of Ar-41 at MPC would be a very small fraction of the immersion dose, and within the principles of their guidance, can be neglected (11-2). (These ICRP recommendations over the years have formed the bases of 10 CFR Part 20). Because the natural tendency of airborne gases is to diffuse and decrease in concentration with distance from the source, the maximum yearly dose to members of the public at the nearest residence (assuming 100% occupancy and a distance of about 1000 ft) will be much less than the licensee's conservative estimate of 2 mrems.

The NRC staff has reviewed this computation, and compared the results with its own independent ones. The staff included the assumption that all of the Ar-41 is released at ground level, instead of at the top of the stack. The results of these computations give reasonable assurance that the doses quoted above would not be exceeded when averaged over a year.

In January 1979 an NRC inspector found a dry water trap in the AFRRI air exhaust system which could potentially release airborne Ar-41 at ground level (11-3). This was a violation of AFRRI's Technical Specifications. However, based on the staff evaluation above, even if all of AFRRI's Ar-41 for a full year were released through the open trap, the doses to the public 1000 ft from the AFRRI stack would still be below the doses quoted above (2mrems). Based on the relative diameters of the pipe to the trap and the exhaust stack, it is reasonably estimated that much less than 1 % of the Ar-41 was released through the groundlevel pipe while the water trap was open.

In the meantime, AFRRI has capped off the pipe to which the water trap was connected, so this malfunction is not possible in the future.

11.4 Conclusion

The staff concludes that the waste management activities of the AFRRI reactor facility have been conducted and are expected to continue to be conducted in a manner consistent with 10 CFR Part 20 and with the ALARA principles (11-1). Among other guidance, the staff review has followed the methods of ANSI/ANS 15.11, 1977, "Radiological Control at Research Reactor Facilities."

Because Ar-41 is the only potentially significant radionuclide released by AFRRI to the environment during normal operations, the staff has reviewed both the history, current practice, and future expectations. The staff concludes that the doses in unrestricted areas as a result of AFRRI's actual releases of Ar-41 have never exceeded--or even approached--the limits specified in 10 CFR Part 20 when averaged over a year. Futhermore, the staff's conservative computations of the dose beyond the limits of the NNMC give reason able assurance that potential doses to the public as a result of Ar-41 would not be significant, even if there were a major change in the operating schedule of the AFRRI reactor.

12 RADIATION PROTECTION PROGRAM

12.1 Summary

AFRRI has developed a structured radiation safety program with an adequate health physics staff and appropriate radiation detection equipment to determine, control, and document occupational radiation exposures. In addition, AFRRI monitors both liquid and airborne effluents at the points of release in order to comply with applicable regulations. AFRRI has also developed an environmental monitoring program to verify that radiation exposures in the unrestricted areas around AFRRI are well within regulations and guidelines, and to confirm the results of calculations and estimates of environmental impacts resulting from the AFRRI's research efforts.

12.2 ALARA Commitment

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

12.3 Health Physics Program

12.3.1 Health Physics Staffing

The normal full-time health physics staff consists of four professionals and six technicians. The staff has sufficient training and experience to direct the radiation protection program for a research reactor. This health physics staff has been given the responsibility, the authority, and adequate lines of communication to provide an effective radiation safety program (1-1).

The health physics staff provides radiation safety support to the entire AFRRI complex, including a linear accelerator and a large cobalt-60 (Co-60) irradiator. However, the staff believes that the AFRRI Health Physics staff is adequate for the proper support of the diverse research efforts within this facility. Additional personnel trained and experienced in radiation safety are available if needed.

12.3.2 Procedures

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

12.3.3 Instrumentation

AFRRI has acquired a variety of detecting and measuring instruments for monitoring any kind of potentially hazardous ionizing radiation (1-1). The instrument calibration procedures and techniques ensure that any credible type of radiation and any significant intensities will be promptly detected and correctly measured.

12.3.4 Training

All AFRRI employees 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 with radiation or radioactive materials. The training program is designed to identify the particular hazards of each specific type of work to be undertaken and methods to mitigate their consequences. Retraining in radiation safety is provided as well. As an example, all reactor operators are given an examination on health physics practices and procedures at least every 2 years. The level of retraining given is determined by the examination results. All of the above mentioned radiation safety training is provided by the health physics staff.

12.4 Radiation Sources

12.4.1 Reactor

Sources of radiation directly related to reactor operations include radiations from the reactor core, ion exchange columns, filters in the water and air clean-up systems, radioactive gases, primarily N-16 and Ar-41 and samples and components used in experiments. Additional radiation sources consist of irradiated fuel and radioactive wastes.

The reactor fuel is contained within stainless steel cladding. Radiation exposures from the reactor core are normally reduced to acceptable levels by water and concrete shielding, and personnel are not routinely allowed in the immediate vicinity of the reactor pool surface during high-power or pulsed operations.

Exposures from incore maintenance and fuel handling are minimized because much of the necessary work is performed under at least 9 ft of water. Highly activated components and spent fuel elements are removed from the reactor tank in adequately shielded casks.

The ion exchange resins and filters are routinely changed before high levels of radioactive materials have accumulated, thereby minimizing personnel exposure.

Concentrations of N-16 in potentially occupied areas of the reactor room are reduced by using the diffuser in the reactor tank to increase the time required for the gas to reach the surface of the water.

This allows the short half-life (7.1 seconds) of the N-16 to reduce further the amount of radioactivity released into the reactor high bay. Personnel exposure to the radiation from chemically inert Ar-41 is limited by dilution and prompt removal of this gas from the reactor room and its discharge to the atmosphere where it diffuses further before reaching occupied areas.

12.4.2 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 normal protective measures of time, distance, and shielding.

In addition to the reactor, AFRRI operates other major radiation sources, including a high performance electron linear accelerator in the same building structure that houses the reactor and a Co-60 irradiator. The accelerator is not under NRC licensing jurisdiction, but it produces small quantities of airborne radionuclides during routine operation. The cobalt irradiator is licensed by NRC under a license that is completely independent of the reactor.

12.5 Routine Monitoring

12.5.1 Fixed-Position Monitors

The AFRRI complex uses a number of fixed-position remote area radiation monitors (RAMs) and constant air particulate monitors (CAMs), as discussed in Section 7.6. The Technical Specifications require that several of these monitors be operational during reactor operations. These include two RAMs in the reactor room, one RAM outside each of the exposure rooms located to detect any significant radiation streaming, and a CAM in the reactor room near the reactor pool. All monitors have adjustable alarm set points. Many read out in the reactor control room, and alarm of the CAM that samples air above the reactor pool is interlocked to trip the reactor room ventilation system to isolate the area from the rest of the building. Several additional RAMs and CAMs are located in areas of potential radiation or airborne activity. The alarm set points of all such instruments have been set at sufficiently low levels to alert personnel promptly of unusual radiation conditions.

12.5.2 Experimental Support

The health physics staff participates in the planning of experiments by reviewing all proposed procedures for methods of minimizing personnel exposures and limiting the generation of radioactive waste. Approved procedures specify the type and degree of health physics involvement in each activity. As examples, standard operating procedures require that initial entry into either exposure room include a survey by health physics personnel using portable instrumentation, and all items removed from the exposure rooms must be surveyed and tagged by health physics personnel.

12.5.3 Special Work Permits

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

12.6 Occupational Radiation Exposures

12.6.1 Personnel Monitoring Program

The AFRRI personnel monitoring program is described in several Radiation Safety Instructions. To summarize the program, personnel exposures have been measured by the use of film badges assigned to individuals who might be exposed to radiation. Currently, thermoluminescent dosimeters (TLDs) are used for these measurements. In addition, self-reading pocket dosimeters are used, and instrument dose rate and time measurements are used to achieve administrative occupational exposure limits of 50 mrems for 1 day and 100 mrems for 1 week, while still complying with applicable limits in 10 CFR Part 20.

12.6.2. Personnel Exposures

The AFRRI personnel annual exposure history for the last 10 years (including personnel at the Uniformed Services University of the Health Sciences (USUHS) which is also on the NNMC grounds), are given in Table 12.1.

Table 12.1 No. of Individuals in Exposure Interval

Year	<0.5 rem	>0.5 but <1.0 rem	>1.0 but <3.0 rems
1971	224	3	3
1972	250	5	2
1973	260	0	8
1974	330	5	2
1975	600	2	0
1976	300	0	0
1977	385	0	0
1978	800	0	0
1979	810	0	0
1980	881	0	0

Most of the larger exposures in the early 1970s resulted from biomedical-related research projects that used large quantities of special short-lived radioisotopes that were not produced by the reactor and thus have no relation to reactor operation or usage (1-1).

12.7 Effluent Monitoring

12.7.1 Airborne Effluents

As discussed in Chapter 7, airborne effluents from the reactor facility consist of activated gases and radioactive particulate matter potentially generated during operations. The effluent stream is filtered to remove most particulate material before discharge to the environment through the AFRRI stack. The filter installation consists of a roughing filter to reduce the loading of the finer filters and a bank of high-efficiency particulate air (HEPA) filters that together remove most of the solid matter in the air stream.

12.7.1.1 Particulate Monitoring

After the air withdrawn from the reactor room and the experimental areas passes through the filtering system, a continuous representative sample is monitored for the presence of radioactive particulate material. This monitoring system consists of a small probe, with a sampling rate which is approximately isokinetic, located in the air discharge duct. This probe withdraws a small continuous air stream that is passed through a filter assembly having a "pancake" G-M detector that monitors the accumulation of trapped radioactive particulates. This particulate monitor can be read on a meter in the reactor control room. The filter media can be removed for laboratory analysis to identify specific isotopes and to determine average concentrations in the total effluent stream.

During normal operations only insignificant quantities of radioactive particulates can or do pass through the filter banks. Therefore, this is a monitoring system; it is not intended to measure routine releases to the atmosphere. Rather, its primary function is to provide prompt information to operating personnel of extreme abnormal conditions, such as a high production rate of radioactive particulates possibly combined with a filter failure.

12.7.1.2 Gaseous Monitoring

The stack gas monitoring system measures the radioactive gases discharged from the entire AFRRI complex. The principal radioactive gases are Ar-41, O-15, and N-13. The reactor is the primary source of Ar-41, whereas the linear accelerator is the source of the O-15 and N-13. The system consists of a multiport sampling probe positioned near the top of the stack, a sampling pump to maintain a constant flow, a filter to remove particulate contaminates, and a proportional counter sandwiched between two 10-liter chambers. The instrumentation read-out consists of a meter and strip chart recorder in the control room. The detector count rate is proportional to the amount of radioactive gases in the chambers and hence to the concentration in the air stream. High concentrations and detector failure activate alarms in the control room. This gaseous monitoring system is periodically calibrated by releasing a small known quantity of Ar-41 into the stack effluent stream.

12.7.2 Liquid Effluent

The reactor itself generates no radioactive liquid effluents during routine operations. However, leaks in the primary coolant system do have the potential for being released, and experimental activities associated with reactor usage also may generate radioactive liquids. All potentially contaminated liquids are collected in holdup tanks. Before release, each tank is sampled and analyzed, and liquids with low concentration of radioactivity are released directly to the sanitary sewer in accordance with 10 CFR 20.303. Higher concentrations of liquid waste may be diluted for release, held for radioactive decay, or they may be solidified and handled as solid waste.

12.8 Environmental Monitoring

AFRRI has developed a program to monitor radiation exposures above background in the surrounding environment from both reactor operations and the research efforts of the complex.

12.8.1 External Radiation Measurements

A perimeter monitoring system of about 30 stations has been established in the unrestricted areas around AFRRI on the NNMC grounds, and there are at least 6 stations on the AFRRI complex roof. An additional five monitoring stations have been established several miles away to obtain the general area background. This measured natural background radiation level is then subtracted from the individual measurements made near the AFRRI complex to determine the magnitude of any penetrating radiation that may have originated from the AFRRI or NNMC complexes. Since initial reactor licensing, film dosimeters provided by a commercial supplier were used to measure the external radiation exposures around the AFRRI complex. TLDs are currently used. These are of equal sensitivity to the film dosimeters but do not fade to the same extent, which both allows llonger monitoring intervals and improves the accuracy of the individual measurements.

12.8.2 Environmental Sampling Program

Samples are collected from the stream (surface water) that flows near the AFRRI complex, the soil from the hillside below the liquid waste tanks, and the vegetation from the area between AFRRI and the USUHS on a quarterly basis, (1-1). In addition, samples of silt or sludge, rainwater or snow, and airborne particulates are collected. These samples are concentrated and then examined for the presence of radioactive material.

Using state-of-the-art procedures and techniques, concentrations of radioactive materials in the environmental samples collected in the vicinity of the AFRRI complex have consistently been indistinguishable from levels found in samples collected several miles from the facility.

12.9 Potential Dose Assessments

Natural background radiation levels in the Washington, D. C., area result in an exposure of about 80 mrems/yr to each individual residing there. At least an additional 10% (approximately 8 mrems/ yr) will be received by those living in a brick or mansonry structure. Medical diagnosis exposures may add to this natural background.

Conservative calculations by the NRC staff based on the amount of reactor-related Ar-41 released from the AFRRI stack predict a maximum annual dose of less than 1 mrem in the unrestricted areas. The results of the environmental radiation dosimeters (film or TLD) located on the NNMC grounds have averaged less than 3 mrems/yr for the last 10 years. The average of the highest individual readings for the last 10 years is less than 15 mrems/yr. These monitors are sensitive to all penetrating gamma- and x-ray radiation.

During the 1960s AFRRI operated an x-ray facility in support of its research program, and a nearby perimeter monitoring station consistently gave a reading much higher than any other, or the average of the others, in the perimeter monitoring set. Both because of the proximity of the x-ray lab and because there is no credible way the reactor airborne effluents could always flow towards that station, it is concluded that readings at that detector station were not related to reactor or other NRC-licensed operations (11-1).

12.10. Conclusion

The staff considers that radiation protection receives appropriate support from the administration. The staff concludes that (1) the program is properly staffed and equipped, (2) the AFRRI staff has adequate authority and lines of communication, and (3) the procedures are correctly integrated into the research plans.

The staff concludes that the effluent and environmental monitoring programs conducted by AFRRI personnel are adequate to promptly identify significant releases of radioactivity and confirm possible impacts on the environment, as well as to predict maximum exposures to individuals in the unrestricted area. These predicted maximum levels are well within applicable regulations and guidelines of 10 CFR Part 20.

Additionally, the staff concludes that the AFRRI 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 during the requested renewal period.

13 CONDUCT OF OPERATIONS

13.1 Organization Structures and Qualifications

Organization and qualifications are discussed in Chapter 7 of the applicant's Safety Analysis Report.

13.1.1 Overall Organization

Responsibility for the safe operation of the reactor facility is vested within the chain of command shown in Figure 13.1. The Director, AFRRI, is the individual responsible to NRC for complying with the license. Because AFRRI is a command of the Defense Nuclear Agency (DNA), the AFRRI Director is subject to the management policies of the Director, DNA, who is assisted by a Board of Governors. In addition to ensuring that the reactor can perform its mission, the AFRRI management is responsible to ensure that both the public and employees are protected from radiation exposures, according to applicable regulations and the reactor license. The Director maintains an independent radiation safety staff to assist him in that function.

13.1.2 Reactor Staff

The reactor branch is comprised of the physicist-in-charge, the reactor operations supervisor, and several reactor operators.

13.2 Training

Most of the training of reactor operators is done by inhouse personnel, but some training has been obtained at the nuclear engineering department of a local university. The applicant's Operator Requalification Program has been reviewed, and the staff concludes that it meets applicable regulations (10 CFR 50.34(b)).

13.3 Emergency Planning

10 CFR Part 50, Appendix E requires that nonpower reactor licensees develop and submit Emergency Plans. The applicant (licensee) submitted a Plan which was developed following the recommended guidance in Regulatory Guide 2.6 (1979, For Comment issue) and guidance in ANS 15.16 (1978 Draft). However, both of these guides are currently being revised, with a final version of Regulatory Guide 2.6 due to be issued in March 1982. The staff's review of the applicant's Emergency Plan is in progress. Upon completion of that review, the staff will issue an appropriate supplement to this Safety Evaluation Report.

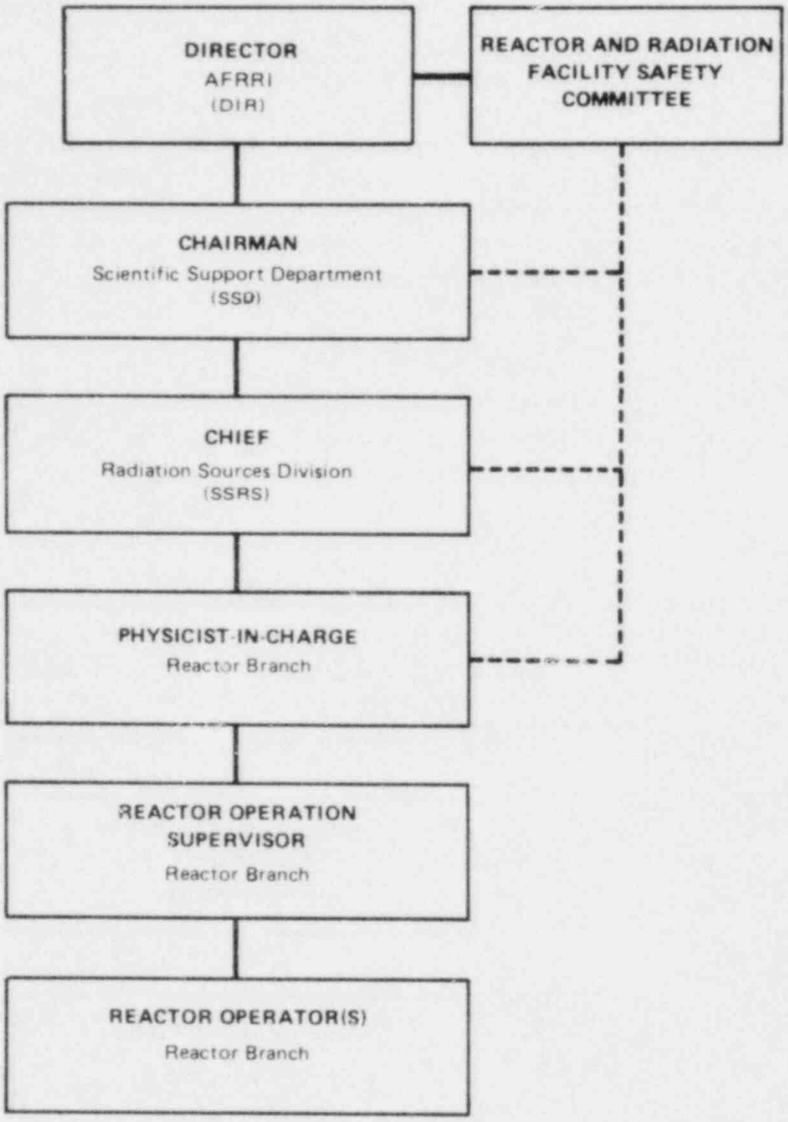

Figure 13.1 Reactor operations chain of command

13.4 Operational Review and Audit

In addition to the line staff for reactor operations and the industrial and radiation safety staff reporting to the AFRRI Director, AFRRI has a committee which oversees the facility operations. This committee consists of some reactor operations personnel, but it also includes qualified people from other parts of the AFRRI staff and non-AFRRI experts in radiological and reactor technologies. The Reactor and Radiation Facility Safety Committee is responsible for reviewing the other major radiation facilities at AFRRI, in addition to the reactor. The committee must review and approve plans for modifications to the reactor, new experiments, and proposed changes to the license or to procedures. The committee is also responsible for arranging for and reviewing audits of reactor facility operations and management, and for reporting the results of them to the Director.

13.5 Physical Security Plan

AFRRI has established and maintains a program designed to protect the reactor and its fuel, and to ensure its security. The NRC staff has reviewed the plan and visited the AFRRI site, and concludes that the plan, as amended, meets the requirements of 10 CFR 73.67 for special nuclear material of low strategic significance. AFRRI's licensed authorization for reactor fuel falls within that category. Both the Physical Security Plan and the staff's evaluation are withheld from public disclosure under 10 CFR 2.790(d)(1) and 10 CFR 9.5(a)(4). Amendment 17 to facility license R-84, dated February 10, 1981, incorporated the Physical Security Plan as a condition of the license.

13.6 Common Defense and Security

The Armed Forces Radiobiology Research Institute (AFRRI) is a command of the Defense Nuclear Agency (DNA) of the U.S. Department of Defense. By charters, both AFRRI and DNA are directed by military officers of the U.S. armed services. Therefore, the staff concludes that renewal of the license for continued operation of the AFRRI reactor will not be inimical to the common defense and security.

13.7 Conclusion

Based on the above discussions, 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 health and safety of the public.

14 ACCIDENT ANALYSIS

14.1 General Summary

As part of its evaluation of several pending license renewals for nonpower reactors, the staff asked one scientific laboratory to analyze generic reactor accidents for uranium-zirconium hydride fueled reactors (14-1) and a second scientific laboratory to evaluate the licensee's submitted documentation. These analyses included the likelihood of various types of accidents and the potential consequences to the public.

Among the potential accidents considered to be credible, the one with the greatest impact on the environment and the unrestricted area outside of the AFRRI building complex is the loss of cladding integrity of one irradiated fuel element in air in the reactor room. For purposes of classification, the staff will call this the "fuel handling accident." In Chapter 7, and in more detail below, the staff has evaluated possible accident scenarios originating in the intact core. None of these pose a significant risk of clad failure. However, it is possible that an operator, in removing a fuel element from the core or in relocating one previously removed following irradiation, could have an accident that would breach the integrity of the cladding. If the cladding were ruptured, noble gas and iodine fission products could escape into the environment.

As discussed in more detail below, this will be designated as the design-basis accident (DBA). A DBA is defined as an accident for which the risk to the public health and safety is greater than that from any event that can be mechanistically postulated. Thus, the staff assumes that the accident occurs, but does not try to describe or evaluate the mechanical details of the accident, or the probability of its occurrence. Only the consequences are described.

14.2 Accidents Analyzed

The following potential accidents or effects were considered to be sufficiently credible to evaluate:

(1) rapid insertion of reactivity (nuclear excursion)
(2) loss of coolant
(3) metal-water reactions
(4) misplaced experiments
(5) mechanical rearrangement of the fuel
(6) effects of fuel aging
(7) handling of irradiated fuel

14.2.1 Rapid Insertion of Reactivity (Nuclear Excursion)

The potential event evaluated is one in which all of the excess reactivity authorized under AFRRI's license conditions is inserted into the reactor instantaneously. However, the staff has not been able to find a credible method for inserting all of this excess reactivity "instantaneously." Both the theory of the neutronic behavior of the uranium-zirconium-hydride fuel and all experimental measurements have shown that this fuel exhibits a strong, prompt, negative temperature coefficient of reactivity. This coefficient derives from the bonding of the hydrogen to the zirconium, and as long as bonding exists, a nuclear excursion is terminated in a self-limiting transient. Various investigators have determined that at temperatures above approximately 1100°C, some local breaking of the bond and consequent dehydriding may occur (14-1). However, if most of the fuel volume is below this temperature, not only does the temperature coefficient terminate a nuclear excursion, but it also causes a loss of reactivity as the steady-state temperature of the fuel is raised. Experimental demonstrations of these results have been verified at many operating reactors using uranium-zirconium-hydride fuel. Thus, through the action of the inherent temperature coefficient, temporary loss of positive reactivity will be caused both by steady-state operation and by pulsing.

Therefore, the staff has considered the scenario of the reactor operating at some steady-state power level between 0 and 1 MW, and then all of the remaining excess reactivity not compensated by increased temperature being inserted rapidly. The staff has found, within the range of reactivity authorized at AFRRI, that the higher the temperature at which the rapid insertion is initiated, the lower the final temperature of the fuel immediately after the transient. This evaluation assumed that all loss of reactivity during the steady-state operation was due to the increase in temperature of the fuel. Thus, the known effect of xenon-135 was ignored. Therefore, the staff has assumed the worst case: initiating the transient with the core at ambient water temperature and zero initial power.

The potential significant consequences of transient heating of the fuel that the staff has considered are: melting, loss of hydrogen bonding, and failure of fuel cladding as a result of high internal gas pressures. Because the maximum authorized reactivity available in the AFRRI reactor is 3.5% $\Delta k/k$, the staff has reviewed the literature for transients with at least this amount of excess reactivity for a reactor core similar to AFRRI's. General Atomic has performed many experiments with reactivity insertions of this size in an 85-element TRIGA core. They measured, among other parameters, the temperature of fuel in the hottest core position, and they examined fuel elements afterwards (14-2,14-3). There was no indication of undue stress in the cladding, and no indication of either cladding or fuel melting. The measured maximum temperature for the 3.5% $\Delta k/k$ pulse was approximately 750°C, and the estimated peak transient temperature at any localized point in the fuel was 1175°C. Because this estimated transient temperature is localized on the periphery of only the hottest fuel elements immediately after the pulse, before a significant amount of heat transfer within the zirconium hydride redistributes it, this temperature will start to decrease within seconds. The local temperature of 1175°C is in the region of the dehydriding temperature of the fuel alloy, but well below the melting temperature of zirconium (>1800°C). However, (1) the temperature coefficient of reactivity was effectively unchanged and still functioned to terminate the nuclear excursion, and (2) there was insufficient hydrogen released to raise the gas pressure within the intact cladding to stress it near its elastic limit or yield point (14-4). Furthermore, the excess reactivity insertion assumed for the transient in the AFRRI accident scenario requires the presence of both the moderator and reflector water, so during the transient, all of the fuel would be totally immersed in water at its initial temperature. Therefore, the cladding would be cooled continuously, and its temperature would remain well below the hottest fuel temperature.

Additionally, data giving the fuel-temperature history following a large pulse (transient) demonstrate that natural convective water cooling of the fuel lowers its temperature several hundred degrees celsius within 2 minutes of the transient (14-4). Hence, if the ambient water is present at least that long after the pulse, most of the pulse energy will have been transferred from the fuel to the water.

14.2.1.1 Conclusion

From the above considerations, the staff concludes that there is no credible nuclear excursion possible with the AFRRI reactor that could lead to fuel melting or cladding failure resulting from high temperature or high internal gas pressure. Therefore, there is reasonable assurance that fission product radioactivity will not be released from the fuel to the environment as a result of a reactor pulse or transient.

14.2.2 Loss of Coolant

A potential accident that would result in increases in temperatures of the fuel and cladding is the loss of water coolant shortly after the reactor has been operating. Because the water is required for adequate neutron moderation, its removal would terminate any significant neutron chain reaction. However, the residual radioactivity would continue to deposit heat energy within the fuel. It is assumed that sufficient water is lost to uncover the core and that subsequent heat removal from the fuel is provided only by air convection. Several investigations have evaluated such scenarios, under various assumptions (14-5). In the AFRRI reactor, the core will be completely immersed in water as long as the level of the water is at least 5 feet above the tank bottom. That would require about 3000 gal of water in the tank. Therefore, about 12,000 gal could be removed before the core is uncovered. If it is assumed that a gross constant leak of 500 gpm occurs, the core would remain covered for at least 24 minutes. If convective water cooling continued that long, for a core that had been operating at 1 MW long enough to achieve fission product equilibrium (to be conservative), the peak temperature that the fuel would reach would be less than 950°C. This maximum temperature would not be reached for about 3 hours. Not only would this maximum temperature not rupture the fuel cladding, but the time scale for the entire event would allow for remedial action.

Section 14.2.1 addresses the dependence of pulse size and the ultimate maximum fuel temperature on the temperature at which the transient is initiated. Accordingly, it would be physically impossible in the AFRRI reactor to produce a large pulse at the end of an extended operation at 1 MW steady state unless the fuel temperature were first lowered to approximately that of the ambient water. Then, for the transient to contribute substantially to the fuel heat content after the loss of coolant, the transient would necessarily have to occur within about 2 minutes of the time that the core becomes uncovered. The staff has been unable to develop a realistic accident scenario which includes all of these assumptions.

14.2.2.1 Conclusion

If all water were lost from the region of the core, the reactor would become subcritical and the production of additional fissions would cease; therefore, only the heat due to fission product beta and gamma rays need be considered.

If the reactor were pulsed shortly after an extended run, the heating due to the additional inventory of fission products would be negligible. Furthermore, as indicated in Section 14.2.1, the fuel temperatures must necessarily be reduced to water ambient before a pulse of any significant size could occur. Therefore, sufficient water would still be present to provide cooling following the pulse. Accordingly, the staff concludes that a relatively rapid loss of coolant from the reactor tank following extended operation at 1 MW would not result in fuel or clad melting or loss of cladding integrity.

14.2.3 Metal-Water Reactions

Chemical reactions, especially oxidation, may occur if sufficiently hot metal is brought into contact with water. This has been an area of concern and study in designing reactors since the early 1950s. Therefore, there is an extensive body of literature on the subject (14-6). From the laboratory tests, it is concluded that the metal (reactor fuel) would have to be heated to very high temperatures (for example, above the melting point) and/or be fragmented into small hot particles and injected into water in order to support a rapid (explosive) chemical reaction. Either of these conditions implies a prior catastrophic event of some sort, which presumably would have to originate with a nuclear excursion or loss of coolant. In Sections 14.2.1 and 14.2.2 these events were shown to be not credible in a 1-MW uranium zirconium-hydride-fueled reactor like the one authorized for operation at AFRRI.

Additionally, some of the studies discussed in reference 14-6 include metal-air and metal-steam chemical reactions. Violent (explosive) reactions do not appear to be possible in air or steam at atmospheric pressure, even though rapid reactions may occur at sufficiently high temperatures with specially prepared samples and conditions. As for the possible metal-water reaction, a prior cataclysmic event would be necessary even to approach those conditions, and the discussions in Section 14.2.1 and 14.2.2 show that such an event is not credible.

In addition to the investigations referenced above, General Atomic has experimentally plunged heated samples of unclad zirconium hydride into water to examine possible conditions for initiating and sustaining a metal water reaction (14-7). Up to temperatures of about 1200°C, there was no chemical reaction of the metal, except for the formation of a relatively inert oxide film. Furthermore, in the hottest unclad test samples, most of the hydrogen may have been driven off, so the metal surface in contact with the water would have been mostly zirconium.

14.2.3.1 Conclusion

Based on the above considerations, the staff concludes that there is reasonable assurance that rapid (violent) metal-water, metal-air, or metal-steam reactions will not occur in a reactor of the TRIGA-type, operating at 1 MW or below, with maximum available excess reactivity as authorized at AFRRI.

14.2.4 Misplaced Experiments

This type of potential accident is one in which an experimental sample or device is inadvertently located in an experimental facility where the irradiation conditions could exceed the design specifications. In that case, the sample might become overheated or develop pressures which could cause failure of the experiment container. As discussed in Chapter 10, all new experiments at AFRRI are reviewed prior to insertion, and all experiments in the region of the core are separated from the fuel cladding by at least one barrier, such as the pneumatic transfer tube, the incore experiment tube, or the core shroud.

14.2.4.1 Conclusion

The staff concludes that the experimental facilities and the procedures for experiment review at AFRRI are adequate to provide reasonable assurance that failure of experiments is not likely, and, even if failure occurred, breaching of the reactor fuel cladding will not occur. Furthermore, if an experiment should fail and release radioactivity within an experimental facility, there is reasonable assurance that the amount of radioactivity released to the environment would not be more than that from the accident discussed in Section 14.2.7.

14.2.5 Mechanical Rearrangement of the Fuel

This type of potential accident would involve the failure of some reactor system, such as the support carriage, or could involve an externally originated event which disperses the fuel and, in so doing, breaches the cladding of one or more fuel elements. The staff has not developed scenarios for accidents such as these, because there have never been any at nonpower reactors. Thus there is no logical basis for deciding if any arbitrary scenario is credible. Instead, a later section of this chapter discusses a scenario assuming the failure of the cladding of an element after extended reactor operation and evaluates possible doses due to various hypothetical scenarios for release of the inventory of radioactivity. This approach should address the spectrum of fuel clad failures. (The scenario in which the initiating event causes a rearrangement of the fuel in such a way that all of the control rods are somehow simultaneously ejected from the core and a nuclear excursion results is encompassed in Section 14.2.1.)

14.2.5.1 Conclusion

The staff concludes that no mechanical rearrangement which is credible would lead to an accident with more severe consequences than those accidents considered in Sections 14.2.1 or in 14.2.7.

14.2.6 Effects of Fuel Aging

The staff has included this process in this section so all credible effects are addressed. However, as discussed in more detail in Chapter 17, fuel aging should be considered normal with use of the reactor and is expected to occur gradually. The reactions external to the cladding that might occur are addressed in Chapter 17. In this section the possibility of internal reactions is discussed. There is some evidence that the uranium-zirconium hydride fuel tends to fragment with use, probably due to the stresses caused by high temperature gradients and high rate of heating during pulsing (14-8). Some of the possible conseqences of fragmentation are: (1) a decrease in thermal conductivity across cracks, leading to higher central fuel temperatures during steady-state operation (temperature distributions during pulsing would not be affected significantly by changes in conductivity because a pulse is completed before signficant heat redistribution by conduction occurs) and (2) fragmentation would allow more fission products to be released into the cracks in the fuel. However, it is not expected that this increase would be large, when the two mechanisms for release are considered. At temperatures above about 400°C, diffusion of the noble gases accounts for a large fraction of the release to the gap. The fragmentation of the fuel would allow diffusion to the nearest surface to occur more rapidly, but there is no apparent reason to expect a larger ultimate release. The other mechanism, low temperature emission from a surface layer into a crack, might increase because of more "gaps," but, on the other hand, the principal gap between clad and fuel almost certainly must become smaller if the fuel body fragments and expands. Furthermore, the cracks would not separate very far, so most fission products would impinge on to the opposite surface and then have to diffuse back out to be released into the gaps.

14.2.6.1 Conclusion

The staff concludes that the two likely processes of aging of the uranium-zirconium-hydride fuel-moderator would not have a significant effect on the operating temperature of the fuel or on the accumulation of gaseous fission products within the cladding. Therefore, the staff also concludes that there is reasonable assurance that fuel aging will not significantly increase the likelihood of fuel-cladding failure, or the quantity of gaseous fission products available for release in the event of loss of cladding integrity.

14.2.7 Handling Irradiated Fuel

This potential accident includes various incidents to one or more fuel elements, with the reactor shut down, in which the fuel cladding might be breached or ruptured. In order to be general, the staff let the scenario include the time scale from immediately after a long run at full licensed power to any longer time, associated for example with moving stored irradiated fuel from a rack in the pool into the reactor room. Also to remain general, the staff did not try to develop a detailed scenario, but simply assumed that the cladding of one fuel element certainly fails and that all of the fission products accumulated in the gap are released abruptly.

Several series of experiments at General Atomic have obtained data on the species and fractions of fission products released from uranium-zirconium-hydride under various conditions (14-9). The noble gases were the principal species found to be released, and when the fuel specimen was irradiated at temperatures below about 350°C, the fraction released could be summarized as a constant equal to 1.5×10^{-5}. The species released did not appear to depend on the temperature of irradiation, but the fraction released increased significantly at much higher temperatures.

General Atomic has proposed a theory describing the release mechanisms in the two temperature regimes which appears plausible, although the data do not agree in detail. It seems reasonable to accept the interpretation of the low temperature results, which implies that the fraction released for a typical TRIGA fuel element will be a constant, independent of operating history or details of operating temperatures, and will apply to fuel whose temperature is not raised above approximately 400°C for any appreciable time. This means that the 1.5×10^{-5} could be reasonably applied to TRIGA reactors operating up to at least 800 KW steady state.

The theory in the fuel temperature regime above approximately 400°C is not as well established. The proposed theory of release of the fission products incorporates a diffusion process which is a function of temperature and time. Therefore, in principle, details of the operating history and temperature distributions in fuel elements would be required to obtain actual values for release fractions at the higher temperatures.

Because the validity of the theory may not justify this detail, and because any prediction of future operating schedules of most research reactors is not justified, the staff selected a release fraction from the General Atomic results which corresponds approximately to AFRRI's authorized maximum measured operating fuel temperature, namely 600°C. Because the General Atomic measurements have been adjusted to infinite operating times at the various temperatures, it is likely that this approach will give a conservatively high value compared to the expected release at the AFRRI reactor. The release fraction the staff selected is 1×10^{-4} of the inventory of both the noble gases and the iodines (see Figure 6, ref. 14-9 (a)). The selection of this value for the fractional release does not represent a disagreement with the value assumed by the applicant in his accident analysis, because he chose an even more conservative value. Furthermore, the staff selection is not to be interpreted as being in disagreement with the use of 1.5×10^{-5} for lower authorized or actual operating temperatures. During steady-state operation at 1 MW, AFRRI's measured fuel temperature does not exceed 450°C, and because the thermocouples are near the axial center of the hottest fuel elements, they measure the region of maximum temperature, well above the core average.

Because the noble gases do not condense or combine chemically, it is correct to assume that any released from the cladding will diffuse in the air until their radioactive decay. On the other hand, the iodines are chemically active, and are not volatile below about 180°C. Therefore, some of the radioiodines will be trapped by materials with which they come in contact, such as water, and structures. In fact, evidence indicates that most of these iodines will either not become or not remain airborne under many accident scenarios applicable to nonpower reactors (14-12). However, to be certain that the fuel-clad-failure scenarios discussed below led to upper limit dose estimates for all events, the staff assumed that 100% of the iodines in the gap become airborne. This assumption will lead to computed doses which may be at least a factor of 100 too high in some cases. The staff has reviewed the various acceptable methods for computing the expected dose beyond the confines of the AFRRI reactor room in case of a fission product release. The methods outlined in various Regulatory Guides for power reactors such as 1.3, 1.145, 1.109, 3.34, and 3.35 give results which are very conservative for nonpower reactors.

In fact, for the quantity of radioactivity that could result from the failure of the cladding of one maximally irradiated AFRRI fuel element, these methods generally give results which are so conservative as to be misleading. The applicant has used a method (14-11) that the staff compared with a recent applicable publication (14-10) and found acceptable. Thus, the staff based its evaluation on the applicant's methods but with different assumptions in most cases. These are discussed below.

(1) A single fuel-element-clad failure in air immediately after an extended 1-MW run which was followed by a 5$ pulse. The staff assumes that the reactor room exhaust dampers close and that all of the noble gases and iodine radionuclides in the fuel-cladding gap are released from the cladding and form a uniform distribution in the reactor room air instantly. Therefore, all of the radioactivity is confined in the room. The initial whole-body (immersion) dose rate to a person in the middle of the reactor room would be approximately 40 mrems per hour. This initial dose rate is an upper limit, because of the conservative assumptions. Because there is no credible way in which this type of accident could occur without the person in the room being alerted immediately, orderly evacuation of the room within minutes would be accomplished. There would be no airborne radioactivity outside of the building in this scenario.

(2) Assume the same event occurred, but that all of the air in the room subsequently leaked out of the building at a uniform rate, with no decrease in source strength due to radioactive decay. (For example, the leakage might be out the building exhaust stack.) The whole-body immersion dose to a person just outside the building for the entire leakage time would be less than 2 mrems and the 50-year committed dose to his thyroid from breathing the iodines in the air would be less than 60 mrems. In this scenario these doses would be upper limits either because the exposed subject would be warned and evacuated or the leakage could be controlled, because it can be assumed that the operation personnel would be on hand and alerted.

50-year committed dose to his thyroid from breathing the iodines in the air would be less than 60 mrems. In this scenario these doses would be upper limits either because the exposed subject would be warned and evacuated or the leakage could be controlled, because it can be assumed that the operation personnel would be on hand and alerted.

(3) The third accident event analyzed is the same as accident (2) but considers the potential exposures to personnel beyond the control of the NNMC authorities.

The nearest residences of the public are about 1000 ft from the AFRRI building complex. However, to add to the conservatism, the staff computed the potential dose to a person at 700 ft, assuming that 100% of the iodines and noble gases released from the fuel-clad escape from the building and are carried by a 3-fps wind, with Pasquill type F atmospheric conditions. This wind speed and stability condition are not frequent at AFRRI, but these assumptions lead to a "worst case" analysis. The staff used the formulation of the applicant as expressed in equations 3 and 4 in Chapter 6 of the SAR. However, as discussed earlier in this section, the staff assumed that only 1/10 as much of the noble gas fission product inventory escapes as did the applicant. Thus, the staff computed whole-body doses that are 1/10 of the applicant's, as listed in Table 7, Appendix C, of the SAR. On the other hand, the applicant assumed an iodine release only 1/10 as great as that assumed by the staff. (The applicant assumed 1% of .1%; the staff assumed 100% of .01%). Thus, at 700 ft from the AFRRI reactor building, the staff computed a total thyroid dose of approximately 10 mrems, whereas the applicant computed less than 2 mrems. As noted above, both of these computations are based on conservative assumptions, so the results are higher than would realistically occur.

14.2.7.1 Conclusion

In accordance with the discussions and analyses above, the staff concludes that if one fuel element from the AFRRI reactor were to release all noble gaseous and iodine fission products accumulated in the fuel-cladding gap, radiation doses to both occupational personnel and to the public in unrestricted areas would be far below the limits stipulated in 10 CFR Part 20. This conclusion is valid even for the very unlikely accident scenario selected, namely that the clad failure occurs immediately after an extended steady-state operation followed by a maximum reactivity transient and that all of the gap radioactivity, including all iodines, escapes and is carried downwind. These assumptions correspond to a very conservative scenario.

Furthermore, from the results the staff obtained, even if one-half of the fuel elements failed simultaneously, the expected whole body doses in unrestricted areas beyond 700 ft would be less than 2 mrems and still fall within 10 CFR Part 20.

The staff assumed in scenarios (2) and (3) that the fail-safe engineered safety feature (the exhaust system dampers) did not function. This adds to the conservatism of the scenarios. Therefore, the staff concludes that even in the event of a multiple fuel clad failure at the AFRRI reactor, there would be no significant risk to the health and safety of the public.

15 TECHNICAL SPECIFICATIONS

The applicant's Technical Specifications evaluated in this licensing action define certain features, characteristics, and conditions governing the continued operation of this facility. These Technical Specifications are 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 the Draft Standard ANS 15.1 (September 1981) as a guide.

Based on its review, the staff concludes that normal plant operation within the limits of the Technical Specifications will not result in offsite radiation exposures in excess of 10 CFR Part 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 of offnormal or accident events.

16 FINANCIAL QUALIFICATIONS

The AFRRI reactor is operated by the Department of Defense in support of its assigned mission. 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 applicant'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 insignificant risk of radiation exposure to the public, and that only an offnormal or accident event could cause some significant exposure. Even a design-basis accident (defined as one worse than can be mechanistically justified) would not lead to a dose to the most exposed individual greater than applicable guidelines or regulations (10 CFR Part 20).

In this section, the staff reviews the impact of prior operation of the facility on the risk of radiation exposure to the public. The two parameters involved are the likelihood of an accident and the consequences if an accident occurred.

Because the staff has concluded that the reactor was initially designed and constructed to be inherently safe, with additional engineered safety features, the staff must also consider whether operation will cause significant degradation in these features. Furthermore, because loss of integrity of fuel cladding is the design-basis accident, the staff must consider mechanisms which could increase the likelihood of failure. Possible mechanisms are: (1) radiation degradation of cladding strength, (2) high internal pressure due to high temperature leading to exceeding the elastic limits of the cladding, (3) corrosion or erosion of the cladding leading to thinning or other weakening, (4) mechanical damage due to handling or experimental use, and (5) degradation of safety components or systems.

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

(1) The present fuel has been in use since 1965 and has been subjected to less than 1% burnup of U-235. Some TRIGA fuel at more extensively used reactors has been in use for at least 10 times as much burnup, with no observable degradation of cladding as a result of radiation. While increased operation is authorized under the present license and is physically possible, it is unlikely that AFRRI's program will change during the renewal period and alter this conclusion.

(2) The possibility of approaching such pressures would occur if the entire fuel element including the cladding were to be heated to more than 930°C (4-2). While it is likely that some points in the fuel would approach this temperature for a few seconds following a 2.8% $\Delta k/k$ (4.00$) pulse, only a simultaneous and instantaneous total loss of coolant could cause the cladding temperature to exceed a few hundred degrees. Because the staff has been unable to construct a credible scenario involving all of these assumptions, the staff concludes that there is no realistic event which would cause the elastic limit of the cladding to be exceeded.

(3) Water flow through the core is obtained by natural thermal convection, so the staff concludes that erosion effects due to high flow velocity will be negligible. High primary water purity is maintained by continuous passage through the filter and demineralizer system. With conductivity below about 5 μmho/cm, corrosion of the stainless-steel cladding is expected to be negligible, even over a total 40-year period.

(4) The fuel is handled as infrequently as possible, consistent with periodic surveillance. Any indications of possible damage or degradation are investigated immediately. The only experiments which are placed near the core are isolated from the fuel cladding by a water gap and at least one metal barrier, such as the pneumatic tubes or the core experiment tube. Therefore, the staff concludes that loss of integrity of cladding through damage does not constitute a significant risk to the public.

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

The second aspect of risk to the public involves the consequences of an accident. Because the AFRRI reactor has not and is not expected to operate on the maximum available schedule, the inventory of radioactive fission products will be far below that postulated in the evaluation of the design-basis accident both by the applicant and the NRC staff (see Chapter 14). Therefore, the staff concludes (1) that the risk of radiation exposure to the public has been acceptable and well within all applicable regulations and guidelines during the history of the reactor, and (2) that there is reasonable assurance that there will be no increase in that risk in any discernible way during this renewal period.

17.2 Multiple or Sequential Failures of Safety Components

Of the many accident scenarios hypothesized for the AFRRI-TRIGA Mark F reactor, none produce consequences more severe than the design-basis accidents reviewed and evaluated in Chapter 14. The only multiple-mode failure of more severe consequence would be failure of the cladding of more than one fuel element. No credible scenario constructed by the staff has included a mechanism by which the failure of integrity of one fuel element can cause or lead to the failure of additional elements. Therefore, if more than one clad should fail, the failures would either be random, or a result of the same primary event. Additionally, the reactor contains redundant safety-related measuring channels and control rods. Failure of all but one control rod and all but one safety channel would not prevent reactor shutdown to a safe condition. The staff review has revealed no mechanism by which failure or malfunction of one of these safety-related components could lead to a nonsafe failure of a second component.

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-84 for its research reactor filed by the Armed Forces Radiobiology Research Institute, dated October 3, 1980, 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 applicant 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; and

(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)

This Safety Evaluation Report for the application filed by the Armed Forces Radiobiology Research Institute (AFRRI), Defense Nuclear Agency for a renewal of license R-84 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 an agency of the U. S. Department of Defense, and is located on the grounds of the National Naval Medical Center, Bethesda, Montgomery County, Maryland. Subject to favorable resolution of one outstanding item discussed in this report, the staff concludes that the facility can continue to be operated by AFRRI without endangering the health and safety of the public.

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