NUREG-0947

Safety Evaluation Report
related to renewal of the operating license for the Texas A&M University Research Reactor

Docket No. 50-128
License R-83

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

March 1983

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ABSTRACT

This Safety Evaluation Report for the application filed by the Texas A&M University (Texas A&M) for a renewal of operating license number R-83 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 and operated by the Texas Engineering and Experiment Station of the Texas A&M University and is located on the campus in College Station, Brazos County, Texas. The staff concludes that the TRIGA reactor facility can continue to be operated by Texas A&M University without endangering the health and safety of the public.

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1 INTRODUCTION AND HISTORY

By letter dated July 2, 1979, accompanied with supporting documentation, the Texas A&M University Nuclear Science Center (Texas A&M/applicant) submitted a timely application to the U. S. Nuclear Regulatory Commission (NRC/staff) for renewal of the Class 104 Operating License (R-83) for its modified TRIGA (training reactor, isotope production, General Atomic) research reactor. The letter requested renewal of the Operating License for 20 years. On April 16, 1982, Texas A&M submitted a supplement to the Safety Analysis Report replacing the originally submitted Technical Specifications. Texas A&M 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 (10 CFR 2.109), until NRC action on the renewal request is completed.

The renewal application is supported by information provided in the Physical Security Plan, the Technical Specifications, the Environmental Impact Appraisal Data, the Safety Analysis Report (SAR), the Reactor Operator Requalification Program, and the 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 purpose of this Safety Evaluation Report (SER) is to summarize the results of the safety review of the Texas A&M modified TRIGA 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 Texas A&M facility at steady-state thermal power levels up to and including 1 MW, plus periodic pulsed operations to be governed by a maximum fuel temperature of 830° C. The facility was reviewed against the requirements of 10 CFR 20, 30, 50, 51, 55, 70, and 73; applicable Regulatory Guides (RGs) (Division 2, Research and Test Reactor); and appropriate accepted industry standards (American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 Series). Because there are no accident-related regulations for research reactors, the staff has compared calculated dose values with related standards in 10 CFR 20, the standards for protection against radiation both for employees and the public.

The staff technical safety review with respect to issuing a renewal operating license to Texas A&M has been based on the information contained in the renewal application and supporting supplements, generic studies performed by national laboratories, site visits, and responses to requests for additional information.

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

The application for a construction permit and operating license was submitted in March 1958. The construction permit (CP) RR-38 was issued in August 1959. This permit was converted to Operating License R-83, which authorized operation of a materials testing, swimming-pool-type reactor at 100 kW.

The reactor first went critical on December 18, 1961. The facility serves many campus departments, other universities and colleges, several city and state agencies, and other industrial and research organizations. By January 1965, the use of the facility had increased to the level where Texas A&M decided to operate on a two-shift basis for three days a week with a one-shift operation for the other two days. Since July 1966, the reactor has routinely operated two shifts for five days a week. In 1968 the reactor was converted to TRIGA fuel and the power level increased to 1,000 kW.

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 independent national laboratory studies and analyses of several severe hypothetical accidents postulated for the TRIGA-type reactor, as well as a detailed review of the damaged fuel elements (NUREG/CR-2387) (discussed in detail in Section 14).

The principal safety matters reviewed for the Texas A&M reactor and the conclusions reached follow.

(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 small fractions of 10 CFR 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 10 CFR 20 and are as low as reasonably achievable (ALARA).

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

(6) The staff's review of the applicant's report (GA-A16613) on the 1976 damaged fuel incident generally concurs with the applicant's analysis and evaluation concerning the reasons for and the remedies to prevent future fuel damage.

However, the staff recognizes that the January 1983 discovery of two additional damaged fuel elements indicates that some factors may have had a greater impact on the fuel damage mechanisms than those proposed in the report. The inspection requirements and the operating limitations included in the Technical Specifications are intended to both verify the conclusions reached in the damaged fuel incident report and preclude the events from reoccurring (see Section 17).

(7) From the financial data and information provided by the applicant, the staff has determined that the applicant has sufficient revenues to cover operating costs and to ensure protection of the public from radiation exposures when operations are terminated.

(8) 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 73.

(9) 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.

(10) The Texas A&M Emergency Plan, though submitted with the license renewal application, is incomplete at the time of publication of this safety evaluation report because of a requirement change by NRC. This item (discussed further in Section 13.3) was submitted separately by letter and report dated October 3, 1982 as part of the current NRC requirements, which were published in the Federal Register in May 1982. The evaluation of the Emergency Plan will be included in the license by amendment at a future date.

1.2 Reactor Description

The Texas A&M University Nuclear Science Center (NSC) reactor has a nominal power level of 1 MW and a maximum pulse level determined by maximum fuel temperature limited to 830° C. The reactor utilizes both standard and FLIP-type* fuel elements using partially enriched uranium fuel homogeneously mixed with a zirconium hydride (ZrH_x) moderator. The current core is entirely FLIP-type fuel elements.

Fuel elements and control rods are contained in bundle assemblies as shown in Figure 1.1. A reactor core loading can have several configurations in the 9-by-6 array of 54 holes. A typical 5-by-5 core loading containing 98 elements and graphite reflectors is shown in Figure 1.2.

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*FLIP (Fuel Life Improvement Program) is a type of long-lived fuel developed by General Atomics Company for TRIGA reactors.

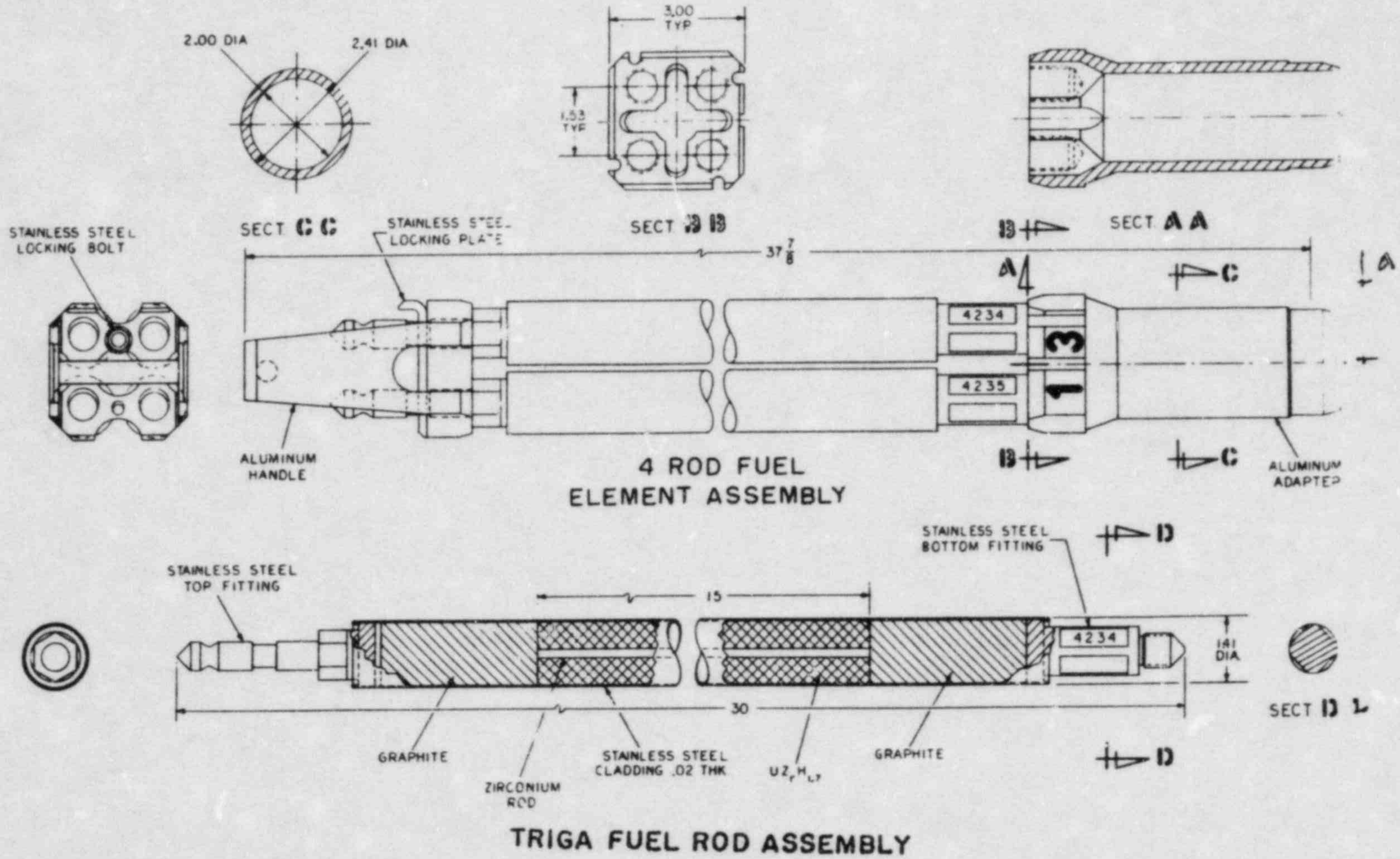
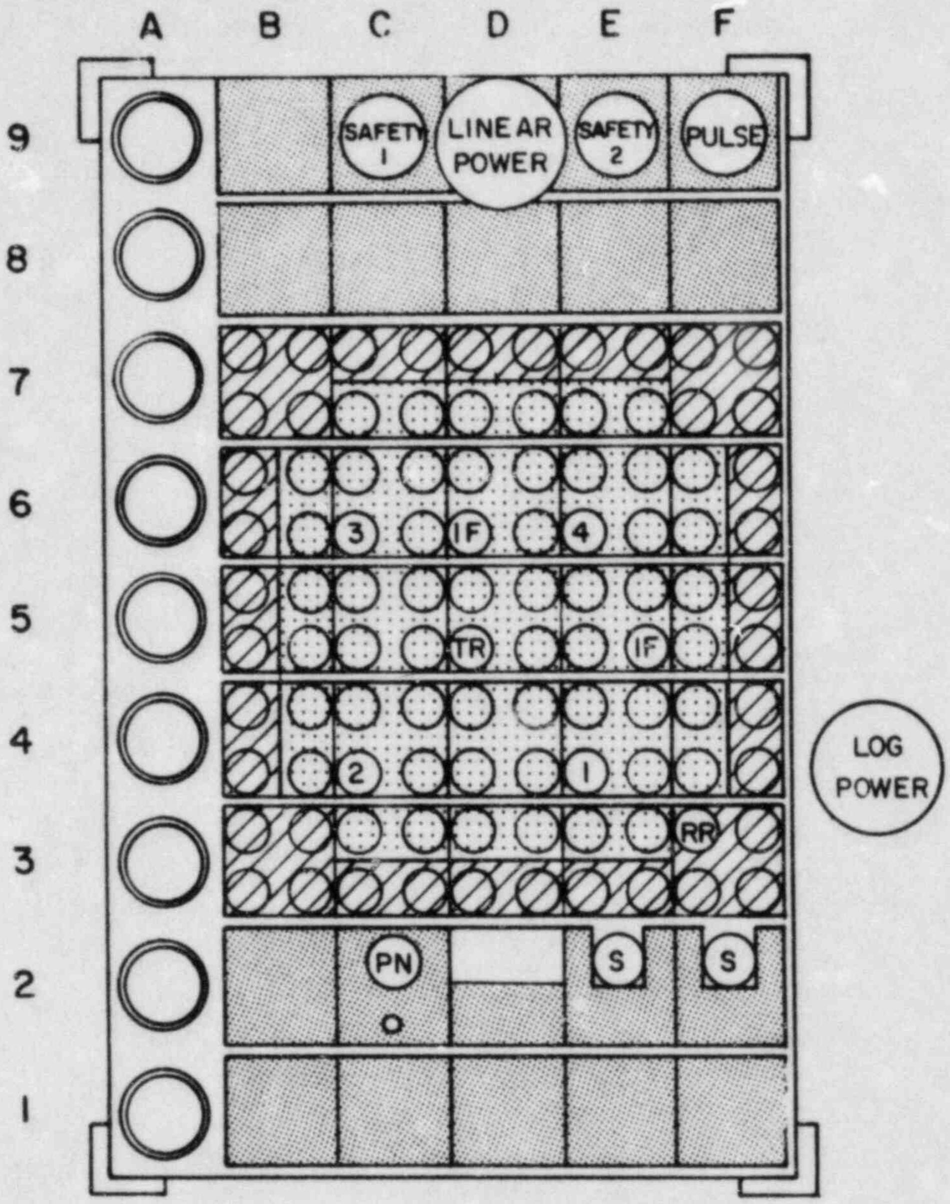


Figure 1.1 TRIGA fuel rod and fuel element assembly



TR Transient Rod With Air Follower

1 Shim Safety Rod With Fueled Follower

RR Regulating Rod With H_2O Follower

IF Instrumented Fuel

S Sb-Be Neutron Source

Standard Fuel

Flip Fuel

Graphite Reflector

PN Pneumatic Tube

Figure 1.2 Typical core configuration

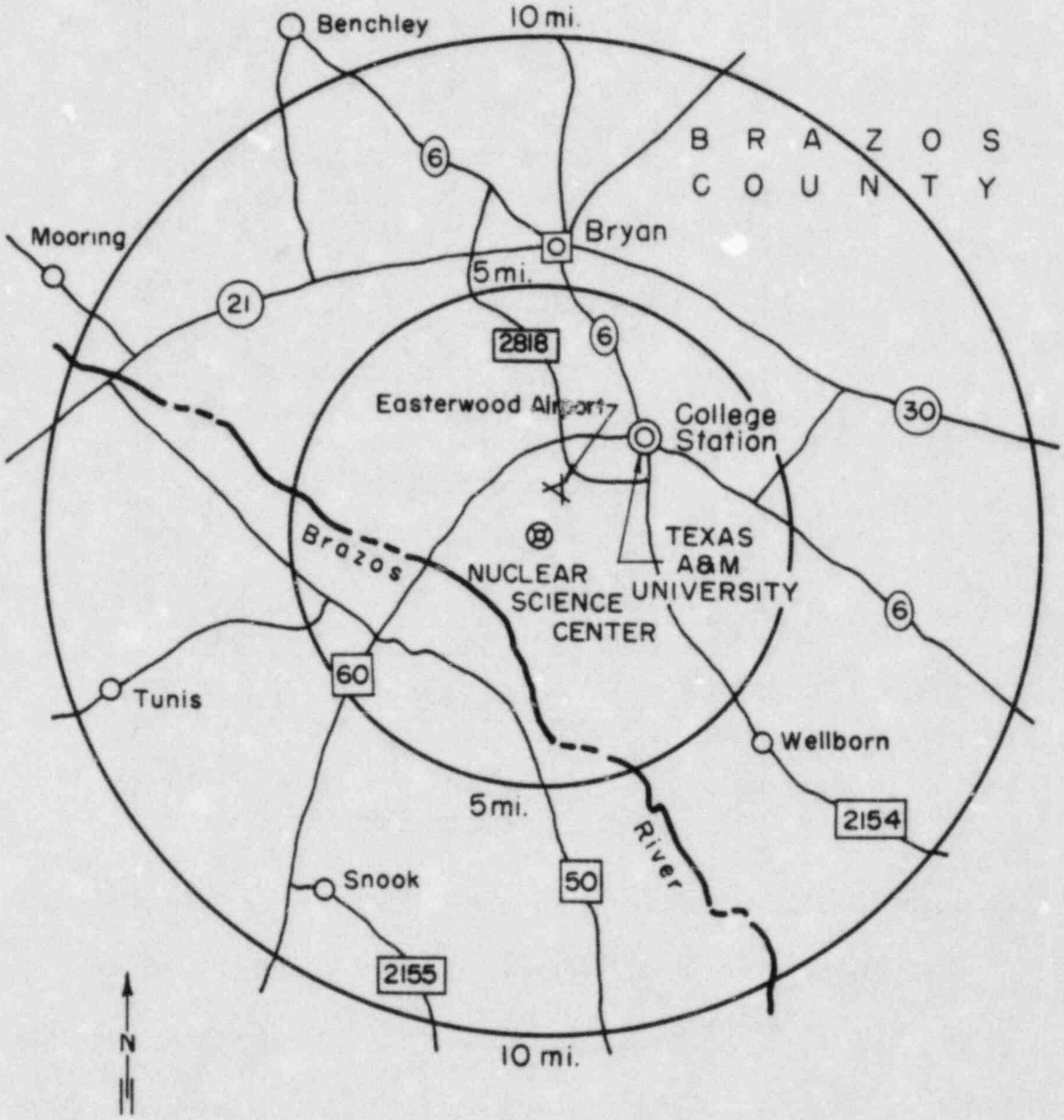


Figure 1.3 Nuclear Science Center regional map

1.3 Reactor Location

The Texas A&M University NSC is an isolated facility consisting of the reactor building and a supporting laboratory services building. The NSC is situated on a rectangular 6-acre site about 1,500 ft south of the north-south runway of Easterwood Airport, 2.5 mi west-southwest of the city of College Station, and 8 mi northwest of Wellborn in Brazos County, Texas. The reactor location is shown in Figure 1.3.

1.4 Shared Facilities and Equipment and Any Special Location Features

The reactor building, which is constructed of reinforced concrete and situated partially below grade, is attached to a laboratory complex dedicated primarily to nuclear science-related research and other reactor utilization. Utilities such as municipal water and sewage, natural gas, and electricity are provided to the complex by the local utilities (Safety Analysis Report, 1979).

The reactor building has its own ventilation control system, capable of isolation or dilution modes of operation to prevent or to exhaust air through an elevated stack located on the roof of the attached laboratory building. This stack also exhausts air from the reactor operating areas at a typical total flow of about 5,000 cfm.

1.5 Comparison with Similar Facilities

Though the original reactor core has been converted from a materials testing reactor (MTR) pool-type unit to a TRIGA type, the fuel is similar to all TRIGA reactors using standard or FLIP-type fuel. There are 58 TRIGA reactors operating throughout the world, 27 of which are in the United States (24 licensed by NRC, the other 3 by the Department of Energy). The instruments and controls are typical of NRC-licensed research reactors.

1.6 Facility Modifications

Major core and facility modifications occurred in 1968 when the reactor was converted from a pool reactor using MTR-type fuel to a TRIGA reactor. At that time the authorized power level was increased to 1,000 kW.

The modification and expansion of the NSC facility included four separate phases that were completed in 1969. The major modifications made during each of these phases is described below (Safety Analysis Report, 1979).

Phase I Pool Modification and Liner

The large reactor pool was modified by installing a multipurpose irradiation cell. This facility allows exposure of large animals or other objects to the radiation from the reactor core. A permanent stainless-steel liner was installed as part of the pool modification to eliminate pool leakage that previously had caused significant operational problems.

Phase II Cooling System

A cooling system was installed to allow steady-state operation at power levels up to 1 MW.

Phase III Conversion of the Reactor Core

The reactor core was converted to employ standard TRIGA fuel elements, and, on July 31, 1968, an amended facility license allowed the Texas A&M reactor to be operated at a maximum steady-state power level of 1,000 kW and a maximum pulse reactivity insertion of 3.00$.

Phase IV Laboratory Building

A laboratory building for the NSC was constructed to complement the university research and reactor operating programs.

1.7 Operational Modifications

Operating experience with the standard TRIGA fuel revealed a high-fuel burnup rate resulting in fuel additions to maintain sufficient reactivity. Core life was extended by modification of the reactor grid plate in late 1970 to provide for the installation of fuel follower control rods. This increased the core life by approximately 1½ years, but reduced the fluxes that were available for irradiation. To increase flux and fuel life, TRIGA FLIP fuel elements were placed in the Texas A&M NSC reactor core. Since June 1973, the Texas A&M reactor has been licensed to operate standard, mixed, or FLIP/TRIGA cores at a maximum steady-state power of 1,000 kW with an initial maximum pulse reactivity insertion of 2.00$. In July 1973, the reactor was first placed into service with a mixed TRIGA core containing 35 FLIP and 63 standard fuel elements. In July 1975, the maximum pulse reactivity insertion was increased to 2.70$. Present reactor operation uses a mixed core loading or a full-FLIP core.

On September 27, 1976, the reactor experienced damage to three fuel elements (General Atomics (GA)-A16613, 1981); however, no fuel element cladding failure was experienced. Texas A&M agreed not to pulse the reactor until a report concerning the analysis evaluation and recommendations was prepared to be submitted and accepted by the staff. This report (GA-A16613, 1981) was submitted to the NRC for review and evaluation. The staff concludes that the bases for the incident and the mitigation of any possible future incidents is acceptable. Consequently, the applicant can now resume pulsing of the reactor.

During the last 5 years, the average annual use of the reactor has been about 2,700 hours per year and 2,400 MW hours of consumption as shown in Table 1.1. In terms of radiation exposure of reactor components or production of radioactive material, this amount of operational use corresponds to about 200 working days per year at maximum authorized steady-state power (Annual Operating Report for 1981).

Table 1.1 Reactor utilization for calendar years 1976-1981

Calendar year	MW hours per day
1976	108.7
1977	104.3
1978	87.7
1979	85.7
1980	91.1
1981	105.4

2 SITE CHARACTERISTICS

2.1 Geography and Demography

The Bryan-College Station area is located approximately 100 mi inland from the Texas Gulf Coast. The reactor facility is located 3 mi from the university campus, 6 mi south of the city of Bryan (estimated population 46,600), 2.5 mi west-southwest of the city of College Station (estimated population 42,400), and 8 miles northwest of Wellborn (estimated population 1,200) in Brazos County, Texas. See Figure 1.3. The nearest nonuniversity occupied dwelling is about 1 mi from the Nuclear Science Center (NSC).

The population within about 1 mi of the site is extremely sparse and is estimated to be 500 persons. The population distribution was calculated for a typical day with the university students, faculty, and staff present on the campus.

2.2 Nearby Industrial, Transportation, and Military Facilities

The Texas A&M reactor facility is located 1,500 ft from the Easterwood Airport. Airline service to and from the Easterwood Airport consists of small commercial feeder lines. The airport is used also by the owners of small private planes. There are no heavy industries or large military installations nearby.

2.3 Meteorology

2.3.1 General

The local weather is determined to a great extent by the high pressure areas that are predominant over the Gulf of Mexico. As a result of this condition, warm southeasterly winds occur a large majority of the time on an annual basis. Average annual rainfall is 30-35 in. Snow occurs only rarely and subfreezing temperatures are encountered infrequently for brief periods during the winter. The average wind frequency distribution is shown in Figure 2.1.

The passage of frontal systems is normally accompanied by northwest winds as shown in the winter wind rose diagram in Figure 2.1. Calms occur an average of 10% of the time, and wind speeds above 21 knots are seldom encountered.

2.3.2 Severe Wind Characteristics

Data on tornado frequency between 1950 and 1976 indicated that 17 tornadoes were reported within a 25-nautical-mile radius of College Station. The mean path length is 2.24 mi, and the mean path area is .23 mi^2. The tornado season is usually from March through June; the months of April and May have the most occurrences of tornadoes over this period with the greatest probability of appearance being in the afternoon hours from about 2:00 to 7:00 p.m. A study by the National Severe Alarms Forecast Center in 1976 of the movement of

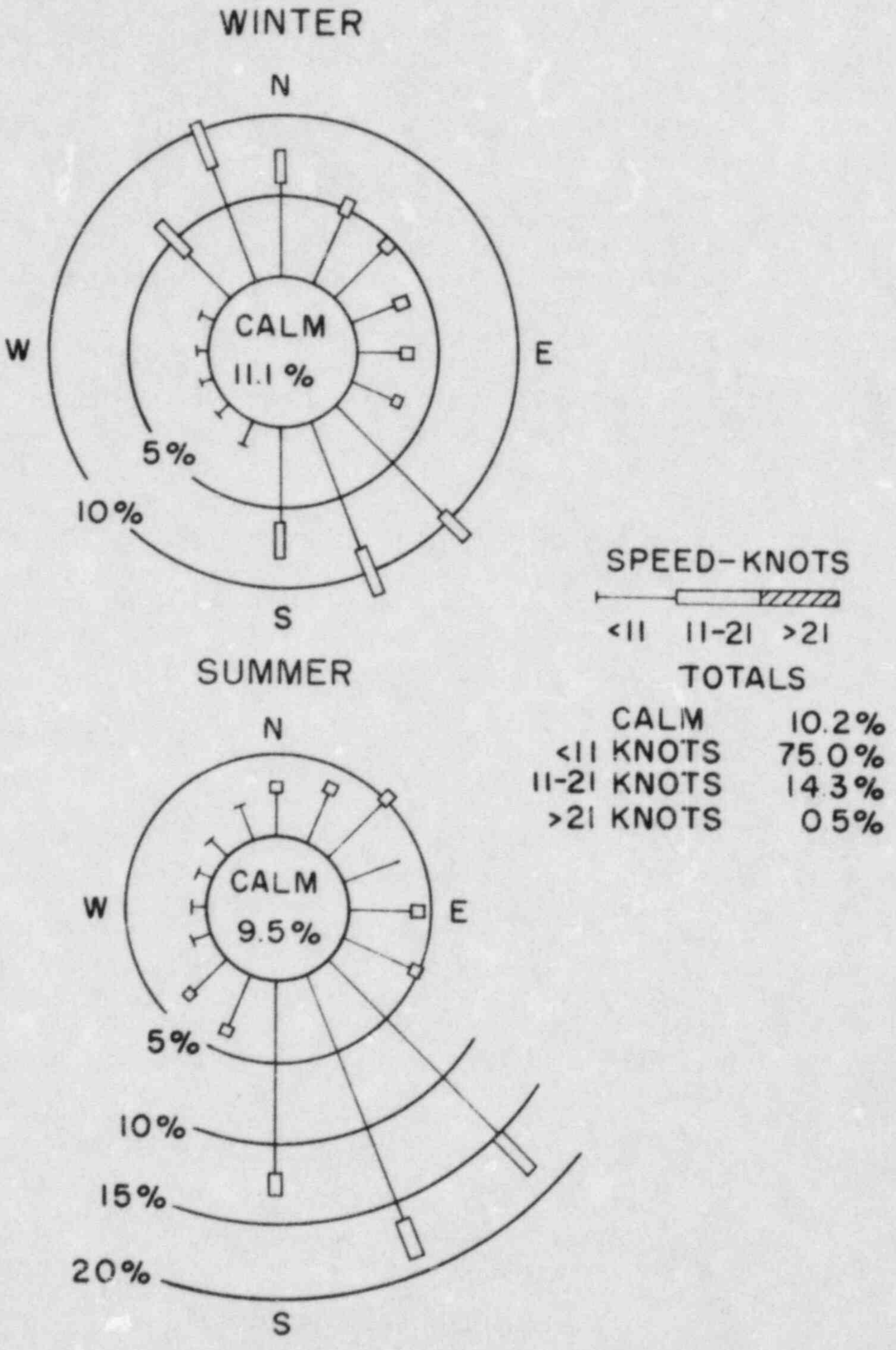


Figure 2.1 Average wind frequency distribution at the Nuclear Science Center

tornadoes indicated that a tornado will have 6% probability of having a westerly component in its direction of movement and a large percentage of these will move to the northwest.

The applicant indicates that the reactor building is designed to withstand 30 psi. In case a tornado passed nearby, the roof would possibly act as a pressure relief mechanism. The applicant indicates that the basic steel structure in the roof would probably remain intact unless direct contact was made on the building by the tornado. The building is designed to withstand a sustained wind velocity of 90 mph.

The radio room at Texas A&M University communications center receives notification of tornadoes sited or detected within a 5-mi radius by the Texas A&M weather radar and the Brazos County, Bryan College Station Disaster Emergency Planning Organization. The method of tornado detection is by Texas A&M radar, area spotters, and the National Weather Service.

In the event of a tornado warning, the reactor is shut down.

2.3.3 Conclusion

Though the remote possibility of tornado damage may exist, the staff concludes that the facility notification and reactor shutdown procedures together with the facility construction provide the necessary safeguards to ensure that the health and safety of the public will not be endangered.

2.4 Geology and Hydrology

The reactor site is located in a geographical region known as the Gulf Coastal Plain. The nearest fault zone (now dormant) lies 100 mi to the west, is known locally as the Balcones Escarpment, and is at the western boundary of the Gulf Coastal Plain.

Drainage of the site is by way of a tributary of White Creek to the Brazos River 3 mi to the southwest. The facility is situated on high ground, and the entire area is well drained by a number of tributaries of White Creek. Based on past history, the site, which is approximately 304 ft above mean sea level (MSL), is not in any flood area. The highest recorded crest on the Brazos River at Bryan (December 1913) was 54 ft above flood stage or 246 ft above MSL (Texas A&M, Department of Geology).

The probability of contaminating drinking water supplies is virtually eliminated since the Brazos River is not used as a source of drinking water and there are no open reservoirs in the surrounding area. The public water supply is pumped from deep wells several miles from the NSC.

Groundwater is not expected to present any problems. The NSC is constructed on a formation known as the Easterwood shale, which is 10 to 300 ft thick. The shallowest acquifer is the Bryan sandstone that underlies the Easterwood shale. It is well below the depths required for the NSC building excavation.

2.5 Seismology

Texas lies in a region of minor seismic activity. The extreme western portion of Texas, over 600 mi west of College Station, is nearest the active belt that lies along the west coast of Mexico and the United States. There are occasional minor shocks of very small magnitude in the state. The only earthquake of any significance occurred on August 16, 1931 near El Paso and was a Class C (6.4) shock (Texas A&M, Department of Geology).

The Texas A&M NSC wall structure and reactor pool walls are heavily reinforced, so it is anticipated that the building would withstand any minor shock that might occur.

2.6 Conclusions

The staff has reviewed and evaluated the Texas A&M reactor site for both natural and manmade hazards; it concludes that there are no significant risks associated with the site that make it unacceptable for the continued operation of the reactor.

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Wind Damage

As indicated in Section 2.3.2, severe wind characteristics associated with tornadoes are prevalent in the region from March through June. However, the reinforced concrete construction of the reactor building, designed for 90 mph winds, and the early warning system for approaching tornadoes that allows the reactor to be shut down in a timely manner, precludes any jeopardy to the reactor personnel or the contiguous populace and environment.

3.2 Water Damage

The reactor site is at el 304 ft MSL. The highest recorded crest of the Brazos River at Bryan was at el 246 ft MSL. Even though the reactor internals are below grade, flood water levels could not enter the site to cause any reactor facility damage.

The shallowest saturated zone lies 200 ft below the reactor and is covered by a thick formation of shale.

There is reasonable assurance, therefore, that there will be no damage caused from any extreme meteorological or hydrological conditions.

3.3 Seismic-Induced Reactor Damage

As stated in Section 2.5, the nearest seismic fault is some 600 mi away, and the incidence of seismic activity has been infrequent. Further, Texas A&M is situated in an area of low probability of seismic activity.

The regional, siting, and design considerations of the reactor building lead the staff to conclude that the risk of wind-, water-, or seismic-induced 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 that also supports the reactor core. The motors, gear boxes, electromagnets, switches, and wiring are above the reactor water level and readily accessible for testing and maintenance. An extensive preventive maintenance program has been in operation for many years at Texas A&M 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 (IE

Inspection Reports and reports of Reportable Occurrences from the licensee, Docket No. 50-128).

3.5 Conclusions

The staff concludes that there is reasonable assurance that there will be no significant deterioration of equipment with time or with operation and that continued operation for the requested period of renewal is in conformance with the performance requirements of the Technical Specifications and will not increase the risk to the public.

4 REACTOR

The Texas A&M TRIGA reactor is a 1-MW pool-type research reactor using light water as the moderator, coolant, and shield and TRIGA-type solid fuel rods.

It currently is authorized to operate either in the steady-state mode up to 1 MWt or in the pulse mode with a step reactivity insertion of up to 2.70$ (1.89% _k/k) or a limiting maximum fuel temperature of 830° C. Figures 4.1 through 4.4 illustrate the reactor building and reactor core/pool, irradiation, experimental and support facilities.

The reactor core is immersed in a large, concrete, stainless-steel-lined, water-filled, open-topped pool. The pool is spanned by a manually operated bridge structure from which the core support structure is suspended. The core is supported by a grid plate into which three- or four-rod clusters of TRIGA fuel rods are positioned as shown in Figures 1.1 and 1.2. Six motor-driven controls rods control the reactor during steady-state operation. One of these rods also is used as the transient rod for pulsing operations.

The reactor is controlled by inserting or withdrawing neutron-absorbing control elements suspended from control drives mounted on the reactor frame structure. Heat generated in the pool during and following operation is dissipated to the atmosphere by means of a cooling-tower heat-exchanger arrangement. A mixed-bed demineralizer system maintains the purity of the pool water.

4.1 Reactor Core

The reactor core includes a rectangular array of approximately 95 fuel rods of which all are currently FLIP rods. Also included are four shim-safety control rods, a transient control rod and guide tube, a regulating rod, and a startup neutron source. The fuel rods are contained in three- or four-rod elements that are supported by a grid plate. The grid plate provides a 9-by-6 array of square holes for fuel elements and for graphite reflector elements in the outer rows on the two ends. The current core contains 23 fuel bundles, each of which is approximately 3 in. square and 38 in. long, as shown in Figure 1.1.

The fueled region of the core is an approximate rectangular parallelepiped that is 15 in. high, 15 in. long, and 15 in. wide. In some core configurations, one or more fuel elements are removed or moved to other locations to allow space for notches for experiments. A fully loaded operational core contains about 11.7 kg of ^{235}U.

4.1.1 Fuel Elements

The Texas A&M reactor uses standard and/or FLIP TRIGA stainless-steel-clad cylindrical fuel rods in which enriched uranium is homogeneously mixed with a ZrH_x moderator. FLIP fuel also contains 1.5 weight-percent erbium as a burnable poison. The fuel part of each rod consists of a cylindrical rod of U-ZrH_x containing 8.5 weight-percent uranium with ^{235}U enriched to less than 20% in standard fuel and to 70% in FLIP fuel. The hydrogen-to-zirconium atom ratio of the

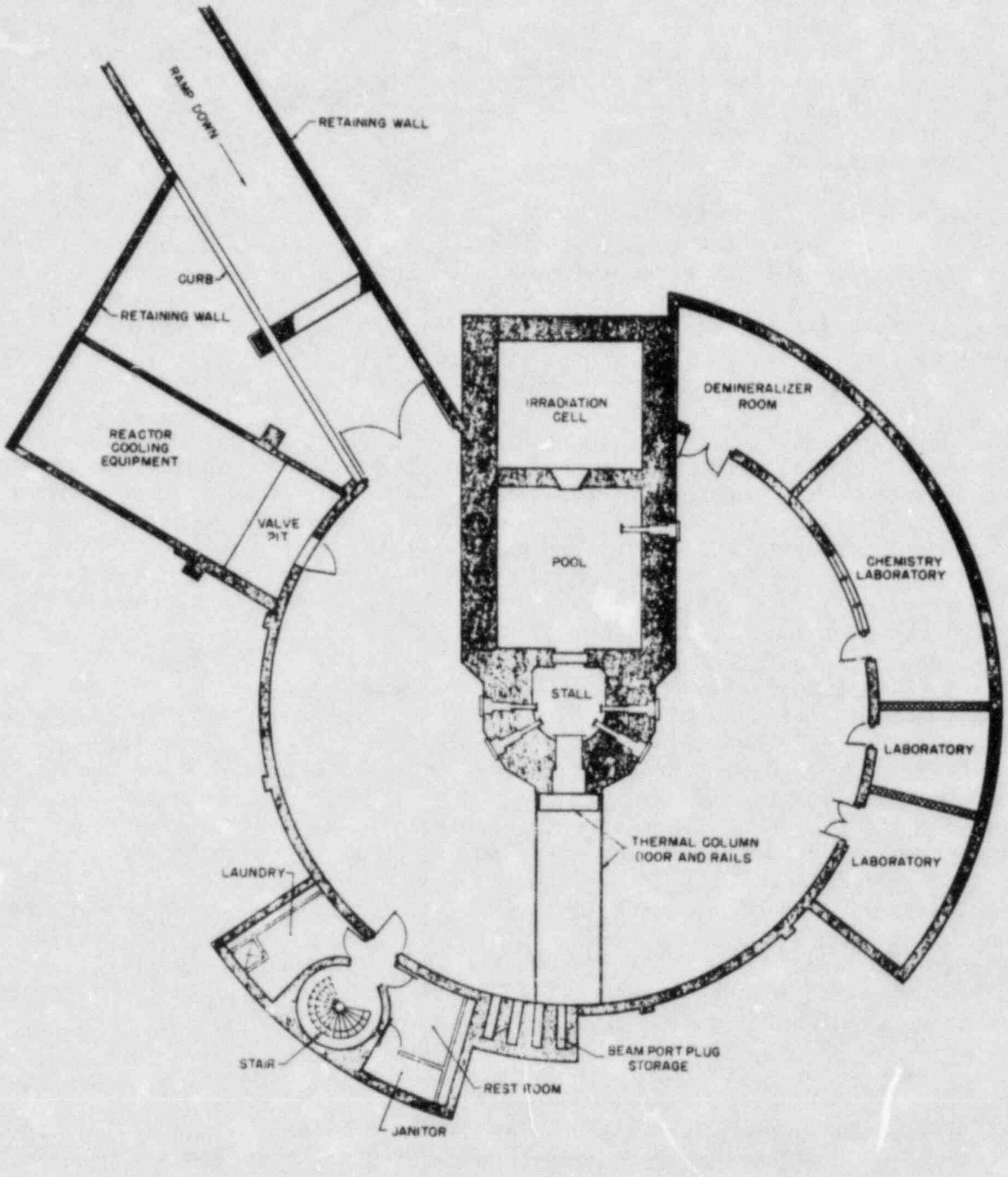


Figure 4.1 Lower research level

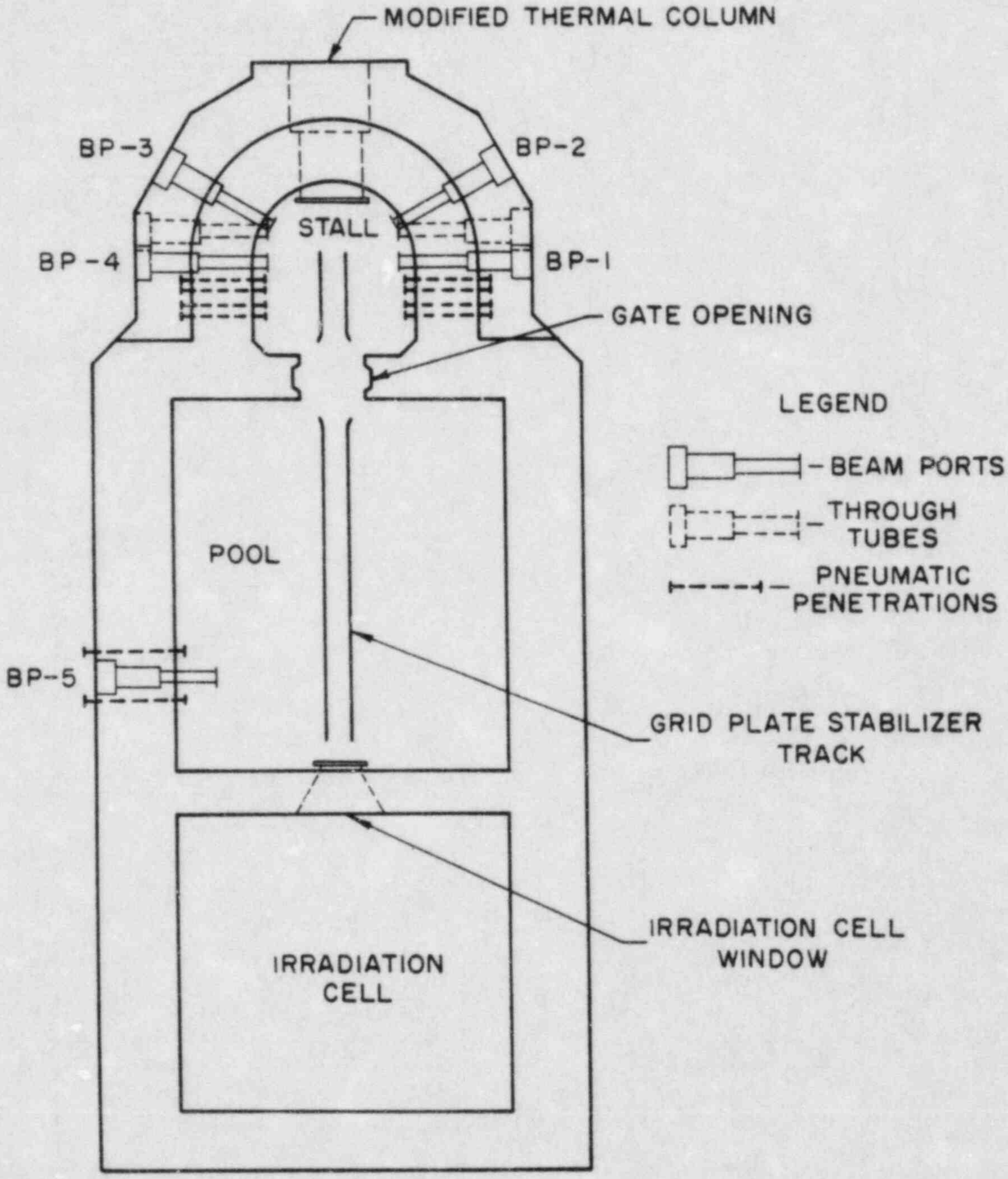


Figure 4.2 Texas A&M NSC reactor experimental facility

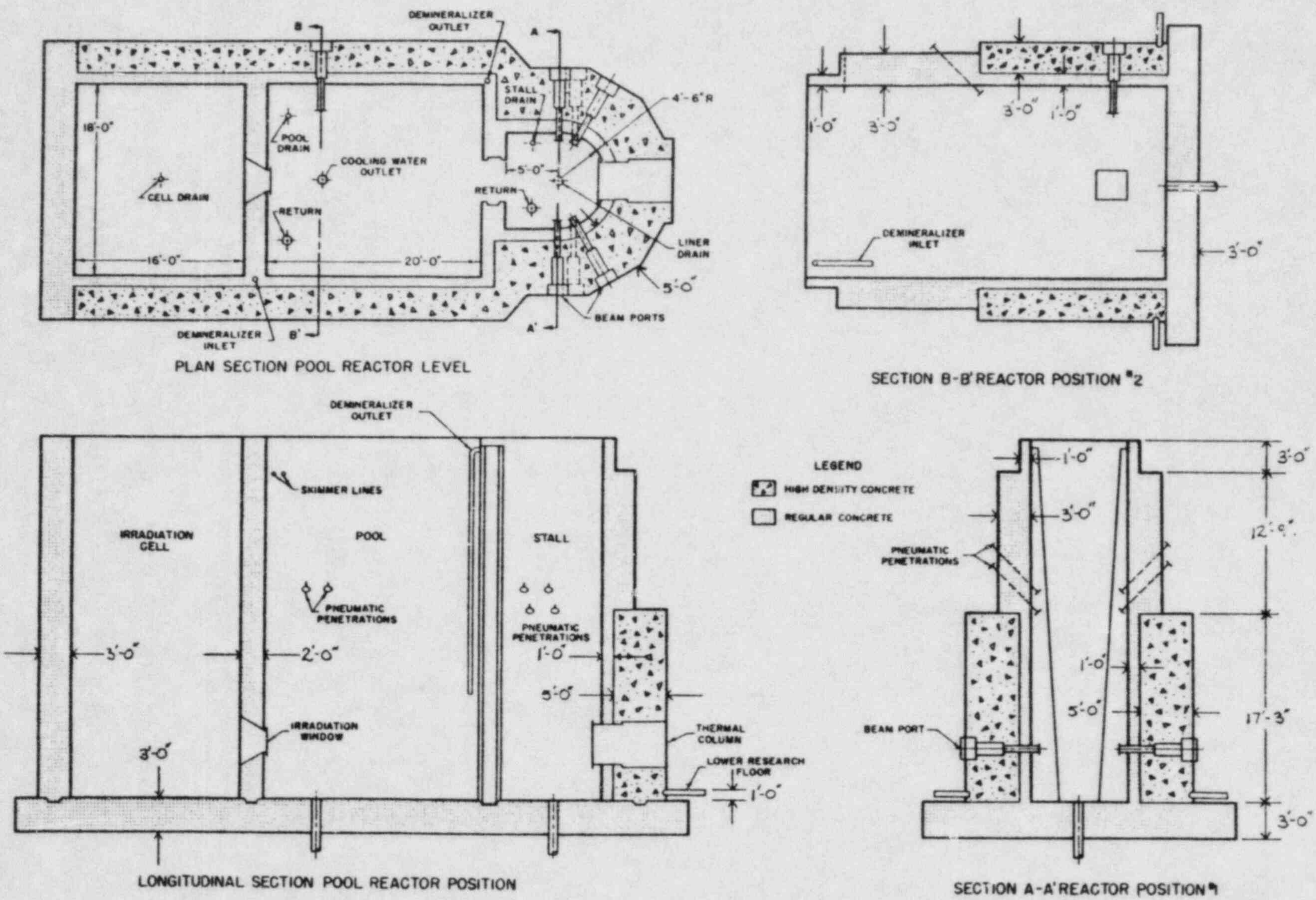


Figure 4.3 Texas A&M NSC reactor pool sections and penetrations

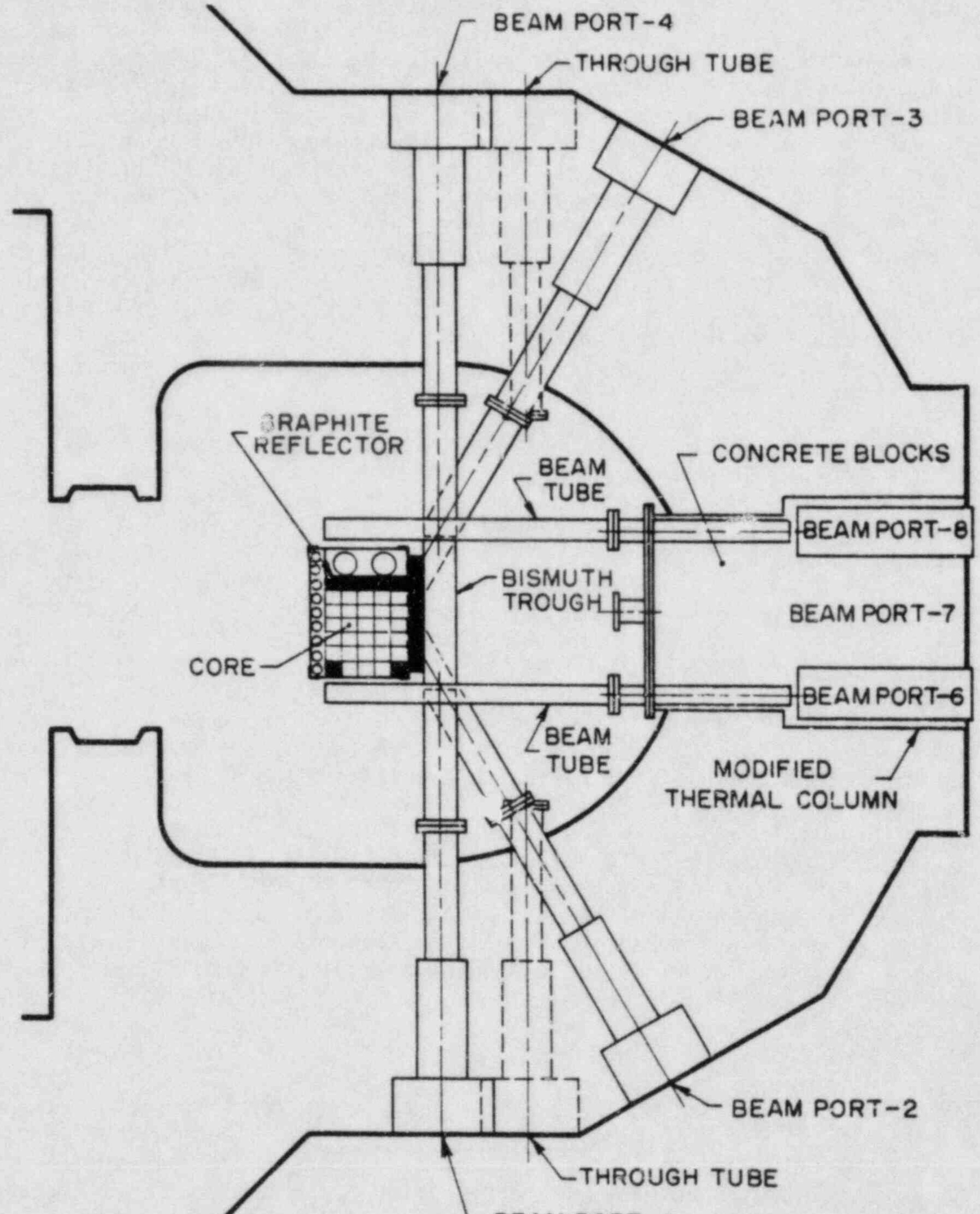


Figure 4.4 Reactor stall and beam port installations

fuel moderator material is approximately 1.7 to 1 in standard and 1.6 to 1 in FLIP fuel. The nominal weight of ^{235}U is 35 g in each standard fuel rod and 123 g in each FLIP rod. The fuel section of each rod is approximately 15 in. long and 1.41 in. in diameter as shown in Figure 1.1. Graphite end plugs (3.45 in. long) are located above and below the fuel section and function as neutron reflectors. The burnable poison (erbium) in each FLIP fuel rod compensates partially for reactivity changes caused by fission-product buildup and uranium burnup. At least one fuel rod position contains a special instrumented element into which thermocouples were fitted during fabrication. In all other respects, this rod is identical to a standard or FLIP fuel rod. The thermocouples monitor the axial temperatures in the instrumented element. The fueled section and the graphite reflectors are contained in a 0.020-in.-thick Type 304 welded stainless-steel-walled can. Each rod, about 30 in. long, weighs about 3.4 kg. As described above, the fuel rods are assembled in three or four fuel element bundles with fittings at the bottom to permit location in the grid plate and at the top to attach lifting handles.

4.1.2 Control Elements

Power levels in the Texas A&M reactor are regulated by four shim-safety control rods, a regulating rod, and a transient control rod. The poison section of each shim safety rod contains borated graphite, B_4C powder, or boron and its compounds in solid form in aluminum or stainless-steel cladding. The transient control rod contains either compacted borated graphite or solid boron compounds also in aluminum or stainless-steel cladding. The regulating rod is either a stainless-steel rod or contains the materials specified for the shim-safety rods. The shim-safety rods and the transient rod have scram capability, but the regulating rod does not. The shim-safety rods have fueled followers, the transient rod has an air follower, and the regulating rod has a water follower. All the rods are cylindrical. The shim-safety rods are 1.36 in. in diameter and 45.8 in. long, the transient rod is 1.25 in. in diameter and 37.6 in. long, and the regulating rod is 1.25 in. in diameter and 20.0 in. long.

Whereas the shim and regulating rods can be located in several optional core positions, the transient or pulse rod can be installed in only two core positions. High pressure air is used to keep this rod in position and also for pulsing. The reactor pool structure consists of a main pool between a stall section and a shield irradiation cell, as shown in Figure 4.2. The main cell dimensions are 18 ft wide, 20 ft long, and 33 ft deep; the total capacity of the cell is 143,000 gal.

4.1.3 Conclusion

The staff has reviewed the information regarding the reactor fuel core arrangement and reactivity control systems and found that the design and performance capability of the components are adequate to ensure the safe operation of the reactor during the proposed licensing period.

4.2 Reactor Pool

The reactor core is located in the pool under approximately 26.5 ft of light, demineralized water. This water serves as radiation shielding, a neutron moderator and reflector, and reactor coolant. The upper 16 ft of the reactor pool wall is constructed of ordinary concrete and the lower part of the wall is constructed of barytes concrete and ordinary concrete. The movable reactor bridge permits operation of the reactor at any position on the pool centerline. Tracks for positioning the reactor on the pool centerline are located on the liner floor. The stall section is 9 ft wide and has a 180° curved surface with a 4.5-ft radius at one end. The main pool is approximately 33 ft deep, 18 ft wide, and 20 ft long. The stall is penetrated by a thermal column, a through tube, and a number of beam ports; the main pool is penetrated by one beam port. For drainage or maintenance, the main pool and stall can be isolated from each other by use of an aluminum gate. Ten-gauge stainless-steel plates line the floors and walls of both to eliminate leakage. A drainage system is provided beneath the liner to collect any leakage that may occur. However, since the liners have been installed, leakage has not occurred.

Natural thermal convection of the pool water disperses the heat generated from reactor operations at all power levels. At normal operating power levels, the bulk pool water is pumped through a shell and tube heat exchanger system and through two 2-MW capacity cooling towers.

4.3 Support Structure

The entire reactor core support structure is suspended from a manually operated movable bridge structure that spans the pool. The movable bridge is mounted on rails to permit the reactor structure to be moved laterally within the pool. The control rod drives also are supported by the bridge structure.

4.4 Reactor Instrumentation

The Texas A&M reactor instrumentation is similar to that found in other research reactor installations at other institutions. Temperature measurements are made in the core by thermocouples fitted into a fuel rod. These measurements are indicated and recorded on the control console. Neutron flux measurements are made with a boron-lined compensated ion chamber and a fission ionization chamber with indicators located on the control console. The reactor power level is determined from the compensated ion chamber using a picoammeter and from the fission chamber using a log power channel. A neutron or gamma detector is used to determine the reactor power in the pulse mode of operation.

The reactor instrumentation is integrated into the overall control and instrumentation system, which is discussed in detail in Section 7.

4.5 Biological Shield

The reactor core is shielded in the lateral direction by the reactor pool water and the concrete walls of the pool. Vertical-direction shielding consists of approximately 26.5 ft of pool water above the core and about 3 ft of pool water and the concrete bottom of the pool below the core.

The upper halves of the walls of the stall are made of 3-ft-thick ordinary concrete and the lower halves are made of 5-ft-thick heavy concrete plus 1-ft-thick ordinary concrete. The bottom of the entire pool is made of 3-ft-thick ordinary concrete. The side walls of the main pool are of 3-ft-thick ordinary concrete for approximately the upper half and 3-ft-thick heavy plus 1-ft-thick ordinary concrete for the lower half. A 2-ft-thick ordinary concrete wall separates the main pool and the irradiation cell.

4.6 Dynamic Design Evaluation

Safe operation of a TRIGA reactor during normal operation is accomplished by the control rods and is monitored by the core power-level detectors. A backup safety feature is the reactor core's inherent large negative temperature coefficient of reactivity (for standard TRIGA fuel) resulting from an intrinsic molecular characteristic of the ZrH_x alloy at elevated temperatures. However, for FLIP fuel (70% enriched), the ^{235}U loading is about 3.5 times that of standard fuel (20% enriched) (Foushee, Shoptaugh, et al., 1971). This causes the neutron mean free path in FLIP fuel to be much shorter resulting in a smaller negative temperature coefficient. Therefore, erbium is used both as a burnable poison and as a material to enhance the prompt negative temperature coefficient of reactivity in FLIP fuel. General Atomic (GA) burnup calculations indicate that after 3,000 MW days of operation, the ^{235}U concentration averaged over the core is $\sim$ 67% and the ^{167}Er concentration is $\sim$ 33% of the core beginning-of-life values (GA-9350, 1969). These calculations show that for a FLIP fuel temperature of 700° C, the magnitude of the prompt negative temperature coefficient decreases from $\sim 1.75 \times 10^{-4}$ to $\sim 1.15 \times 10^{-4}$ $\Delta k/(k\cdot/^{\circ}C)$, or $\sim$ 34%. The decrease is smaller for lower fuel temperatures.

Because of the large prompt negative temperature coefficient, step insertions of excess reactivity resulting in an increasing fuel temperature will be compensated for rapidly and automatically by the fuel matrix. This will terminate the resulting excursion without any dependence on the electronic or mechanical reactor safety systems or actions of the reactor operator. This inherent characteristic of both the standard and the erbium-loaded FLIP 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 (GA-4314, 1980 and Simnad, et al., 1976). Abnormal operations (accidents) are discussed in Sections 14.

It is necessary, therefore, that the decrease in the magnitude of the prompt negative temperature coefficient of reactivity caused by erbium burnup be taken into account in both steady-state and pulsing operations for FLIP fuel that has reached significant burnup. The prompt negative temperature coefficient for FLIP fuel is discussed further in Section 14. This change in the temperature coefficient in FLIP fuel resulting from the erbium burnup is very gradual. Texas A&M personnel have made the necessary judgments for erbium burnup as the core ages.

4.6.1 Excess Reactivity

Pulse Operation

Excess reactivity in the Texas A&M reactor core is limited by the existing Technical Specifications, which stipulate the maximum temperature of the hottest fuel elements:

- Both the shutdown margin and the experiment reactivity are tightly controlled by the Technical Specifications.

- Previous experiments conducted by General Atomic have given no evidence of fuel damage resulting from rapid reactivity insertions up to 3.5% $\Delta k/k$ for full standard or full FLIP cores (GA-4314, 1980; Simnad et al., 1976; GA-6874, 1967).

- Evidence is available that indicates there will be no fuel damage if the maximum fuel temperature does not exceed 830° C (GA-A16613, 1981).

4.6.2 Shutdown Margin

The resubmitted Technical Specifications (Section 3.1.3) state:

> The reactor shall not be operated unless the shutdown margin provided by control rods is greater than 0.25$ with:
>
> (a) the highest worth nonsecured experiment in its most reactive state,
>
> (b) the highest worth control rod and the regulating rod (if not scramable) fully withdrawn, and
>
> (c) the reactor in the cold critical condition without xenon.

Because the regulating rod currently cannot be scrammed, this specification must be met with that rod fully withdrawn. The Technical Specifications limit nonsecured experiments to reactivity worths less than 1.0$ (0.70% $\Delta k/k$) and any single experiment to a reactivity worth less than 2.00$ (1.4% $\Delta k/k$). 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 shim-safety control rod ranges from 1.82$ (1.3% $\Delta k/k$) to 4.45$ (3.1% $\Delta k/k$) for the current 91-rod all FLIP core. The change in reactivity caused by complete operational withdrawal of the transient control rod is approximately 3.25$ (2.3% $\Delta k/k$). Full operational withdrawal of the regulating rod causes a reactivity change of about 0.80$ (0.56% $\Delta k/k$). And full operational withdrawal of the highest-worth control element causes a reactivity change of approximately 4.40$ (3.1% $\Delta k/k$).

The excess reactivity of the current reactor core is approximately 6.00$. The shutdown margin of this core with the highest-worth control rod and regulating rod fully withdrawn is: (3.25 + 2.71 + 1.82 + 2.48) - 6.00 = 4.26$. With all control rods inserted, this core is subcritical by 9.50$. Therefore, the current loading complies with the minimum shutdown margin limit in the Technical Specifications and permits experiments of total positive reactivity worth up to about 4.00$ (2.8% $\Delta k/k$) to be performed.

However, as the reactor has not been pulsed since the fuel damage incidents (see Section 17) and the current full FLIP fuel reactor core has not been pulsed, the applicant has proposed a cautious calibration program for the new core and the Technical Specifications have been modified to increase the frequency of visual inspections of the core fuel elements.

4.6.3 Conclusions

The staff concludes that the inherent large, prompt, negative temperature coefficient of reactivity of both the standard and erbium-loaded FLIP fuel provides a basis for safe operation of the Texas A&M 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 no more than 6.00$ (4.2% $\Delta k/k$) and experiments with positive reactivity worth less than 4.00$ (2.8% $\Delta k/k$) or other reactivity combinations totaling no more than 10.00$ (7.0% $\Delta k/k$) positive reactivity, the worth of the control rods will ensure a shutdown margin within Technical Specifications even if the most reactive control rod were operationally removed from the core.

4.7 Functional Design of Reactivity Control System

4.7.1 Standard Control Element Drives

The shim-safety and regulating rods are attached to their drive mechanisms by use of electromagnets. The drive unit for each shim-safety rod and for the regulating rod is an electrically driven screw through a gear reducer for movement of the control rod. If the electromagnets are deenergized for any reason, the shim-safety rods are released and fall by action of gravity into the reactor core within approximately 0.7 sec, resulting in a scram.

4.7.2 Transient Control Rod Drive

The drive unit for the transient or pulse rod is a combination pneumatic-electromechanical system that allows the reactor operator to use the transient rod as a control rod. The pneumatic portion of the drive system consists of an accumulator tank, air compressor, solenoid valve, pneumatic lines, cylinder, and actuating piston. Compressed air is used to drive the transient rod out of the core rapidly. If the air supply is interrupted, the rod falls by gravity into the core. The electromechanical portion of the drive consists of an electric motor, a ball-nut assembly, a threaded air-cylinder, and a worm-gear drive assembly. The threaded air-cylinder can be raised or lowered independently of the piston and control rod by means of the electric drive. This controls the upper limit of the transient rod travel and, hence, the amount of reactivity inserted for a pulse. This system is discussed further in Section 7.

4.7.3 Scram-Logic Circuitry

The reactor is equipped with a scram-logic system that receives signals from core instrumentation (neutron flux detectors and thermocouples). A wide range of scram modes is built into the overall logic circuitry. These scram modes are listed in Table 4.1, and additional details of the scram-logic system are provided in Section 7.

4.7.4 Conclusions

The Texas A&M reactor is equipped with safety and control systems typical of most nonpower reactors. 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.

Table 4.1 Scram modes

Scram	Mode
Console scram button	Initiated at the discretion of the reactor operator.
High power level detector power supply	Scrams on loss of supply voltage.
Preset timer	Transient rod scram 15 s or less after a pulse.
Fuel rod temperature	Scrams if fuel rod temperature exceeds 525° C. This circuit also scrams the reactor if a thermocouple fails. An open thermocouple produces a high-temperature scram. A shorted thermocouple produces a low-temperature scram.
High power level	Scrams if the reactor power level exceeds 125% of full-licensed power.
Experiment	Scrams the reactor if specified safety levels or events are observed during the performance of an experiment.

In addition to the active electromechanical safety controls for normal and abnormal operation, the large, prompt, negative temperature coefficient of reactivity inherent in the U-ZrH_x fuel moderator, discussed in Section 4.6, terminates reactor transients that produce large increases in temperature. 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 fail-safe engineered safety feature.

Therefore, the staff concludes that the reactivity control systems are of the design and function adequate to ensure safe operation and safe shutdown of the reactor under all normal operating conditions.

4.8 Operational Procedures

Texas A&M has implemented a 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 new experiments involving the use of this reactor are reviewed by the Texas A&M Reactor Safety Board 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.

4.9 Conclusions

The staff concludes that the Texas A&M reactor is designed and built according to good industrial practices. It consists of standardized components representing hundreds of reactor-years of operation and includes redundant safety-related systems.

The staff review of the Texas A&M reactor facility has included studying its specific design and installation, its control and safety instrumentation, and its specific preoperational and operating procedures. These features are similar to those typical of other research reactors operating in many countries of the world, more than 60 of which are licensed in the U. S. by the NRC. Based on the review of the Texas A&M reactor the performance requirement of the Technical Specifications and experience with these other research reactor facilities, especially TRIGA reactors, the staff concludes that there is reasonable assurance that the Texas A&M reactor is capable of safe operation for the period of the license renewal, as limited by its Technical Specifications.

5 REACTOR COOLANT AND ASSOCIATED SYSTEMS

The Texas A&M reactor coolant system consists of the primary cooling system, the secondary cooling system, the primary coolant purification system, and the reactor pool ^{16}N diffuser system.

5.1 Primary Cooling System

The reactor is located in a stainless-steel-lined concrete pool containing 143,000 gal of demineralized water for heat removal (see Figure 4.3). As shown in Figure 5 1, the primary coolant pump takes suction from the reactor pool at a flow rate of 1,000 gpm, pumps it through the tube side of a heat exchanger at a pressure of 17.5 psi, and pumps it back to the pool. In the event of a sudden failure of a primary coolant line or component, there are two raw-water supply lines adjacent to the pool that can supply about 400 gpm to the pool. Alarms located in the control room indicate primary or secondary pump failures. Because of the large pool and low rate of heat rise, there is no need to automatically scram the reactor should these pumps fail.

5.2 Secondary Cooling System

The secondary coolant pump circulates water at 1,575 gpm from a 2-MW cooling tower sump through the shell side of the system heat exchanger at a pressure of 22 psi and back to the cooling tower. The secondary coolant is automatically chemically treated at the cooling tower to prevent system corrosion or scaling.

5.3 Primary Water Purification and Makeup System

The Technical Specifications require a reactor primary coolant purity of not less than 5×10^{-6} mhos/cm^3. Conductivity is checked periodically during reactor startup procedures.

Reactor coolant purity is maintained by pumping 75 gpm of pool water through a gravel and charcoal filter, a regenerative mixed-bed demineralizer, and a cartridge-type micron filter bank before it is returned to the pool. Pool makeup water enters the system upstream of the gravel and charcoal filter and is introduced into the pool after filtration and demineralization. Another part of the primary coolant purification system pulls water from the pool surface through two floating skimmers, pumps it through a 3 micron filter bank and sends it back to the pool. This system is operated manually as needed.

5.4 Core Diffuser System

The core diffuser system pumps pool water through a nozzle above the core. With the water jet directed downward and tangential, the resulting core water circulation pattern delays the rise of the ^{16}N and ^{41}Ar gases produced by activation of dissolved oxygen and argon in the pool water. This minimizes radiation background at the pool surface. The diffuser system is used whenever the reactor is operated above 400 kW.

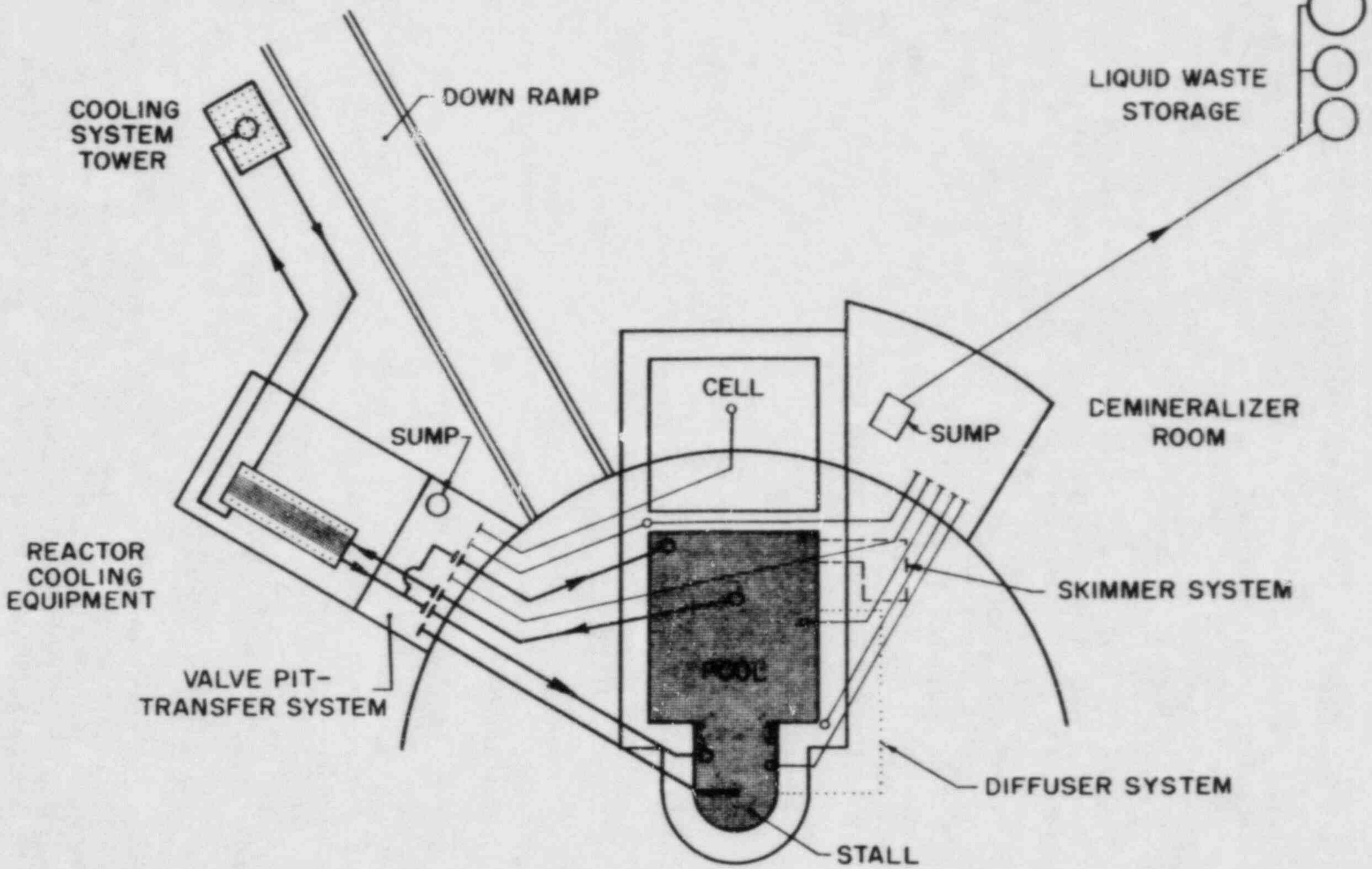


Figure 5.1 Texas A&M NSC reactor pool water systems

5.5 Conclusions

The staff concludes that the reactor coolant system is adequate to maintain fuel temperatures within safe limits during steady and pulsed operating conditions and that no failure of a component or combination of components will cause melting of a fuel element or cladding. Therefore, there will be no radioactive material released to the environment due to failure of the primary or secondary coolant systems.

In addition, the purity of the primary coolant system is deemed to be sufficiently high to minimize the corrosion considerations concomittant with the 40-year total expected life of the primary coolant system components.

6 ENGINEERED SAFETY FEATURES

The only engineered safety feature associated with the Texas A&M reactor facility is the ventilation system. The only significant airborne radioactive materials formed as a result of normal reactor operations are ^{41}Ar and ^{16}N. As the ^{16}N is considered to be decayed by the time the air enters the ventilation system, the only gas of concern is ^{41}Ar.

6.1 Ventilation System

The ventilation system consists of separate but interlocked supply and exhaust systems and their related control systems.

6.1.1 Supply System

Four air supply-circulation units and a central exhaust system control pressure, temperature, and humidity within the reactor building. The four air supply-circulation units simultaneously supply outside air to the reactor building and recirculate the conditioned building air. The amount of outside air delivered to the reactor building is manually set by the adjustment of remote controlled dampers. The central exhaust system operates separately from the supply-circulation units to collect and exhaust air from the building through automatic controlled exhaust dampers. The central exhaust system automatically maintains a negative pressure in four defined building pressure zones. The four zones are delineated as follows with Zone I at maximum negative pressure and Zone IV the least negative pressure:

Zone I - Beam ports, thermal columns, through tubes
Zone II - Main research area (upper and lower)
Zone III - Control room
Zone IV - Locker room, rest rooms and electric shop

These zones are maintained under negative pressure to ensure the control of air flow and, therefore, radioactivity levels within the areas (see Figure 6.1).

The central exhaust system equipped with a single 5,000 cfm fan, draws air from each of the zones, past particulate and gaseous radiation monitors and discharges the air up an 85-ft stack. The four supply air-circulation units may be operated from a control panel in the reception room. The manual-remote controlled fresh air dampers are adjusted at the control panel to supply air volume within the automatic control range of the central exhaust system. The control panel is used also to close down both the supply and exhaust systems to isolate the reactor building.

6.1.2 Exhaust Filter System

An emergency bypass filter bank consisting of absolute particulate and carbon filters is downstream of the exhaust blower. Though normally bypassed, all

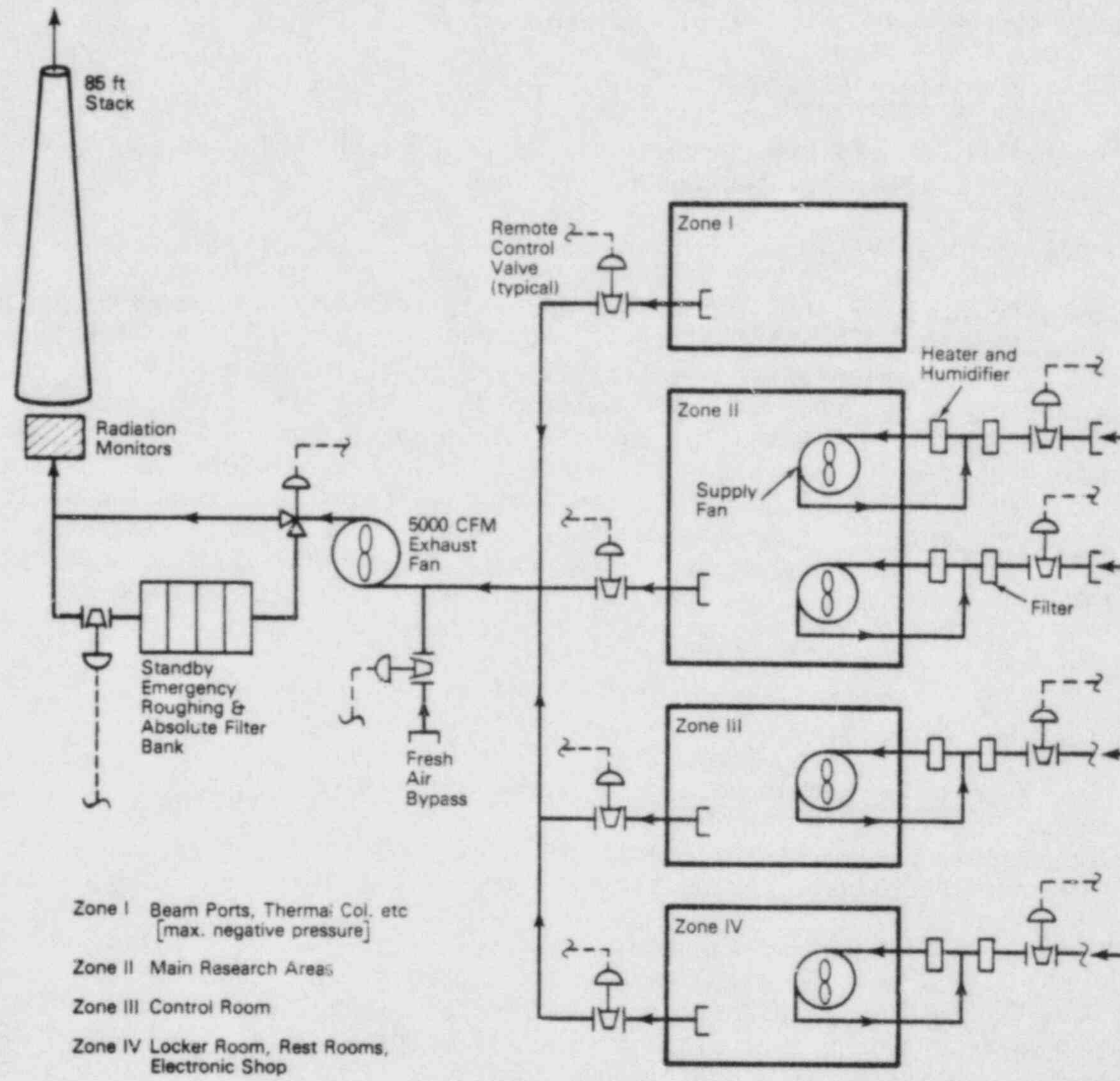


Figure 6.1 Ventilation system schematic

exhaust flow can be diverted through the filter bank by remotely operating the various damper valves.

6.1.3 Interlocking Controls

When abnormally high levels of radioactivity in the exhaust flow cause alarms by either the exhaust particulate or fission product monitors, the exhaust fan and the supply-circulation units are automatically stopped. In addition, the outside air supply dampers on the supply-circulation units, the central exhaust outside air bypass damper, and the exhaust stack damper close, isolating the reactor building. This ventilation system shutdown also can be initiated manually from the control room and by controls located on a panel in the facility reception area. All ventilation system controls located on the reception room panel can be operated in event of an emergency that causes evacuation of the other primary areas.

6.2 Conclusion

The reactor building ventilation system design and procedures are adequate to control the release of airborne radioactive effluents in compliance with the requirements of 10 CFR 20 and to minimize releases of airborne radioactivity in the event of accident conditions. Therefore, the staff concludes that the public will be adequately protected from airborne radioactive hazards related to reactor operations.

7 FACILITY CONTROL AND INSTRUMENTATION

The control and instrumentation systems provide the means for operating the various components of the Texas A&M reactor and experimental facilities in a safe and efficient manner. The control and instrumentation systems are shown in Tables 7.1 and 7.2.

Table 7.1 Minimum reactor safety circuits (Application Supplement 1982)

Safety Channel	Number operable	Function	Effective Mode S.S.	Effective Mode Pulse
Fuel element	1	Scram at LSSS*	X	X
High power level	2	Scram at 125%	X	
Console scram button	1	Scram	X	X
High power level detector power supply	2	Scram on loss of	X	
Preset timer	1	Transient rod scram 15 s or less after pulse		X
Log power	1	Prevent withdrawal of shim-safeties at 4×10^{-3} W	X	
Log power	1	Prevent pulsing above 1 kW		X
Transient rod	1	Prevent application of air unless fully inserted	X	
Shim-safeties and regulating rod position	1	Prevent withdrawal		X

*LSSS = limiting safety system setting

Table 7.2 Summary of information displayed and recorded on reactor console (Application Supplement 1982)

	Control	Indication	Record
Reactor safety systems			
Log N-power		X	X
Log N-period		X	
Linear power		X	X
Safety amplifier		X	
Pulse power (integrated)		X	
Fuel temperature		X	X
Rod drives	X	X	
Manual scram	X	X	
Other scrams		X	
Facility and reactor condition alarms		X	
Water systems			
Pool water cooling	X	X	X
Pool recirculation	X	X	
Pool skimmer	X	X	
Diffsuer	X	X	
Transfer	X	X	
Secondary treatment	X	X	X
Personnel control and radiation protection			
Area radiation monitors		X	
Facility air monitors		X	X
Air handling system shutdown	X	X	
Emergency evacuation horn	X	X	
Irradiation cell exhaust	X		
Television monitors		X	
Facility "door open" alarm	X		
Experimental Facilities			
Pneumatic system	X	X	
Motor rotisserie	X	X	
"C-2" experiment personnel control alarm	X	X	

7.1 Control Systems

There are several different control systems within the complex, 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. The applicant indicated that care has been taken to physically isolate the power and control cables from the instrumentation wiring to avoid electrical noise on the instrumentation channels. Specific details of the various control systems are discussed in the following sections.

7.1.1 Reactor Control

The reactor is controlled from the operator's console by adjusting the control rods. Rod position is indicated by synchronous transmitter and limit-switch circuitry with digital readout and limit-indicating lights at the control console. The readout indicators are conveniently placed for direct operator observation. There are six motor-driven control rods consisting of four shim-safety rods, a regulating rod, and a transient rod.

The shim-safety rods can be operated singly or in combination. The transient rod also is actuated by a pneumatic actuator to produce reactivity pulses. No emergency backup power system is required because a loss of power causes the control rods (except the regulating rod) to drop into the reactor core by gravity producing an automatic reactor scram. The decay heat can be dissipated by the reactor pool water sufficiently to preclude temperature rises that would cause fuel or cladding melt.

7.1.2 Core Support Carriage Control

The reactor core, the control rod drives, the ion chamber cannisters, and the diffuser system are supported by a bridge that spans the reactor pool. The bridge is mounted on four wheels and travels on rails provided outside the core pool and facilitates lateral movement of the core and appurtenances. The bridge is hand operated, and its speed of travel is limited by mechanical gearing. This precludes the possibility of inadvertent reactor movement because of system failure. Electric power, control-circuit wiring, and compressed air are supplied to the bridge. Cable slack for the bridge movement is provided by a cable that lies in a covered trough that is parallel to the reactor pool.

7.1.3 Reactor Scram System

Scram circuits are indicated in Table 7.1. A reactor scram occurs by removing the power to electromagnets in the safety-shim rods and in the transient rod actuation system, which allows the rods to drop into the reactor core by gravity.

Any of the following occurrences will cause an open circuit and a loss of power to the electromagnets and produce a reactor scram:

(1) electric power failure

(2) reactor period of 3 sec or less (not required in the Technical Specifications)

(3) maximum are fuel temperature of 950° C or more for FLIP fuel elements (this is protected by a limiting safety system setting of 525° C in the instrumented fuel element)

(4) manual scram

(5) bridge lock scram

(6) experiment scrams including the radiography cave door and irradiation cell door

(7) reactor power above set point (125% or less of expected full power)

7.1.4 Ventilation System Controls

A detailed description of the ventilation system is provided in Section 6.1.

The supply and exhaust systems are interlocked to operate as one system. Following setting of the pressure (i.e., negative) desired, the supply and exhaust systems automatically adjust until the desired pressure setting is achieved. Controls are located in the reactor control room, and emergency controls are replicated in the facility reception area. A fission product monitor on the bridge over the reactor pool and an exhaust stack particulate radioactivity alarm will automatically annunciate in the control room and shut down the ventilation system.

7.1.5 Reactor Cooling System

Reactor pool water is circulated through the tube side heat exchanger. A secondary loop cools the heat exchanger using a cooling tower. The water pressure in the shell side is higher than in the tube side so that the reactor pool water will not reach the cooling tower if a leak developed between the two systems. Water in both cooling loops is treated to purify the water to appropriate levels. The pumps are controlled from the control room.

The reactor cooling system control panel is located in the reactor control room. Procedures require that before operation of the system, valve positions must be established and verified by the reactor supervisor. The system is controlled by the reactor operator using the start-stop switches to the cooling tower fan, primary pump, and secondary pump. A multipoint temperature recorder and primary flow rate indicator provide continuous performance data for the system, and a conductivity meter is used for readings of the bulk pool water purity. A graphic panel also is provided for display of the cooling system flow schematic. Alarms are provided for primary pump failure, secondary pump failure, and loss of secondary flow.

Makeup water is added to the primary coolant pool manually as needed to maintain the appropriate water level. The level is maintained in the secondary coolant loop through a float-operated valve.

7.2 Instrumentation System

The instrumentation system consists of nuclear and nonnuclear components, annunciators, readout devices, digital indicators, chart recorders, meters, and gauges. In addition, there are several radiation monitors for radiation protection purposes.

7.2.1 Reactor Nuclear Instrumentation System

There are five permanently installed channels for reactor nuclear instrumentation. Four of these are used in the steady-state mode; the other is used in the transient mode. A brief description of these channels is provided below.

7.2.1.1 Log Power Channel

The log power channel consists of a fission chamber detector, preamplifier, amplifier, rate meter, and log power recorder. The log power channel has a range of 10 decades of reactor power, and interlocks are provided to prevent startups without at least 2 counts per second and to prevent pulsing at powers above 1 kW. An optional scram may be activated in the event of a reactor period of 3 sec or less.

7.2.1.2 Linear Power Channel for Steady-State Operation

The linear power channel consists of a compensated ion chamber, a linear picoammeter, digital power readout, linear power recorder, and a servo controller. The detector is positioned above a tapered graphite reflector element that scatters the neutron flux from the core face to the detector.

This configuration provides excellent linearity and significantly reduces the contribution resulting from gamma rays so that the system is sensitive and accurate at low power levels even after extended operation at 1 MW. This channel may be connected to the servo controller, which operates the regulating rod to maintain a constant power level during operation. A permit switch allows manual or automatic operation when the reactor power level reaches +5% of the set point on the linear recorder.

7.2.1.3 Safety Power Measuring Channel (2 detectors)

The safety power channel amplifier consists of two identical, isolated sections, each consisting of an uncompensated ion chamber detector, a linear amplifier, two bistable trips, and power supplies. This amplifier is the primary component of the safety circuitry in the reactor control system. The instrument supplies current to the control rod magnets and provides the mechanism for scramming the reactor. This reactor scram level is set at 125% or less of full authorized reactor power.

Additional scrams that are connected in the external scram chain are the limiting safety system setting (LSSS), the manual scram, the bridge lock scram, and various experiment scrams. Experiment scrams are provided in areas where an accident or other unusual circumstances could cause the individuals working with the experiment to receive high radiation exposures unless the reactor were rapidly scrammed.

7.2.1.4 High Level Pulsing Channel

The reactor can be switched to pulse-mode operation if the steady-state reactor operation is less than 1 kW. If this pulsed mode is selected, the normal neutron channels are disconnected and the high-level pulsing chamber becomes the monitoring channel. The pulsing chamber is a gamma or neutron ion chamber that is adjacent to the reactor core. The output is routed to an integrator circuit that provides a digital display of the integrated pulse power. The integrator also provides outputs to record the reactor power level and the integrated power as a function of time.

7.2.2 Temperature Measurement Channels

These channels are used to monitor the temperature of the pool water, the irradiation cell, and the instrumented fuel element. The system is composed of thermocouples, a selector switch, and a digital thermocouple indicator connected to a temperature recorder. The recorder is a part of the scram system and will provide a scram signal if the instrumented fuel element temperature reaches 525° C.

7.2.3 Building Exhaust Monitoring and Area Monitoring Equipment

The air that is exhausted from the building is continuously monitored for ^{41}Ar activity. This activity is monitored with a gas detector that uses a 3-in. NaI (Tl) scintillation crystal and a gamma spectrometer. The detector, which is calibrated for ^{41}Ar activity, continuously samples air from the building exhaust plenum. The system is equipped with an adjustable contact that provides an audible alarm on the console and a warning light on the console and in the reception room. In addition, the exhaust stack is monitored for particulate activity with a moving-tape-type continuous monitor. This monitor samples air from the building exhaust plenum and is equipped with an alarm circuit that activates an audible alarm and a warning light indicating the alarming channel. The circuit also causes an automatic shutdown of the air-handling system to isolate the facility.

An area radiation monitoring system provides a continuous indication at the reactor console and in the reception room of the radiation level in each of the monitored areas. An adjustable contact on each indicating meter provides an alarm on the console annunciator panel. A warning light on the indicating meter and on the detector identifies the particular area involved.

7.2.4 Nonnuclear Instrumentation

The nonnuclear instrumentation used for personnel protection and in the ventilation and cooling systems is described below. A summary of the information displayed and recorded on the reactor console is listed in Table 7.2.

- o All electromechanical devices, such as relays and servo mechanisms, are rated for service parameters in excess of those actually experienced.
- o Several of the more crucial circuits are self-checking, and the reactor is provided with interlocks and scram modes in the event of internal malfunctions or failures of these circuits.
- o Redundant instrumentation provides operations personnel with a variety of information during reactor operations.
- o The primary instrumentation and control systems are supplemented with experiment instrumentation and controls that are interlocked and alarmed to provide extra safety margins.

- Components such as power supplies, preamplifiers, and amplifiers are all constructed to performance specifications in excess of those required.

- The reactor is provided with a minimum of 10 safety circuits, all of which must be functional to prevent automatic reactor scram. Table 7.1 lists the minimum required safety circuits.

- The control console is provided with indicators, annunciators, and recording devices to ensure operator awareness of a wide variety of parameters. Table 7.2 summarizes the displayed and recorded information.

Accordingly, the staff believes that the control and instrumentation systems, coupled with administrative devices such as the facility component checklist (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.

7.2.4.1 Cooling System

A multipoint recorder located in the control room provides a continuous record of cooling system temperatures and flow rates. A conductivity meter is used to monitor the purity of the primary cooling system water. A graphic panel in the control room displays the status of the cooling system, and alarms are provided for pump failure and loss of secondary flow.

The reactor pool level is visible from the control room. Instrumentation that alarms on the reactor console when the pool level drops 10% below the normal operating level, is available also.

7.2.4.2 Ventilation System

The status of all supply and exhaust fans is displayed in the reception room of the reactor building. All air-handling system fans indicate a loss of power, and the building pressure failure is annunciated in the reactor control room.

7.3 Conclusions

The staff concludes that the control and instrumentation systems at Texas A&M are well designed and maintained and that 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.

8 ELECTRICAL POWER

8.1 Main Power

The Texas A&M reactor facility's electrical power is supplied by a substation on the main campus power system.

8.2 Emergency Power

Battery-powered emergency lights are activated automatically if there is an ac power failure. No reactor system is supplied with emergency power because a power failure automatically shuts down the reactor, and adequate core cooldown is provided by the reactor pool water.

8.3 Conclusion

The staff concludes that the electrical system is acceptable for safe reactor operation and that additional emergency power capability is unnecessary.

9 AUXILIARY SYSTEMS

9.1 Ventilation System

The ventilation system is considered an engineered safety system and is described in Section 6.

9.2 Fuel Handling and Storage

New fuel elements are stored in cadmium-lined tubes in the fuel storage room, which is located under the ramp up to the reactor room. The number and spacing of the storage tubes ensure a k_{eff} less than 0.8 for all possible conditions of moderation. A criticality meter and alarm are provided in accordance with 10 CFR 70.24(a).

Irradiated fuel elements are stored under water in the reactor pool in racks mounted on the pool walls or in a covered rack attached to the pool floor. The geometry of the pool storage facilities ensures a k_{eff} of less than 0.8 when it is completely filled with elements of the greatest activity possible.

Refueling, fuel bundle maintenance, and fuel inspection are performed under water in the reactor pool, using specially designed hand-held tools.

9.3 Compressed Air System

There are two separate compressed air systems. A large compressor supplies air at $\sim$ 100 psi throughout the facility for general use. A smaller compressor supplies dry, filtered air at $\sim$ 20 psi to pneumatic instruments and air motors. A line takes air from the facility air system, passes it through a filter, dryer, and regulator and supplies it to the reactor control system for pulsing operations.

9.4 Fire Protection System

Fire protection is provided at the Texas A&M reactor complex by numerous fire extinguishers that are located throughout the site. Smoke detectors are located throughout the NSC complex.

The College Station Fire Department provides the reactor complex with fire protection services and is on call 24 hours a day. Fire department personnel receive training on an annual basis to become familiar with radiological hazards and the reactor facility.

9.5 Liquid Waste Collection System

The liquid waste collection system is described in detail in Section 11.2.2. All liquid radwaste drains to a central area containing three holdup tanks. Samples are taken from the tanks to determine subsequent means for handling or discharging the liquid waste.

9.6 Conclusion

The staff concludes that the Texas A&M reactor facility's auxiliary systems are adequate to support reactor operation in a safe and reliable manner.

10 EXPERIMENTAL PROGRAMS

The Texas A&M reactor serves as a source of ionizing and neutron radiation for research and isotope production. In addition to inpool irradiation capabilities, the experimental facilities include a pneumatic transfer system, a thermal column, several beam ports, and an irradiation cell.

10.1 Experimental Facilities

10.1.1 Pool Irradiations

The open pool of the reactor permits experiments submerged in the vicinity of the core to be irradiated. The decision to perform experiments in the reactor pool--as opposed to using the pneumatic transfer system or a beam tube--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 controlled by its effect on the reactivity and is limited by the Technical Specifications.

10.1.2 Pneumatic Transfer Systems

The pneumatic transfer system allows small, sealed samples to be rapidly transported between the reactor and the radiochemistry laboratories. The irradiation terminus is in the reflector region adjacent to the core, and the receiver terminus is a shielded box in a ventilated hood in any of several laboratories. Carbon dioxide is used to move the samples and at the same time to minimize ^{41}Ar production.

10.1.3 Thermal Column

The thermal column is a stack of machined graphite blocks within the pool shield wall. With the reactor positioned against the inner face of the thermal column, the graphite moderates the energetic neutrons escaping from the reactor to provide an external beam of thermal neutrons for experimental use.

The building ventilation system maintains a negative pressure on the thermal column so that air flows into the cavity and is discharged by the reactor complex exhaust stack after being monitored. The thermal column has been modified to provide three additional beam ports.

10.1.4 Beam Ports

There are seven penetrations at various angles and heights through the pool shield wall, including a through tube. In addition (as noted above), the thermal column has three beam ports. These penetrations are normally filled with shielding material; however, the shielding can be removed to provide an external radiation beam for experimental use when the reactor is positioned in the stall end of the pool. These ports also can be used to place samples near the reactor for irradiation. One beam port is surrounded by a concrete block structure and is equipped with a hydraulic shutter and film holder and is used as a neutron radiography facility.

10.1.5 Irradiation Cell

A cell approximately 18 ft wide, 16 ft deep, and 10 ft high is located at core level at the west end of the reactor pool. The roof of the cell is constructed of 4 ft of concrete block with a 5- by 5-ft opening directly over the cell window. A motor-driven concrete shield 2 ft thick is installed over the opening.

Access is provided to the cell by an elevator that is raised and lowered from the upper research level by the overhead crane. The elevator can accommodate sample containers up to 51 in. wide by 49 in. deep by 73 in. high. With the exception of the elevator opening, the upper level of the cell is decked with steel plate. A small section of the deck plate is hinged to provide access to the emergency ladder, which runs from the upper level to the top of the concrete shield.

The irradiation cell window is cast into the 2-ft-thick wall that separates the cell from the reactor pool. The window is 2 ft^2 on the pool side and flares out to 4 ft^2 on the cell side. A 1/2-in. aluminum plate is bolted to the pool side of the cell window to provide a watertight barrier. The cavity formed by an aluminum gasketed plate attached to the window on the irradiation cell side is used as a water shutter. A plenum at the bottom of the plate allows for filling and dumping of the water that is collected within the shutter. A console on the upper research level floor area provides controls for filling, dumping, and determining the level of water within the shutter. Annunciator lights on the shutter console and in the control room indicate the water level in the shutter.

Electrical power for the motor-driven shield is supplied through a breaker and a reversing switch to open or close the cell. The breaker can be locked to control opening of the shield. Mechanical stops are always attached on the reactor bridge rail to prevent inadvertent movement of the reactor to closer than 8 ft from the irradiation cell window. Also, a bridge interlock scram is provided in case the irradiation cell door is opened when the reactor is positioned within 8 ft of the cell.

An exhaust duct extends to the bottom of the cell for continuous removal of air from the cell to remove ^{41}Ar activation in the cell. The duct discharges to the central building exhaust ahead of the stack gas monitor. Thus, all air from the cell will be monitored before it is discharged to the outside environment. The controls for the cell air exhaust are located on the reactor console. Also, an experiment scram button and an intercom are located inside the cell.

10.2 Experimental Review

Before any new experiment using the reactor or experimental facilities can be conducted, it is reviewed by the Texas A&M Reactor Safety Board. This 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.

10.3 Conclusion

The staff concludes that the design of the experimental facilities combined with the detailed review and administrative procedures applied to all research activities is adequate to ensure that experiments (1) are unlikely to fail, (2) are unlikely to release significant radioactivity to the environment directly, and (3) are unlikely to cause damage to the reactor's 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

The major radioactive waste generated by reactor operations is activated gases, principally ^{41}Ar. A limited volume of radioactive solid waste, primarily resins and filter materials, is generated by reactor operations, and some additional solid waste is produced by the associated research programs. No radioactive liquid wastes are generated directly by normal reactor operations. However, liquid radioactive waste is produced by the regeneration of the resin bed in the water demineralizer system. Additional small amounts of radioactive liquid waste are developed as a result of several of the Texas A&M research activities.

11.1 As Low As Reasonably Achievable (ALARA) Commitment

The Texas A&M reactor is operated with the philosophy of minimizing the release of radioactive material to the environment. The University administration instructs all research personnel to develop procedures to limit the generation and subsequent release of radioactive waste materials.

11.2 Waste Generation and Handling Procedures

11.2.1 Solid Waste

Solid waste generated as a result of reactor operations and experiments consists primarily of ion exchange resins and filters, potentially contaminated paper gloves, glassware, and occasional small, activated components. This solid waste generation has typically contained a few millicuries of radionuclides per year.

The solid waste is collected and stored by the Health Physics staff to accumulate a sufficient volume before being transferred to the Radiological Safety Officer for disposal. Packaging and shipping in approved containers to an approved disposal site are conducted according to applicable Federal and state regulations.

11.2.2 Liquid Waste

Normal reactor operations produce no radioactive liquid waste. However, some of the research activities conducted within the reactor complex are capable of generating such waste. Liquid-waste drains in the reactor building and equipment areas empty into the demineralizer room sump, and the liquid is pumped to one of three holdup tanks; thus, there is no direct flow into the Texas A&M sanitary sewer system or to the environment. Other laboratories and experimental areas in the reactor complex where radioactivity may be used also are provided with waste lines that flow into this sump. Thus, all potentially contaminated liquids are collected finally in the 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 20, the contents are discharged to a nearby creek. If the concentrations are initially above 10 CFR 20 levels, the contents of the tank are diluted to below those levels before discharge.

The reactor complex is capable of solidifying small volumes of highly contaminated liquid for shipment offsite as solid waste.

11.2.3 Airborne Waste

No fission products escape from the fuel cladding during normal reactor operations. The potential airborne wastes are gaseous ^{41}Ar and neutron-activated dust particulates, which are produced principally by the neutron irradiation of dissolved gases in the pool water and by air and airborne particulate materials in the thermal column and beam ports. This air is constantly swept from the experimental areas and discharged to the environment through the reactor facility stack.

If an emergency should occur, the reactor building exhaust system diverts the air to the bypass absolute and charcoal filter system that collects more than 99.9% of the particulate matter and most of the iodines. These filters would eventually be disposed of as potentially solid radioactive waste.

A stack monitoring system measures the stack gaseous and particulate concentrations in the effluent. During normal operations nc measurable radioactive particulates or iodines are released in the air effluents from the reactor complex stack. Texas A&M personnel have measured the release of ^{41}Ar over the years with gas sampling instruments calibrated with known quantities of ^{41}Ar. From annual operating reports, the annual release of ^{41}Ar measured at the stack has been a fraction of the maximum permissible concentration (MPC) in restricted areas and 2-3% of the MPC at the boundary of the facility. Total gaseous activity discharged is about 2-4 Ci annually. Environmental surveys (Texas A&M Environmental Impact Appraisal, 1979) indicate doses at the boundary that are small fractions of 10 CFR 20.

11.3 Conclusion

The staff concludes that the waste management activities of the Texas A&M reactor facility have been conducted and are expected to continue to be conducted in a manner consistent with 10 CFR 20 and with ALARA principles. Among other guidance, the staff review has followed the methods of ANSI/ANS 15.11, 1977.

Because ^{41}Ar is the only potentially significant radionuclide released by the reactor to the environment during normal operations, the staff has reviewed the history, current practice, and future expectations of operations. The staff concludes that the doses in both restricted and unrestricted areas as a result of actual releases of ^{41}Ar have been small fractions of MPC limits specified in 10 CFR 20, Appendix B, when averaged over a year. Furthermore, the staff's conservative computations of the dose beyond the limits of the reactor complex and the environmental surveys performed by the applicant give reasonable assurance that potential doses to the public as a result of ^{41}Ar would not be significant, even if there were a major change in the operating schedule of the Texas A&M reactor.

12 RADIATION PROTECTION PROGRAM

Texas A&M University has a structured radiation safety program with a Health Physics staff equipped with radiation detection equipment to determine, control, and document occupational radiation exposures at its reactor facility. In addition, the reactor complex monitors both liquid and airborne effluents at the points of release to comply with applicable regulations. Texas A&M also has developed an environmental monitoring program to verify that radiation exposures in the unrestricted areas around the facility are within regulations and guidelines and to confirm the results of calculations and estimates of environmental effects resulting from the Texas A&M reactor research programs.

12.1 ALARA Commitment

As stated in Section 11.1, the Texas A&M administration has formally established the policy that all operations are to be conducted in a manner to keep all radiation exposures ALARA. All proposed experiments and procedures at the reactor are reviewed for ways to minimize the potential exposures of personnel. All unanticipated or unusual reactor-related exposures are investigated by both the Health Physics and the operations staff to develop methods to prevent recurrences.

12.2 Health Physics Program

12.2.1 Health Physics Staffing

The normal full-time Health Physics staff at the Texas A&M NSC consists of three professionals and several technicians, one professional, one graduate student, and one technician are located at the reactor facility. The onsite staff has sufficient training and experience to direct the radiation protection program for a research reactor. The Health Physics staff has been given the responsibility, the authority, and adequate lines of communication to provide an effective radiation safety program.

The Texas A&M University's Health Physics staff provides radiation safety support to the entire university complex, including an accelerator complex and many radioisotope laboratories. The staff believes that the Health Physics staff at the reactor facility is adequate for the proper support of the research efforts within that facility.

12.2.2 Procedures

Detailed written procedures have been prepared that address the Health Physics staff's various activities and the support that it is expected to provide to the routine operations of the Texas A&M reactor. These procedures identify the interactions between the Health Physics staff and the operational and experimental personnel. The procedures specify numerous administrative limits and action points, and they specify the appropriate responses and corrective action to be taken if these limits or action points are reached or exceeded. Copies of these procedures are readily available to the operational and research staffs and the Health Physics and administrative personnel.

12.2.3 Instrumentation

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

12.2.4 Training

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

12.3 Radiation Sources

12.3.1 Reactor

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

The reactor fuel is contained in stainless-steel cladding. Radiation exposures from the reactor core are reduced to acceptable levels by water and concrete shielding, and personnel are normally restricted from the immediate vicinity of the reactor pool during pulsed operation. The ion exchange resins and filters are routinely changed before high levels of radioactive materials have accumulated, thereby minimizing personnel exposure.

Concentrations of ^{16}N 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 sec) of the ^{16}N to reduce further the amount of radioactivity released into the reactor room. Personnel exposure to the radiation from chemically inert ^{41}Ar is limited by dilution and prompt removal of this gas from the reactor room and experimental areas and its discharge to the atmosphere, where it diffuses further before reaching occupied areas.

12.3.2 Extraneous Sources

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

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

12.4 Routine Monitoring

12.4.1 Fixed-Position Monitors

The Texas A&M NSC reactor facility uses several fixed-position radiation monitors and constant air monitors. These include a gas monitor and an air particulate monitor in the reactor building and a fission product monitor on the bridge above the reactor. Area radiation monitors are located at strategic points throughout the building in regions where radiation levels might increase and reflect an abnormality or hazard in operations. All monitors have adjustable alarm set points and read out in the control room.

12.4.2 Experimental Support

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

12.4.3 Special Work Permits

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

12.5 Occupational Radiation Exposures

12.5.1 Personnel Monitoring Program

The Texas A&M reactor facility personnel monitoring program is described in the Radiation Safety Instructions. To summarize the program, personnel exposures are measured by the use of film badges and thermoluminescent dosimeter (TLD) finger badges assigned to individuals who might be exposed to radiation. In addition, TLDs and self-reading pocket dosimeters and instrument dose rate and time measurements are used to achieve administrative occupational exposure limits of 100 mrems per 2-week exposure period, which easily complies with applicable limits in 10 CFR 20.

12.5.2 Personnel Exposures

The Texas A&M reactor personnel annual exposure history for the last 5 years is given in Table 12.1. Over 95% of the Texas A&M reactor facility personnel receiving exposure are in a range of less than 20% of the limits of 10 CFR 20.

Table 12.1 Number of individuals in exposure interval

	Number of individuals in each range				
Whole body exposure range (rems)	1977	1978	1979	1980	1980
No measurable exposure	5	37	9	18	15
Measurable exposure less than 0.1	16	13	30	19	29
0.1 to 0.25	15	4	4	9	6
0.25 to 0.5	2	2	2	5	6
0.5 to 0.75	5	0	4	1	2
0.75 to 1	1	0	0	0	0
1 to 2	1	2	1	0	3
2 to 3	0	0	0	1	0
Number of individuals monitored	45	58	50	53	61

Source: Texas A&M Annual Operating Reports (1977-1981).

12.6 Effluent Monitoring

12.6.1 Airborne Effluents

As discussed in Section 11, airborne effluents from the reactor facility consist principally of ^{41}Ar from activation of gases.

The stack monitoring system measures the radioactive gases and airborne particulates discharged from the entire reactor complex. The only identifiable radioactive gas is ^{41}Ar. The system consists of a sampling line positioned in the exhaust duct at the base of the stack, a sampling pump to maintain a constant flow, a filter to remove particulate contaminants, and a scintillation detector positioned in the middle of a known volume. The instrumentation read-out consists of rate meters located in the control room and reception room and a strip-chart recorder in the control room. The detector count rate is proportional to the amount of radioactive gases in the chamber and hence to the concentration in the air stream. High concentrations activate alarms in the control room. This gaseous monitoring system is periodically calibrated by releasing a small, known quantity of ^{41}Ar into the stack effluent stream. ^{41}Ar

effluent releases for reporting year 1981 are indicated in Table 12.2. The maximum permissible concentration (MPC) for ^{41}Ar is 2×10^{-8} μc/cc. Table 12.2 indicates discharge concentrations of approximately 1/100 MPC at the discharge stack and concentrations of less than 1/1000 at the fence.

Table 12.2 ^{41}Ar effluent releases for 1981

Month	Exhaust volume (cc)	Concentration* (μCi/cc)	Concentration** (μCi/cc)	Percent MPC**	Total radioactivity (Ci)*
January	6.31×10^{12}	3.89×10^{-8}	1.95×10^{-10}	4.86×10^{-3}	2.45×10^{-1}
February	5.91×10^{12}	1.20×10^{-9}	6.00×10^{-12}	1.50×10^{-4}	7.09×10^{-3}
March	6.31×10^{12}	9.48×10^{-9}	4.74×10^{-11}	1.19×10^{-3}	5.98×10^{-2}
April	6.12×10^{12}	1.20×10^{-9}	6.00×10^{-12}	1.50×10^{-4}	7.34×10^{-3}
May	6.31×10^{12}	1.20×10^{-8}	6.00×10^{-11}	1.50×10^{-3}	7.57×10^{-2}
June	6.12×10^{12}	7.12×10^{-9}	3.56×10^{-11}	8.90×10^{-4}	4.36×10^{-2}
July	6.31×10^{12}	9.40×10^{-11}	4.70×10^{-13}	1.18×10^{-5}	5.93×10^{-4}
August	6.32×10^{12}	2.48×10^{-13}	1.24×10^{-15}	3.10×10^{-8}	1.56×10^{-6}
September	6.12×10^{12}	5×10^{-8}	2.50×10^{-10}	6.25×10^{-3}	3.06×10^{-1}
October	6.31×10^{12}	3×10^{-10}	1.50×10^{-12}	3.75×10^{-5}	1.89×10^{-3}
November	6.12×10^{12}	3×10^{-10}	1.50×10^{-12}	3.75×10^{-5}	1.84×10^{-3}
December	6.31×10^{12}	3×10^{-10}	1.50×10^{-12}	3.75×10^{-5}	1.89×10^{-3}

Total volume $7.45\ 10^{13}$ cc
Annual average release* $1.01\ 10^{-8}$ μCi/cc
Total radioactivity released* $7.51\ 10^{-1}$ Ci

*As measured in the central exhaust stack
**As determined at 100 meters, approximate boundary of exclusion area, with 200/1 dilution factor (SAR, pp 117-119, June 1979)
Source: Texas A&M Annual Operating Report 1981

Radioactive airborne particulates in the exhaust stream are monitored using a moving-tape-type instrument; a representative sample drawn from the exhaust plenum is pulled through a moving-tape filter past an end-window Geiger-Muller tube. A high-level alarm will produce an automatic shutdown of the air-handling system, which isolates the facility.

In the emergency mode, the effluent stream is diverted to a standby roughing and absolute filtering system and charcoal filters in series to remove most particulate material before discharge to the environment through the reactor building stack. Details of this system are discussed in Section 6.

The staff concludes that monitoring procedures of airborne effluents that are small fractions of the MPC specified in 10 CFR 20, are acceptable.

12.6.2 Liquid Effluent

The reactor generates very limited radioactive liquid waste during routine operations. Sources of liquid waste are described in Section 11.2.2. All potentially contaminated liquids are collected in holdup tanks as shown in Figure 12.1. When full, each tank is isolated, mixed, and sampled. The sample is analyzed, and liquids with a low concentration of radioactivity are released directly to the environment in accordance with 10 CFR 20.303. Higher concentrations of liquid waste may be diluted for release or held for radioactive decay, or they may be solidified and handled as solid waste.

The staff concludes that monitoring procedures for liquid effluents, which indicate discharges that are less than specified in 10 CFR 20, are acceptable.

12.7 Environmental Monitoring

Texas A&M University and the Texas Department of Health have developed a program to monitor radiation from reactor operations in the surrounding environment.

The program uses six stations that have been established around the perimeter of the reactor facility and the collection, analysis, and evaluation of soil, water, vegetation, and milk samples. In addition, 11 TLDS are used to measure the external radiation exposures, which are compared with measurements made southeast of Easterwood Airport and approximately 800 m east of the reactor facility.

In addition, vegetation and water samples are collected from the Texas A&M reactor creek, White Creek, the upper and lower Brazos River, and the sanitary outflow. These samples are analyzed by the Texas Department of Health for gross gamma and gross beta radioactivities. Isotope identification is attempted for samples exhibiting unusual levels of activity. From Texas A&M Annual Operating Reports, sample analyses indicate doses that are small fractions of permissible levels specified in 10 CFR 20. The 1980-81 report is shown in Table 12.3.

12.8 Potential Dose Assessments

The Texas A&M Annual Report for 1981 indicates that approximately 4.7 Ci of ^{41}Ar are released annually. Texas A&M personnel using a dilution factor of 5×10^{-3}, obtained a concentration at the property line of less than 10^{-13} μc/cc, or approximately 0.8% of the MPC for ^{41}Ar as specified in 10 CFR 20, Appendix B, Table II.

Conservative calculations by the staff, based on the amount of ^{41}Ar released from the reactor complex stack, predict a maximum annual dose of only a fraction of a millirem in the unrestricted areas. These are verified by the

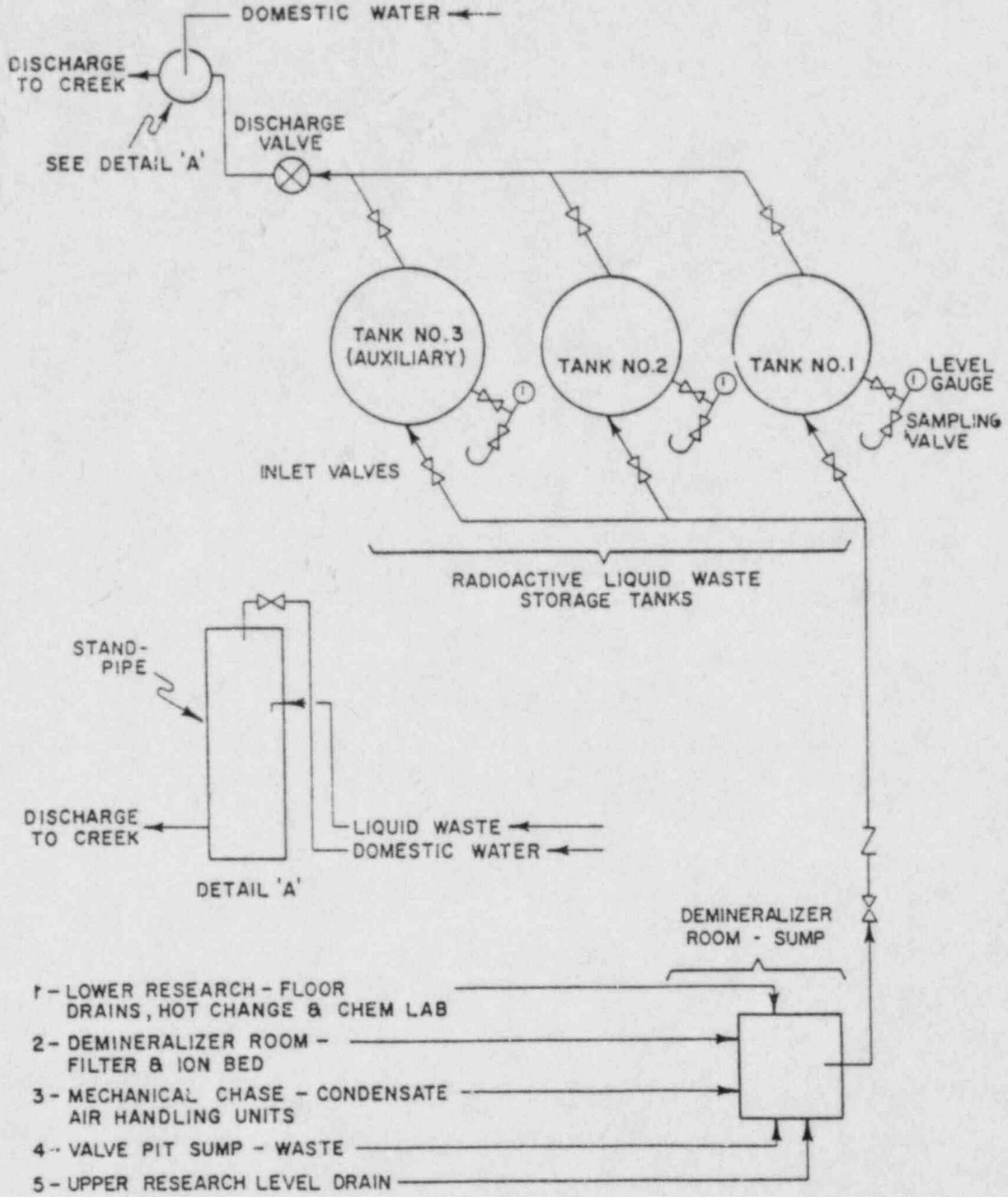


Figure 12.1 Radioactive liquid waste disposal system

Table 12.3 Environmental radiation monitoring program integrated radiation exposure October 15, 1980 to December 11, 1981

Station number	Location	Exposure (gross mR)	Exposure (net mR)	Average exposure rate (μR/hr)
1	NW corner - firemans training school	19	0	0
2	Fence corner west of TLD Station 4	112	51	4.98
3	Back fence south of TLD Station 2	73	29	4.11
4	West corner NSC* and calibration fence	108	47	4.59
5	Fence NSC front gate	81	20	1.95
6	East corner NSC and calibration fence	399	338	32.98
7	Easterwood Airport fence north of stock tank	36	0	0
8	Evergreen tree in open field west of calibration fence	24	5	1.42
9	Fence by trailers next to NSC	48	6	0.89
10	Fence 50 ft from TLD Station 9	52	10	1.49
11	Fence by aluminum gate by Easterwood Airport	66	5	0.49

*NSC - nuclear science center
Source: Texas A&M Annual Operating Report 1981.

environmental surveys, and, except for the radwaste storage building, the net radiation levels detected by the environmental radiation dosimeters located near the reactor facility have been indistinguishable from ambient background.

12.9 Conclusions

The staff considers that radiation protection receives appropriate support from the Texas A&M University administration. The staff concludes that (1) the program is properly staffed and equipped, (2) the Texas A&M reactor facility's

Health Physics staff has adequate authority and lines of communication, and (3) the procedures are integrated correctly into the research plans.

The staff also concludes that the effluent and environmental monitoring programs conducted by Texas A&M personnel are adequate to promptly identify significant releases of radioactivity and confirm possible effects on the environment, as well as to predict maximum exposures to individuals in the unrestricted area. These observed and predicted radiation levels are small fractions of the applicable limits to 10 CFR 20.

Additionally, the staff concludes that the Texas A&M 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. The staff, therefore, 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 Overall Organization Structure and Qualifications

The Texas A&M NSC is operated by the Texas Engineering Experiment Station (TEES). The facility is under the direct control of the Director of the NSC or a licensed senior operator designated by the Director to be in control. The Director of the NSC is responsible to the Director of the TEES for safe operation and maintenance of the reactor and its associated equipment. The Director of the NSC, who reports to the Director of the TEES through the Head of Nuclear Engineering Research (TEES), reviews and approves all experiments and experimental procedures before they are used in the reactor.

The safety of the operation of the Texas A&M NSC reactor, as it is related to the University Administration, is shown in Figure 13.1.

The Reactor Safety Board (RSB) reports to the Director of the TEES on all matters or policy pertaining to safety. The RSB consists of at least three members knowledgeable in fields related to nuclear safety. The University Radiological Safety Officer is an ex officio member of the RSB. The Safety Officer reviews, evaluates, and approves safety standards associated with operation and use of the Texas A&M reactor. Jurisdiction of the RSB includes all nuclear operations in the facility and general safety standards. A written chart for operations of the Reactor Safety Board includes provisions for

(1) at least annual meetings

(2) at least quarterly audits

(3) review and approval of new experiments in the reactor and changes to the facility, procedures, and Technical Specifications

(4) review of operations and abnormal occurrences

The Radiological Safety Officer provides onsite advice concerning personnel and radiological safety and provides technical assistance and review in the area of radiation protection.

13.2 Nuclear Science Center Organization

The Nuclear Science Center organization is shown in Figure 13.2.

The Director of the NSC has overall responsibility for ensuring nuclear safety and providing administration of the Texas A&M NSC.

The manager of reactor operations is responsible for the day-to-day operations of the reactor.

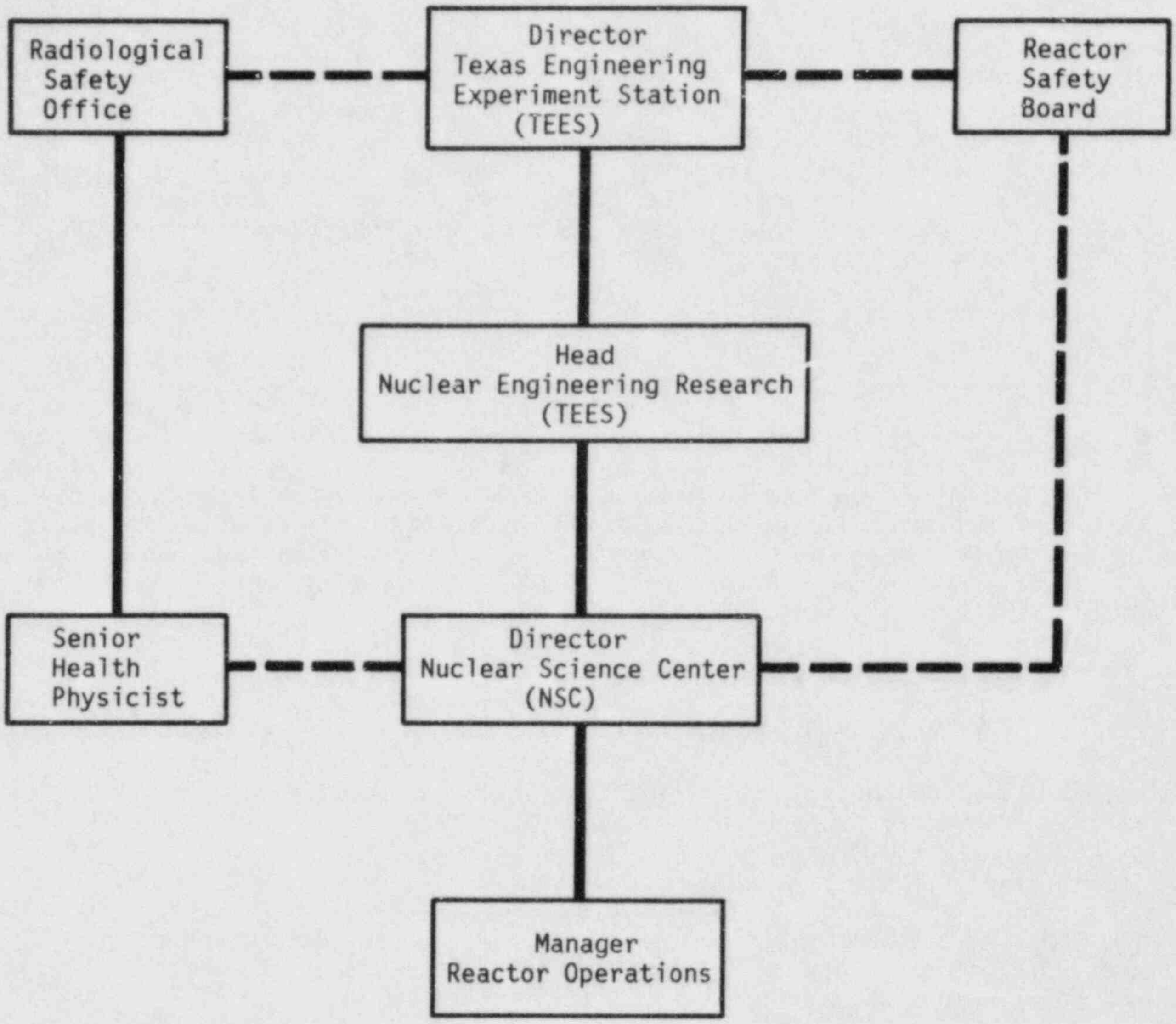


Figure 13.1 Administrative organization of Texas A&M Nuclear Science Center

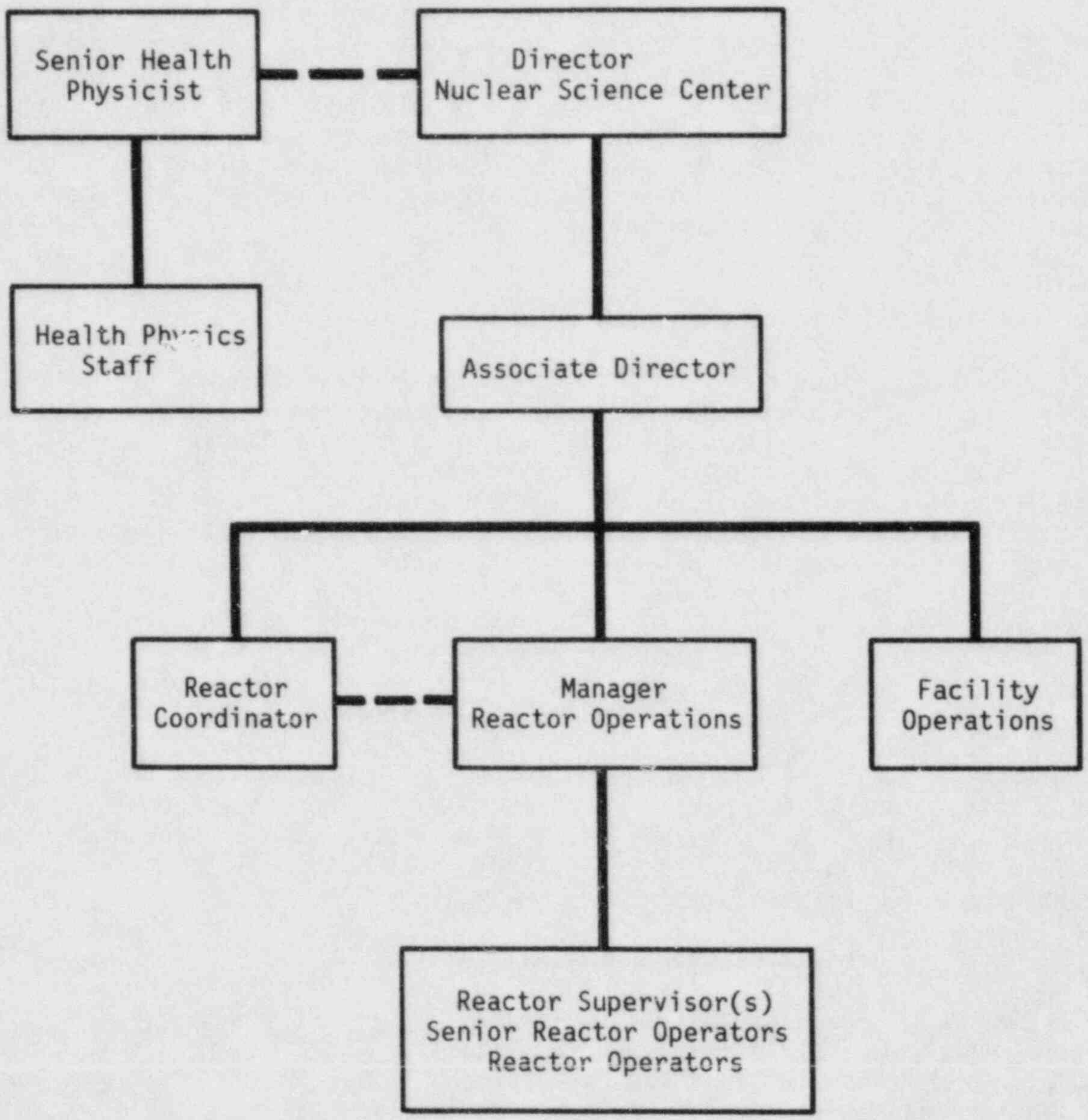


Figure 13.2 Nuclear Science Center administrative organization chart

The Radiological Safety Office (Senior Health Physicist) is responsible for providing onsite advice, technical assistance, and review in all areas related to occupational and radiological safety.

13.3 Training

A training program for reactor operations personnel exists to prepare personnel for the NRC Operator or Senior Operator examinations. This training program normally contains 20 hours of lecture and outside study and requires approximately 20 reactor startups. The training program includes a pulse operation training for requalification of reactor operations personnel. At the conclusion of the program, the Director or Associate Director of the NSC examines the trainees to ascertain whether or not they are qualified to take the NRC examination.

13.4 Emergency Response Plan

10 CFR 50.54 and Appendix E to 10 CFR 50 require that nonpower reactor applicants/licensees develop and submit emergency plans. The applicant submitted a plan that was developed following the recommended guidance in RG 2.6 (1979, For Comment Issue) and guidance in ANSI/ANS 15.16 (1978 Draft). However, RG 2.6 was reissued for comment in March 1982; and Draft 2 of ANSI/ANS 15.16 was reissued November 1981, which made it necessary for the applicant to resubmit his emergency response plans. Accordingly, by letter dated October 1982, Texas A&M submitted a revised Emergency Plan dated October 1982 for staff review and approval.

The staff reviewed the plan against the requirements of Appendix E to 10 CFR 50, the guidance criteria set forth in proposed Revision 1 to RG 2.6 and the ANSI/ANS-15.16.

Based on the review, the staff concludes that the Emergency Preparedness Plan for the Texas A&M University System, Nuclear Science Center Reactor, dated October 1982, as amended November 1, 1982, meets the requirements of the Commission's regulations and is acceptable.

13.5 Physical Security Plan

Texas A&M has established and maintained a program designed to protect the reactor and its fuel and to ensure its security. The NRC staff has reviewed the plan and visited the site. The staff concludes that the plan, as amended, meets the requirements of 10 CFR 73.67 for special nuclear materials of moderate strategic significance. Texas A&M'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 73.21.

13.6 Conclusion

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

14 ACCIDENT ANALYSIS

As part of its evaluation of several pending license renewals for nonpower reactors, the staff contracted with Battelle Pacific Northwest Laboratories to analyze generic reactor accidents for U-ZrH_x fueled reactors (NUREG/CR-2387) and contracted with the Los Alamos National Laboratory to evaluate the licensee's submitted documentation and analysis of potential site-specific events. These analyses included the various types of possible accidents and the potential consequences to the public.

Among the potential accidents considered to be credible, the one with the greatest effect on the environment and the unrestricted area outside of the Texas A&M reactor complex is loss of the cladding integrity of one irradiated fuel element in air in the reactor room. The staff will call this the fuel-handling accident. In more detail below, the staff has evaluated possible accident scenarios that originate in the intact core. None of these pose a significant risk of cladding failure. However, it is possible that an operator, while removing a fuel element from the core or relocating one previously removed after 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. This will be called 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 attempt to describe or evaluate the mechanical details of the accident or the probability of its occurrence. Only the consequences are considered. The following potential accidents or effects were considered to be sufficiently credible for evaluation and analysis.

(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

14.1 Rapid Insertion of Reactivity

This potential event is one in which the maximum excess reactivity readily available is inserted into the reactor instantaneously.

The theory of neutronic behavior of the U-ZrH_x fuel and all experimental measurements have shown that this fuel exhibits a strong, prompt, negative temperature coefficient of reactivity. For standard TRIGA fuel, 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 1,100° C, some local breaking of the bond and consequent dehydriding may occur (NUREG/CR-2387). However, if most of the fuel volume is below this temperature, the temperature coefficient not only terminates 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 that use U-ZrH_x fuel (GA-4314, 1980). Because of the action of the inherent temperature coefficient, temporary loss of positive reactivity will result from both steady-state and pulsing operations.

The staff concluded that the practical limitations of design and mechanical response, preclude the occurrence of any postulated theoretical events and consequences that could theoretically add enough excess reactivity under accident conditions to create an excursion that would not be thermally terminated before fuel damage occurred.

For standard fuel, the rise in temperature of the hydride increases the probability that a thermal neutron in the fuel element will gain energy from an excited state of an oscillating hydrogen atom in the lattice. As the neutrons gain energy from the ZrH_x, the thermal neutron spectrum in the fuel element shifts to a higher average energy (the spectrum is hardened), and the mean free path for neutrons in the element is increased appreciably. The average chord length is comparable with a mean free path and the probability of escape from the element before being captured is significantly increased as the fuel temperature is raised. In the water, the neutrons are rapidly thermalized again so that the capture and escape probabilities are relatively insensitive to the energy with which the neutron enters the water. The heating of the moderator mixed with the fuel in a standard TRIGA element thus causes the spectrum to harden more in the fuel than in the water. As a result, there is a temperature-dependent disadvantage factor for the unit cell in the core that decreases the ratio of absorptions in the fuel to total-cell absorptions as the fuel element temperature is increased. This brings about a shift in the core neutron balance, giving a loss of reactivity in standard fuel.

For a TRIGA-FLIP fuel element, the uranium loading is about 3.5 times that of a standard TRIGA element, and this causes the neutron mean free path in the FLIP element to be much shorter. For this reason, the escape probability for neutrons in the fuel is not greatly enhanced as the fuel-moderator material is heated. In the TRIGA-FLIP fuel the temperature-hardened spectrum is used to decrease reactivity through its interaction with a low-energy resonance material. Thus, erbium (with its double resonance at 0.5 eV) is used in the TRIGA-FLIP fuel both as a burnable poison and as a material to enhance the prompt negative temperature coefficient. The neutron spectrum shift pushes more of the thermal neutrons into the ^{167}Er resonance as the fuel temperature increases. As with a standard TRIGA core, the temperature coefficient is prompt because the fuel is mixed intimately with a large portion of the ZrH_x moderator; thus, fuel and solid moderator temperatures rise simultaneously, producing the temperature-dependent spectrum shift.

For these reasons, more than 50% of the temperature coefficient for a standard TRIGA core comes from the temperature-dependent disadvantage factor, or "cell effect," and 20% each from Doppler broadening of the ^{238}U resonances and temperature-dependent leakage from the core. These effects produce a temperature coefficient of $\sim -9.5 \times 10^{-5}$ $\Delta k/k/C°$ which is relatively constant with temperature

On the other hand, for a TRIGA-FLIP core the results of cell structure on the temperature coefficient are small. Almost the entire coefficient comes from temperature-dependent changes in ηf within the core and $\sim$ 80% of this effect is independent of the cell structure. Here, η is the average number of neutrons emitted per thermal neutron absorbed in fuel, and f is the ratio of thermal neutrons absorbed in the fuel to the total number of thermal neutrons absorbed everywhere in the reactor.

14.1.1 Excess Reactivity Scenario

The current reactor core is a full-FLIP core containing 91 fuel rods. Full FLIP cores containing up to 98 rods have been used in the past and may be used in the future. The staff knows of no credible method of rapidly inserting the total excess reactivity into the core. Therefore, the staff believes that the worst case credible nuclear excursion leading to maximum stressing of the fuel would be caused by the rapid insertion of 3.25\$ (2.3% $\Delta k/k$) of excess reactivity when the reactor is operating at a low steady-state power (less than 1 kW). This amount of reactivity corresponds to the worth of a FLIP fuel bundle in the most reactive position in the core and also to the worth of the transient rod if fully withdrawn.

The staff has considered the scenario where the reactor, operating at a given steady-state power level between 0 and 1 MW, has all of the remaining excess reactivity not compensated for by increased temperature inserted rapidly into the core. The staff has found that for 3.25\$ $\Delta k/k$ within the range of reactivity authorized at Texas A&M, the higher the initial fuel temperature at which the excess reactivity is inserted, the lower the maximum temperature attained immediately after the transient. This evaluation assumes that all loss of reactivity during the steady-state operation was a result only of the increase in temperature of the fuel. The known effect of ^{135}Xe poisoning was ignored, this assumption is conservative.

In a similar study using its BLOOST 2 code with experimental parameters and core power distributions for the Texas A&M reactor and assuming adiabatic processes, General Atomics (GA) calculated the rapid reactivity insertion required at a steady-state power of 1 MW to produce a peak core temperature of 950° C (Safety Analysis Report, 1979). For comparison, the corresponding value required for pulsing from 300 W was also computed. Calculations were performed for three different 98-fuel-rod cores: a full FLIP core and two combinations of mixed cores. In each case, the reactor required considerably more reactivity insertion from 1 MW to reach a final temperature of 950° C than when pulsed from 300 W. The staff agrees with the results of the GA study, which predicts higher final fuel temperatures for equal reactivity insertions when pulsing from the lower power level. Thus, initiating the reactivity transient with the reactor core at ambient water temperature and zero initial power will result in the maximum final fuel temperature and, therefore, will produce the maximum potential effect on the reactor and its associated components.

GA has performed many pulsing operations in both standard and FLIP core involving step reactivity insertions up to 3.5% $\Delta k/k$ (5.00\$) and peak temperatures up to 1,100° C (1,000° C for standard fuel) with no observed fuel damage. The maximum temperature would be reached by only a small fraction of the fuel in the hottest rod, so the average temperature and, hence, the hydrogen pressure would be well below the limit for rupture of the cladding. The moderating and reflecting water is necessary for the reactor to operate in the steady-state mode and to support a transient nuclear excursion. Therefore, the fuel would be totally immersed in water, the cladding would be cooled continuously, and its temperature would remain well below the highest 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 Celsius degrees within 2 min of the transient (GA-5400, 1970). Hence, if the ambient cooling 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.

However, GA has done few pulsing operations in mixed cores. In 1976, the applicant discovered four damaged FLIP fuel rods (none had ruptured cladding) in a mixed core that had been pulsed 54 times with 2.70$, and a total of ~ 80 times at levels between 2.00$ and 2.70$. The damaged rods were all positioned adjacent to the transient rod throughout their operating history. It is emphasized that the cladding did not rupture in any of the fuel rods.

After a lengthy study conducted by the applicant, GA, and Argonne National Laboratory (ANL) (GA-A16613, 1981), the fuel damage mechanism was determined to involve long-term, high-temperature, steady-state operation which caused redistribution of the hydrogen by migration. Large pulses after the steady-state operation resulted in sufficiently high pressures in the hydrogen-rich areas to cause swelling, porosity, and rapid hydrogen redistribution, leaving the region relatively depleted in hydrogen content. Temperatures above 530° C in hydrogen-depleted regions can cause additional volume changes of up to 15%. This mechanism is independent of the erbium content of the fuel and, therefore, is applicable to FLIP and standard TRIGA fuels.

The peak temperatures for a 2.70$ pulse in the four damaged fuel rods as calculated by the applicant are given in Table 14.1 with the observed condition of the rods.

Table 14.1 Calculated peak temperature for a 2.70$ pulse

Calculated peak temperature, 2.70$ pulse °C	Rod condition
908	Maximum damage
874	Damage
920	Damage
890	Small bump (passes go/no-go gauge)

Because temperature is the parameter that ultimately determines the damage-free operation of the U-ZrH_x the applicant has proposed to incorporate a fuel temperature limit into the Technical Specifications. The applicant proposed to take the lowest of the four temperatures above 874° C, as defining the onset of damage, and considers a reduction by a factor of 2 in equilibrium hydrogen pressure from that corresponding to this temperature to be a "reasonable and prudent" safety factor. This established a limit of 830° C for pulsing in any core that has had more than 8 MW-days of steady-state burnup. The 8 MW-days are based on the evaluation for the Washington State University (WSU) (NUREG-0911) reactor mixed core in which no fuel drainage was observed following that amount of burnup.

One difference between the WSU reactor and the Texas A&M reactor is that the former uses a water-followed transient rod and the latter uses an air-followed one. During a pulse in the WSU reactor, the water replacing that part of the transient rod out of the core causes neutron flux peaking, which produces increased heating of the adjacent fuel rods. However, the additional water volume may increase the cooling of the adjacent fuel rods in the WSU reactor compared with that of the adjacent fuel rods during a pulse in the Texas A&M reactor. These two effects in the WSU reactor core tend to cancel one another, but if the increased cooling dominates, this may contribute to the lack of damage observed in the WSU core.

Texas A&M University has proposed a new Technical Specification limiting equivalent reactivity insertions based on temperature observations associated with their damaged fuel experience. These limitations should prevent a recurrence in their particular reactor core. The proposed Technical Specification provides the following limiting conditions:

(1) The reactivity to be inserted for pulse operation shall not exceed that amount which will produce a peak fuel temperature of 830° C. In the pulse mode the pulse rod will be limited by mechanical means so that the reactivity insertion will not inadvertently exceed the maximum value.

(2) The maximum fuel temperature is monitored by the instrumented fuel element that is adjacent to the fuel element that will experience the greatest temperature rise. This proposed specification is intended for all cores (that is, full standard, mixed, and full FLIP). The 830° C limit on peak core temperature corresponds to a maximum allowable reactivity insertion of 2.27$ in a 35-FLIP-63-standard-rod core. This is 0.35$ less than the 2.70$ pulses identified for the above-mentioned damaged fuel rods.

14.1.2 Conclusions

The staff believes that the fuel damage mechanism proposed by the applicant is reasonable. Furthermore, the staff agrees that a safety factor of 2 in equilibrium hydrogen pressure is adequate, and that, based on a damage threshold of 874° C and a prior GA fuel testing program (GA-A16613, 1981), a maximum fuel temperature limit of 830° C for pulsing is justified.

The staff concludes that with the operating conditions imposed by the Technical Specifications and the inherent safety of the TRIGA fuel, there is no credible nuclear excursion possible with the Texas A&M 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 the reactor can be operated safely and there will be no significant health effects to the public.

14.2 Loss of Coolant

A potential accident that would result in increases in temperatures of the fuel and cladding is the loss of 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 in the fuel.

14.2.1 Loss of Coolant Accident Scenarios

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 (GA-5400, 1964; GA-6596, 1970). In the Texas A&M reactor, the core is completely immersed in water as long as the level of the water is at least 6½ ft above the tank bottom.

The 6½-ft level corresponds to about 22,600 gal of water in the tank. Therefore, about 85,400 gal could be removed before the core is uncovered. If it is assumed that a gross constant leak of 1,500 gpm occurs, the core would remain covered for at least 57 min. If convective water cooling continued that long, peak fuel temperature less than 959° C would be reached in 3 hours, assuming that the core had been operating long enough at 1 MW to achieve fission product equilibrium (to be conservative). Not only would this maximum temperature not lead to rupture of the fuel cladding, but the time scale for the entire event would allow for remedial action. In addition, pool water level and pool temperature would alarm in the control room to permit orderly shutdown of the reactor.

Section 14.1 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 Texas A&M 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 min of the time that the core becomes uncovered. For the above reasons, it is not considered possible that the reactor would continue to be operated to reach the above conditions.

14.2.2 Conclusions

If the reactor were pulsed shortly after an extended run, the heating resulting from the additional inventory of fission products would be negligible. Furthermore, as indicated in Section 14.1, the fuel temperatures necessarily must be reduced to that of 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 loss of coolant following extended steady-state operation would not result in a temperature rise of the hottest fuel element that would cause fuel or cladding melting.

14.3 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, and there is an extensive body of literature on the subject (Baker and Just, 1962; Baker and Liimatakinen, 1973; Battrey et al. 1965). 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 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 14.1 and 14.2, these events were shown not to be credible in a 1-MW U-ZrH_x fueled reactor like the one authorized for operation at Texas A&M.

Additionally, some of the studies (Baker and Liimatakinen, 1973) 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 Sections 14.1 and 14.2 show that such an event is not credible.

In addition to the investigations referenced above, GA has experimentally plunged heated samples of unclad ZrH_x into water to examine possible conditions for initiating and sustaining a metal-water reaction (Lindgran and Simnad, 1979). Up to temperatures of about 1,200° C, there was no chemical reaction of the metal except for the formation of a relatively inert oxide film. Furthermore, most of the hydrogen may have been driven off in the hottest unclad test samples, so the metal surface in contact with the water could have been mostly zirconium.

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 TRIGA-type reactor that is operating at 1 MW or below with maximum available excess reactivity as authorized at the Texas A&M reactor.

14.4 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 that could cause a failure of the experiment container. As discussed in Section 10, all new experiments at the Texas A&M reactor are reviewed before 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 tubes, beam ports, or through tube.

The staff concludes that the experimental facilities and the procedures for experiment review at Texas A&M 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 14.7.

14.5 Mechanical Rearrangement of the Fuel

This type of potential accident would involve the failure of some reactor system, such as the support structure, or could involve an externally originated event that disperses the fuel and, in so doing, breaches the cladding of one or more fuel elements.

There is no logical basis for deciding if any arbitrary scenario is credible. Instead, Section 14.7 discusses a scenario assuming the failure of the cladding of an element after extended reactor operation and evaluates possible doses resulting from various hypothetical scenarios for release of the inventory of radioactivity.

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 discussed in Section 14.1.1.

The staff concludes that no mechanical rearrangement that is credible would lead to an accident with more severe consequences than those accidents considered in Sections 14.1 or 14.7.

14.6 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 Section 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 also are addressed in Section 17. This section addresses the possibility of internal reactions.

14.6.1 Fuel Aging Scenario

There is some evidence that the U-ZrH_x fuel tends to fragment with use, probably because of the stresses caused by high temperature gradients and the high rate of heating during pulsing (GA-A16613, 1981; GA-4314, 1980). Some of the possible consequences 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 significant 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 the principal gap between cladding 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 onto the opposite surface and then have to diffuse back out to be released into the gaps.

14.6.2 Conclusion

The staff concludes that the two likely processses of aging of the U-ZrH_x 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 increase the likelihood of fuel-cladding failure or significantly increase the quantity of gaseous fission products available for release in the event of loss of cladding integrity.

However, as pointed out in Section 4.6, the decrease in magnitude of the prompt negative temperature of reactivity (caused by erbium burnup) must be considered in operations with FLIP fuel that has reached significant burnup. This consideration is accounted for in the maximum fuel temperature limitation of 830° C.

14.7 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.

14.7.1 Fuel-Handling Scenario

The fuel-handling accident scenario includes 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, the staff did not try to develop a detailed scenario, it 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 GA have obtained data on the species and fractions of fission products released from U-ZrH_x under various conditions (Baldwin, Foushee, and Greenwood, 1980; Foushee and Peters, 1971; Simnad, Foushee, and West, 1976). 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.

GA has proposed a theory describing the release mechanisms in the two temperature regimes that appears to be valid, 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, be independent of operating history or details of operating temperatures, and 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 that 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 applicant selected a release fraction from the GA results that corresponds to a fuel temperature of approximately 400° C. Because the GA 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 Texas A&M reactor. The release fraction the applicant selected is 2.6×10^{-5} of the inventory of both the noble gases and the iodines (Foushee and Peters, 1971). During steady-state operation at 1 MW, the Texas A&M reactor's measured fuel temperature does not exceed 525° C, and because the thermocouples are near the axial center of the hottest fuel rods, they measure the region of maximum temperature, which is well above the core average.

Because the noble gases do not condense or combine chemically, it is correct to assume that any release 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 that are applicable to nonpower reactors (Simnad and Dee, 1968; Simnad, 1980; Kessler, 1966). However, to be certain that the fuel-cladding-failure scenarios discussed below arrive at upper limit dose estimates for all events, the applicant assumed that 100% of the iodines in the gap become airborne if water is absent from the pool. This assumption will lead to computed doses that may be at least a factor of 100 too high in some scenarios, e.g., those in which the pool water is present. The staff has reviewed the various acceptable methods for computing the expected dose beyond the confines of the 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 that 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 fuel rod, these methods generally give results that are extremely conservative because outside the reactor room the calculative method includes the assumption that the individual is surrounded by a semi-infinite cloud. This conservative approach was used in the staff calculation for the following fission product pathway and exposure scenarios.

(1) In a single fuel-rod-cladding failure in air immediately after an extended 1-MW run, the applicant assumes that the reactor building exhaust dampers close and that all of the noble gas 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 4 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) The same event occurred as in scenario (1), 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 because of 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 50 mrems and the dose commitment to the thyroid from breathing the iodines in the air would be less than 30 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 scenario (2), but the analysis considers the potential exposures to personnel beyond the control of the Texas A&M authorities.

    The land adjacent to all sides of the reactor complex is owned and controlled by Texas A&M University. To add to the conservatism, the applicant computed and the staff verified the potential dose to a person at 100 m (328 ft), assuming that 100% of the iodines and noble gases released from the fuel-cladding escape from the building and are carried by a 3-ft-per-sec wind, with Pasquil Type F atmospheric conditions. This wind speed and stability condition are very infrequent at Texas A&M, but these assumptions lead to a "worst case" analysis. The staff used the method of the applicant (Safety Analysis Report 1982). Thus, at 328 ft from the reactor building, the staff verified a whole-body exposure of 0.1 mrem and a total thyroid dose of approximately 18 mrems. The allowable annual doses in 10 CFR 20 are 5 rems and 30 rems respectively. As these computations are based on conservative assumptions the results are higher than would realistically occur.

14.7.2 Conclusions

In accordance with the discussions and analyses above, the staff concludes that if one fuel rod from the Texas A&M reactor were to release all noble gases 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 a small fraction of the limits stipulated in 10 CFR 20. These assumptions correspond to a very conservative scenario.

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-cladding failure at the reactor, there would be no significant risk to the health and safety of the public.

14.8 Conclusion

Based on the above review and analyses, it has been shown that a single fuel rod cladding failure will produce exposures that are small fractions of 10 CFR 20 and that even if several fuel rods failed at once, the expected dose equivalents in unrestricted areas would still be well below 10 CFR 20 limits. Therefore, the staff concludes that the design of the facility and the Technical Specifications provide reasonable assurance that the Texas A&M reactor can be operated with no significant risk to the public's health and safety.

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 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 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 Texas A&M reactor is owned and operated by a state university in support of its role in education and research. Therefore, the staff concludes that funds will be made available, as necessary, to support continued operations and eventually to shut down the facility and maintain it in a condition that would constitute no risk to the public. The 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 both normal operation and an accident to the reactor causes insignificant risk of radiation exposure to the public and that even a maximum hypothetical accident would only result in a dose to the most exposed individual that is a small fraction of applicable guidelines or regulations (10 CFR 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.

17.2 TRIGA Fuel Damage Incident at Texas A&M

17.2.1 Background

During a loading operation on September 27, 1976, three* demaged fuel elements were discovered in the core. Fuel element deformation but no cladding failure were noted (see Figure 17.1). A report on the discovery and preliminary analysis was submitted from the applicant to the staff on November 1, 1976. At that time it was determined that no valid conclusion concerning the cause of the damage was possible without metallographic examination of the damaged fuel elements. Argonne National Laboratory (ANL-West) had TRIGA-FLIP fuel and the necessary analytical facilities to perform these examinations. Because of delays as a result of hot cell modifications, the low priority of this project, and coordination of personnel from Texas A&M, Argonne National Laboratory, and General Atomics Company, it was several years before data were collected and the evaluation of the work completed. The final report was issued in December 1981 (GA-A16613) with the proposed damage mechanism receiving consensus from those concerned.

The damaged fuel elements were FLIP fuel, fabricated with a hydrogen-to-zirconium ratio of 1.6. This hydrogen-to-zirconium value was carefully selected to take advantage of several properties of zirconium hydride. Figure 17.2 reproduces a phase diagram of ZrH_x presented by Simnad (GA-4314, 1980). The diagram indicates that a hydrogen-to-zirconium atom ratio of 1.6 produces the delta phase and remains in this phase to well over 1,000°C. This avoids substantial volume changes resulting from phase transformations that occur at approximately 530°C at lower hydrogen ratios (GA-9064, 1970 and GA-6874, 1966). It also produces a material that exhibits very small density changes with varying hydrogen content (GA-4314, 1980). An additional advantage of the delta phase is that it has much greater creep strength than the beta phase. However, if the fuel is used

*In addition to the three elements, one fuel element that was slightly warped passed the deformation test and was, therefore, determined to be undamaged by definition.

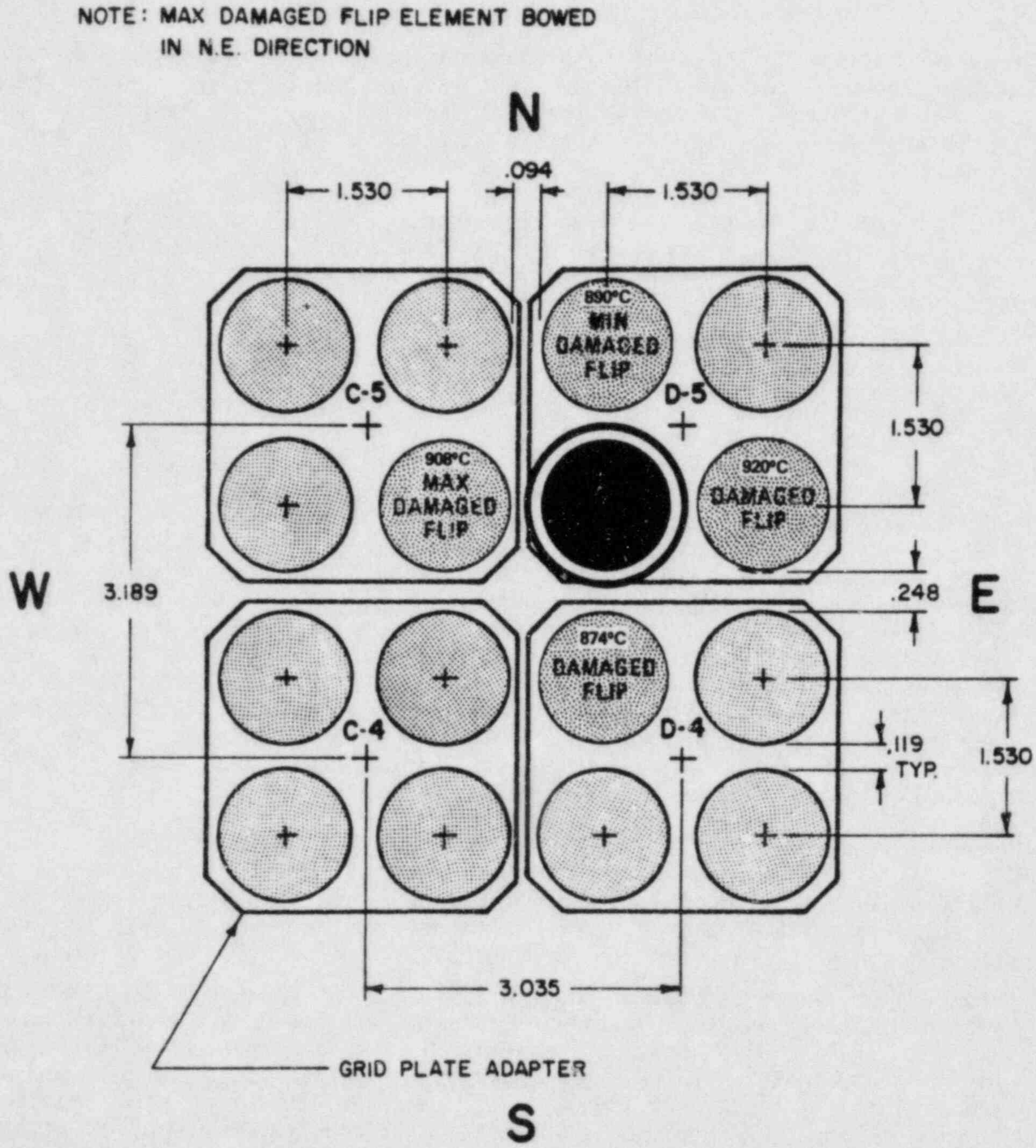


Figure 17.1 FLIP fuel element spacing in the Texas A&M reactor core near transient rod

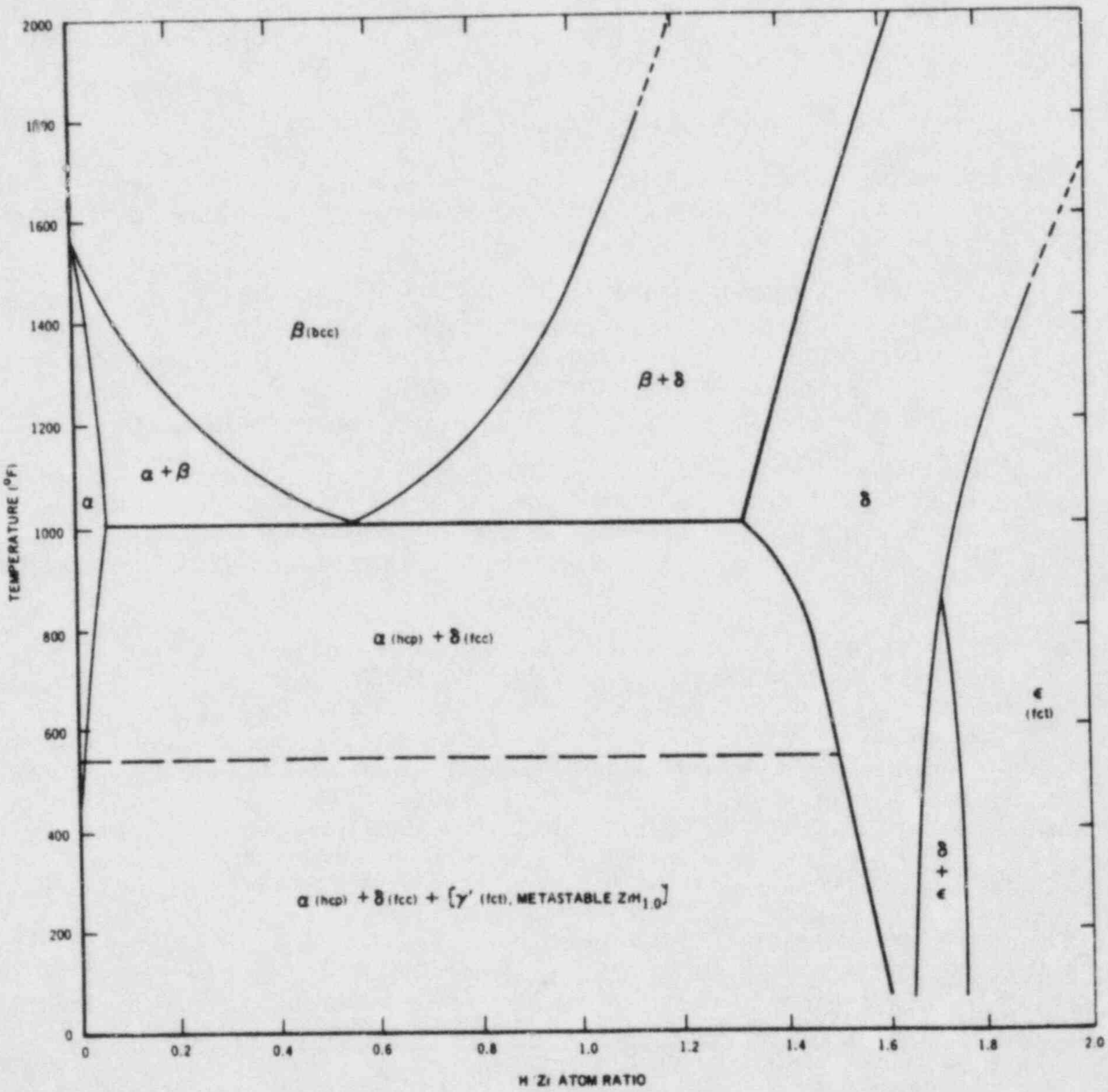


Figure 17.2 Zirconium hydride phase diagram showing boundary determination

in a 1-MW core with a steady-state utilization factor, the ideal properties and characteristics mentioned above will not be totally retained. When subjected to temperature gradients, hydrogen will migrate from the hotter fuel regions to the cooler regions with the migration rate being greatly influenced by time, temperature, and the temperature gradient. Figure 17.3 shows that when operating at long term 1-MW steady state, there can be 300C° temperature difference across only 1 cm of fuel. Thus, during this time, there will be a long-term loss of hydrogen from the hotter regions to the cooler outer regions. Because there is negligible migration below 250°C (Texas A&M SAR Amendment, April 16, 1982), the outer skin of the element will retain the original hydrogen-to-zirconium ratio of 1.6.

17.2.2 Results of Investigative Program

After an extensive review of other similar TRIGA operations and metalographic and neutrographic examinations of the damaged fuel, the following mechanisms were postulated by the applicant (GA-A16613):

(1) During continuous operation for long periods, their is hydrogen migration radially and axially from regions of higher temperatures to regions of lower temperatures. This produces local areas of high concentrations of hydrogen, which could be sources of high pressure at higher temperatures experienced mostly during long-term operations (see Figure 17.1).

These higher concentrations are limited to regions that are some distance below the surface. The higher concentrations in the subsurface regions would lead to higher internal gas pressures at a hot spot during pulse operations that those that would occur with a nominal hydride composition of ZrH 1.6.

(2) During steady-state operation, essentially no hydrogen migrates to the immediate surface region of the fuel or to the central zirconium rod (which is in the hottest part of the element) because of the temperature gradient. The high central temperature forces hydrogen away from the center, and very low migration rates at the immediate fuel surface region temperature prevent hydrogen buildup in this zone (see Figure 17.4). During high power pulsing, the higher temperatures produced generate higher pressures in those regions of high hydrogen concentrations; this results in swelling of the fuel and increased pore size.

(3) The "hydrogen-depleted" regions result from the loss of hydrogen to the cooler parts of the fuel. This hydrogen evolves from the hot spots during high-power pulsing and appears to have been absorbed by the cooler regions (especially the central zirconium rod) thereby depleting the hot spots of hydrogen. Under high-power steady-state operation, the hydrogen in the depleted hot spots would be replenished (but at a much slower rate) by diffusion in the solid state and by migration in the gas phase from neighboring regions.

(4) The central axial zirconium rod in the center of the fuel element appears to be a source of stress generation under the conditions encountered in the hottest parts of the fuel. These rods can swell up to 15% in volume upon absorption of hydrogen to give an hydrogen-to-zirconium ratio of 1.7. Under extreme swelling conditions, the initial clearance between the

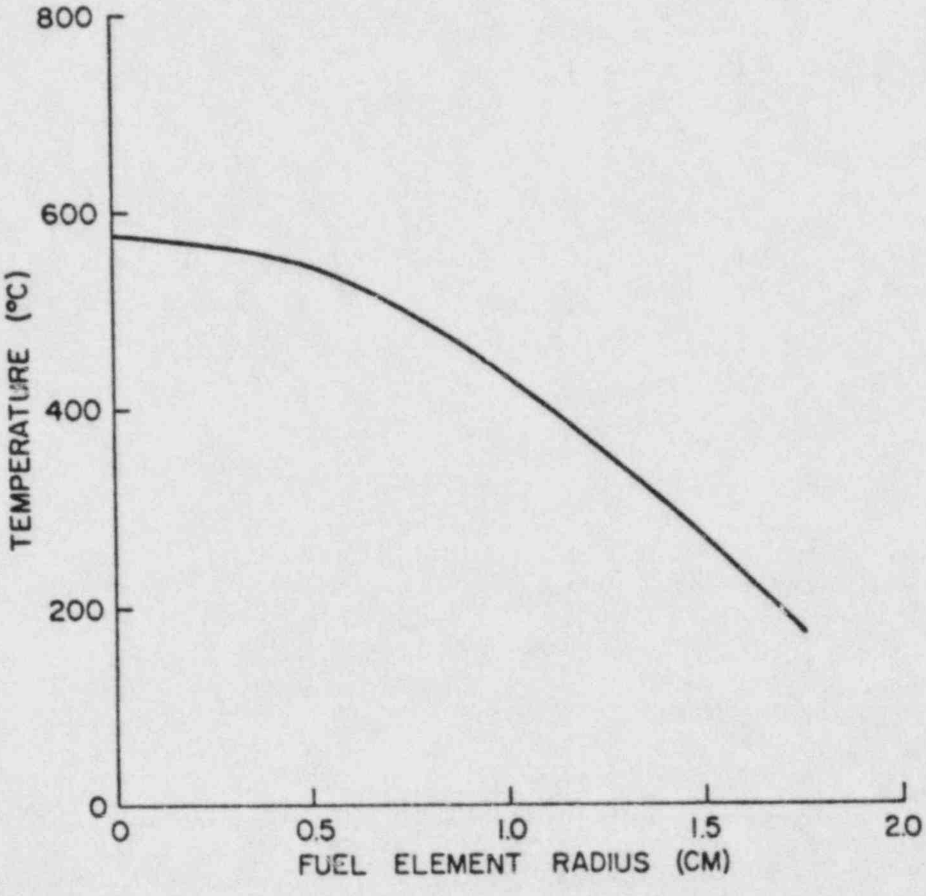


Figure 17.3 Radial temperature distribution in maximum power element at 1 MW, Core IIIA

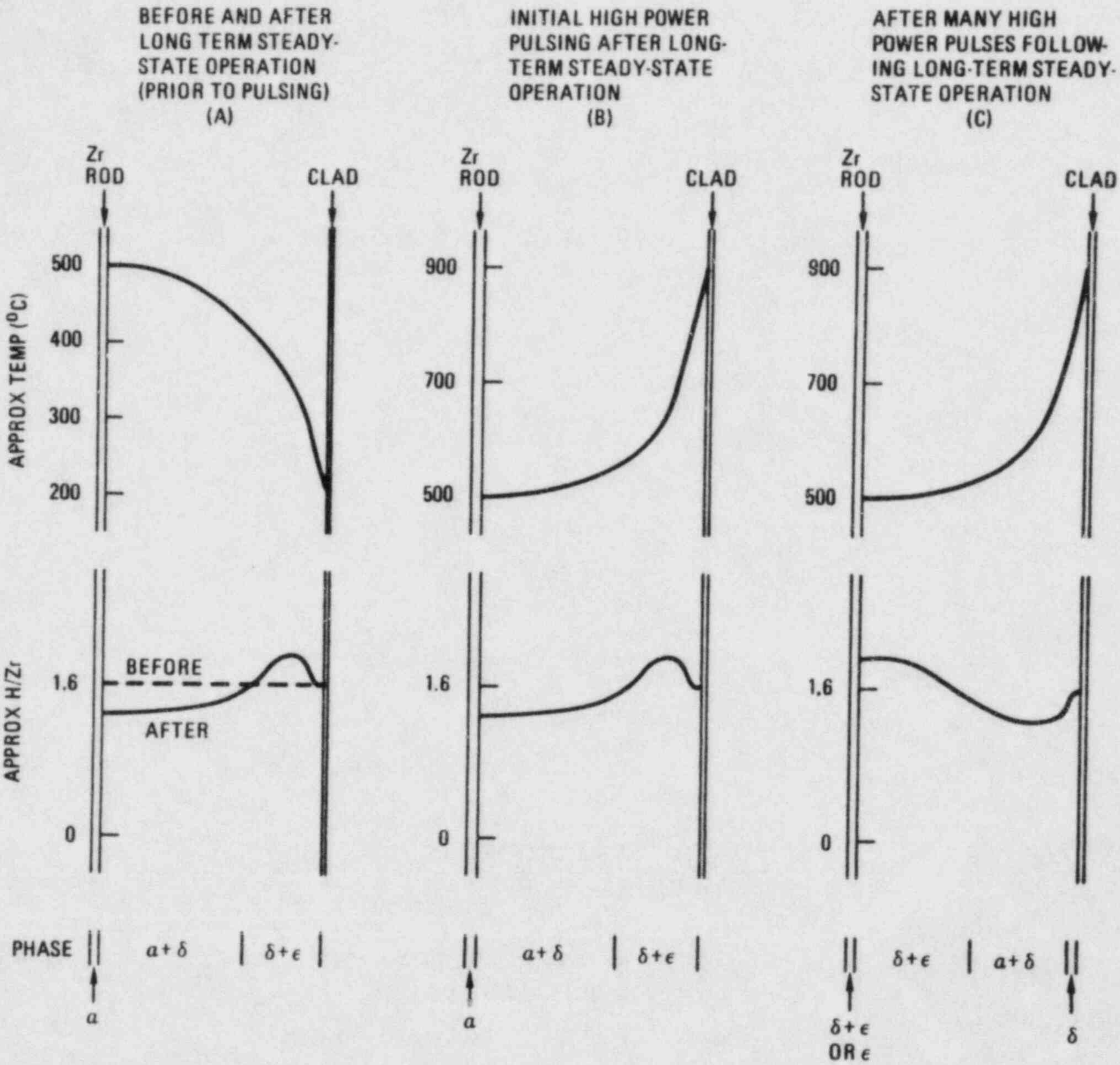


Figure 17.4 Qualitative graphic description of possible changes in hydrogen distribution leading to fuel damage

zirconium rod and the fuel appears to be too small and the zirconium rod will swell against the fuel. It appears as if stresses, generated by the swelling zirconium rod, were large enough to crack the fuel. The neutron radiographs and metallography of the most damaged highest-temperature portions of the fuel element indicate complete hydriding of the central zirconium rods, whereas the low temperature, undamaged fuel shows little hydriding in the zirconium rod. In some cases the expanded zirconium rod actually bonded to the fuel at points of contact.

(5) The structure of the hydrogen-depleted region in the distressed fuel apparently contains significant quantities of alpha-phase material formed by loss of hydrogen from the original delta phase. The fine pores in this region are largely in the form of a maze of pores at the grain boundaries. As the orignal delta phase transforms to the alpha phase, the pores that are present in the alloy are apparently swept to the newly formed grain boundaries of the alpha phase. Also, the change in density upon transformation from delta to the denser alpha phase will favor the formation of voids that will be trapped at the new grain boundaries.

7.2.3 Evaluation

The staff agrees with General Atomics' and Texas A&M's conclusions (GA-A16613) that the proposed mechanism deemed responsible for damaging the fuel is long-term, high-temperature, steady-state operation that causes redistribution of the hydrogen by migration. This, followed by large pulses (without substantial down time), results in sufficiently high pressures in the hydrogen-rich areas to cause swelling, porosity, and rapid hydrogen redistributing, leaving the region below average in hydrogen content. The central zirconium rod, used only in FLIP fuel, could also have been a significant cause for the physical phenomena observed. Temperatures above 530°C in the hydrogen-depleted region can cause additional volume changes of up to 15%. It also is possible that through continued operation, hydrogen would be replenished by redistribution and the process repeated.

The applicant feels that the damage threshold was only slightly exceeded in the Texas A&M core. He suggests that this is evident in the physical examinations of the fuel and the calculated spread in the power generation among the four damaged fuel elements, which was about 5%. Examination showed that the damage ranged from very slight to significant. This is further reinforced by the fact that the maximum temperature in the Texas A&M core was calculated to be only 20C° greater than that in the FLIP core at General Atomics, which had longer steady-state operation and higher pulses but no fuel damage. Because the hydrogen pressure increases nearly exponentially with fuel temperature, small temperature changes can make a very significant difference in hydrogen pressure and in fuel damage.

The lower General Atomics fuel temperature of only 20C° less resulted in no apparent fuel damage. In fact, this relationship makes the fuel damage mechanism act, for practical purposes, as if there were a threshold temperature for damage. If damage begins to appear, a reduction in measured temperature as small as 20C° (40C° in peak pulsing temperature) will reduce hydrogen pressures by about 33% and is likely to stop any progressive damage.

The remedies recommended by Texas A&M include

(1) Annual Fuel Inspections

The Technical Specifications, Section 4.2.4, indicate annual visual inspection for damage, deterioration, dimensional changes, and bending for the four fuel elements that occupy the four positions in the core with the highest power density. If any of these four elements exhibit any damage as defined in (a), (b), and (c) of this specification, they must be replaced.

(2) Limiting Fuel Temperatures

Texas A&M has an instrumented fuel element adjacent to the position expected to yield the highest temperatures. The highest temperature to be allowed in any fuel element is 830°C. Texas A&M has calibrated the thermocoupled fuel element so that relationship is known between that fuel element position and the position of the expected hottest fuel element. The temperature limit of 830°C is approximately 60C° below the lowest temperature that exhibited damage to any fuel element. As the internal pressure is materially affected at higher temperatures (approaching asymptotic increases) the 60C° increment appears to be a conservative value.

It should be emphasized that had a cladding rupture of a fuel element occurred it still would not result in any significant hazard to the contiguous public. Section 14.7.1 of this report reviews the effects of a postulated cladding failure associated with a fuel-handling accident. The assumptions also included the conservative suppositions that all of the fission products accumulated in the gap were abruptly released from one maximally irradiated fuel rod. The calculated exposures were as follows:

(1) 4 mrems per hour whole-body dose to a person remaining in the reactor room

(2) 50 mrems whole-body dose and 30 mrems thyroid dose to a person outside the building for the entire time interval of the accident scenario, which considered a constant leakage rate over at least 24 hours

(3) 0.1 mrem whole-body dose and 18 mrems thyroid dose for constant leakage rate as in (2), but for exposures at the Texas A&M property line

All exposures for the conservative assumptions are small fractions of the standards prescribed in 10 CFR 20.

17.2.4 Conclusions

Though all the answers to all the questions may not be provided in the General Atomics report (GA-A16613) on the Texas A&M fuel damage incident, the staff agrees with the evaluation of the principal reasons for the incident and with the remedies of an annual inspection and relatively low maximum fuel temperature to preclude similar future incidents.

17.3 Effect of Prior Operation on Potential for Accidents

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. Futhermore, because loss of integrity of fuel cladding is the design-basis accident, the staff must consider mechanisms that could increase the likelihood of failure. Possible mechanisms are (1) radiation degradation of cladding strength, (2) high internal pressure caused by 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 as a result of 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 are presented above are as follows:

(1) Annual reports indicate that reactor utilizaton has been between 25% and 35% since 1971. The hypothetical accident scenarios considered in Section 14 of this report assume 1 year constant operation with a commensurate buildup and release of fission products from a postulated cladding failure. The consequences of these hypothetical accidents are exposures that are a fraction of those in 10 CFR 20.

(2) There is the possibility of approaching such pressures, if the long-term operating conditions and pulse reactivity insertion limits were greater than those which produced the damaged fuel described in Section 17.2. However, as operation is limited by the maximum fuel temperature limit of 830°C and as the damaged fuel incident indicated that the cladding still maintained its integrity, it is highly unlikely that future fuel cladding damage will occur or that any fuel still in the core that experienced the prior conditions that led to the fuel damage incident will be a candidate for fuel cladding failure if the core is operated within the specified limits.

(3) Water flow through the core is obtained by natural thermal convection, so the staff concludes that erosion effects as a result of 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 mmho-cm^{-1} 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 the annual inspection requirements of the Technical Specifications. Any indications of possible damage or degradation are investigated immediately. The only experiments that 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 experimental tube. In addition, Section 14 of this report indicates that a fuel-handling accident will generate exposure doses that are a small fraction of those in 10 CFR 20. Therefore, the staff concludes that the possibility of loss of integrity of cladding through damage does not constitute a significant risk to the public.

(5) Texas A&M 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 apparaent significant degradation of safety equipment, and because there is strong evidence that any future degradation will lead to prompt remedial action, there is reasonable assurance that there will be no significant increase in the likelihood of an 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. The inventory of radioactive fission products in the Texas A&M reactor will be far below that postulated in the evaluation of the design-based accident both by the applicant and the NRC staff (see Section 14). Therefore, the staff concludes (1) that the risk of radiation exposure to the public from any postulated accident will be 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.

17.4 Multiple or Sequential Failures of Safety Components

Of the many accident scenarios hypothesized for the Texas A&M TRIGA reactor, none produce consequences more severe than the accidents reviewed and evaluated in Section 14. The only multiple-mode failure of more severe consequences could 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 cladding 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.

The staff concludes that the Texas A&M reactor can be operated for the duration of this license period without any significant hazard to the public, because the postulated hypothetical multiple-failure scenario produces consequences less than the maximum hypothetical accident.

18 CONCLUSIONS

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

(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 small fractions of 10 CFR 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 10 CFR 20 and are 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 staff's review of the applicant's report (GA-A16613) on the 1976 damaged fuel incident generally concurs with the applicant's analysis and evaluation concerning the reasons for and remedies to prevent future fuel damage.

The staff recognizes, however, that the January 1983 discovery of two additional damaged fuel elements indicates that some factors may have had a greater impact on the fuel damage mechanisms than those proposed in the report. The inspection requirements and the operating limitations included in the Technical Specifications are intended to both verify the conclusions reached in the damaged fuel incident report and preclude the events from reoccurring (see Section 17).

(7) 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 to ensure protection of the public from radiation exposures when operations are terminated.

(8) 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 73.

(9) 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.

(10) The Texas A&M Emergency Plan, though submitted with the license renewal application is incomplete at the time of publication of this safety evaluation because of a recent requirement change by NRC. This item, discussed in Section 13.3, will be submitted by November 3, 1982 as part of the current NRC requirements, published in the Federal Register in May 1982.

(11) The application for renewal of Operating License R-83 for its research reactor filed by the Texas A&M University, dated July 2, 1979 as amended, complies with the requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter 1.

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

(13) 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.

(14) 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.

(15) 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 Texas A&M University for a renewal of operating license R-83 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 and operated by the Texas A&M University and is located 3 miles from the Texas A&M campus in College Station, Texas, Brazos County, Texas. The staff concludes that the TRIGA reactor facility can continue to be operated by Texas A&M without endangering the health and safety of the public.

17. KEY WORDS AND DOCUMENT ANALYSIS	17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

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