AEROTEST OPERATIONS INC
AEROTEST RADIOGRAPHY AND RESEARCH REACTOR
LICENSE NO. R-98
DOCKET NO. 50-228

UPDATED SAFETY ANALYSIS REPORT
(DATED February 28, 2005)

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

*REDACTED TEXT AND FIGURES BLACKED OUT OR DENOTED BY BRACKETS

AEROTEST OPERATIONS, INC.

Aerotest Radiography and Research Reactor (ARRR)

UPDATED SAFETY ANALYSIS REPORT (USAR)

REVISION 0 [PROPOSED R0]

DOCKET No. 50-228
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Aerotest Operations, Inc.
3455 Fostoria Way
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Table of Contents

1.0 THE FACILITY

1.1 INTRODUCTION

This Updated Safety Analysis Report (USAR) supports an application by Aerotest Operations, Inc. to the Nuclear Regulatory Commission (NRC) for the utilization of a steady state 250 kW TRIGA-type reactor. The reactor is known as the Aerotest Radiography and Research Reactor (ARRR). It is owned by OEA, Inc., a wholly owned subsidiary of Autoliv ASP, Inc. and is operated by Aerotest Operations personnel. This document addresses the safety issues associated with the ARRR and provides an evaluation which demonstrates that its operation does not result in undue risk to the health and safety of the public.

1.1.1 Purpose of Facility

The ARRR provides a neutron source for research and development and services, mainly neutron radiology. Neutron radiology is used for non-destructive testing and failure analysis for the aerospace, military, and industrial communities. Its uses include the following:

(1) Research and development for new neutron radiography equipment and image quality indicators including conversion screens, neutron detectors, and beam definition devices.

(2) The presence, absence, or correct placement of explosives, adhesives, o-rings, plastic components, and similar materials are verified.

(3) Ceramic residual core in investment cast turbine blades are detected prior to assembly in aircraft engines.

(4) Welds are verified for integrity in propellant tanks prior to space flight.

(5) Sustained-release drug delivery systems are inspected prior to being prescribed to patients with cancer and various neurological and skeletal disorders for chronic pain therapy.

(6) The ARRR also provides services for activation analysis, irradiations, and radiation damage studies.

1.1.2 Location of Facility

The facility is located in the town of Danville, California, which is approximately 10 miles east of Oakland in the San Francisco Bay Area. The site is surrounded by Pacific Gas and Electric property on three sides. The fourth side is on one side of an extension of Fostoria Way. Industrial buildings are located on the opposite side of the street.

1.2 PRINCIPAL SAFETY CONSIDERATIONS

The specific site was originally selected to be above any potential flooding level and where no earth fractures were known to exist. Although the ARRR was originally in an area of sparse population, this is no longer applicable. However, over 40 TRIGA reactors are in operation world-wide, half of which are in the United States. Many of these reactors are located on university campuses and in hospitals with surrounding highly populated areas.

The ARRR uses TRIGA fuel-moderator elements developed by General Atomics which consist of a solid homogenous alloy of uranium fuel and zirconium-hydride moderator (U-ZrH). These unique fuel elements provide the TRIGA reactor with a large prompt negative temperature coefficient. This means that any increase in power heats both the fuel and the moderating material in the fuel simultaneously, and immediately the fuel becomes less effective (decreases in reactivity). This causes the reactor to return automatically to normal operating levels within milliseconds. This characteristic is intrinsic to the TRIGA reactor core, and permits safe, steady state operation and pulsing to high power levels without the use of mechanical or electronic control devices to maintain the reactor at safe power levels throughout and after the transient. The ARRR design and Technical Specifications do not allow the ARRR to be pulsed. However, the instantaneous introduction of a sufficient amount of excess reactivity to cause a large increase (100 fold) increase in reactor power will not result in any damage to the fuel elements.

The maximum hypothetical accident analyzed for the ARRR is a stepwise insertion of all available excess reactivity coincident with a defect in the cladding of a fuel element. The maximum hypothetical accident analyzed for a TRIGA fueled reactor is a fission product release directly to the atmosphere following a fuel handling accident that causes clad rupture and severely damaged fuel. As described in USAR Chapter 13, Accident Analyses, an analysis has shown that no significant hazard to the public will result from either accident. The other abnormal operations considered in the analyses described in USAR Chapter 13 include:

(1) Insertion of excess reactivity from any of the following: startup rod withdrawal, uncontrolled rod withdrawal, step insertion of all excess reactivity, fuel loading accident, and experiment removal accident.

(2) Loss of Coolant and Loss of Coolant Flow.

(3) Experiment Malfunction.

(4) Loss of Normal Electrical Power.

(5) External Events including earthquake or damage to the building by other causes.

(6) Mishandling or Malfunction of Equipment including detonation during storage or radiography of explosive devices at the ARRR facility.

1.3 GENERAL DESCRIPTION OF THE FACILITY

1.3.1 Building

The reactor building is made of steel with internal rooms built of fire resistant framing and sheetrock covering. An automatic sprinkler system covers the entire building. The building has airflow control from the standpoint that certain rooms

are maintained at a positive pressure relative to the reactor room. The building as a whole is not sealed or contained and requires no air-locks.
The reactor control room and certain offices are housed in a single building. The control room and offices are in areas where a fresh air intake is used to maintain the positive pressure relative to the reactor room. The 40 feet x 80 feet main building has a [redacted] capacity bridge crane that can cover the entire area. The reactor tank is embedded in the floor, extending [redacted] below and [redacted] above the floor surface. A [redacted] thick by [redacted] high block wall made of normal density concrete encloses the reactor area above the floor level. The [redacted] of this shield is covered with an [redacted] thick wooden shield.

A [redacted] perimeter fence surrounds the facility to form the exclusion area. Details of the building are shown in Figure 1.1.

1.3.2 Reactor

The ARRR was designed and constructed by the Nuclear Division of Aerojet General in 1964. The reactor fuel elements, reflector elements, control rods, control rod drive mechanisms, and control rod drive controls were purchased from General Atomics and were incorporated without any significant changes. A standard "G" ring core grid plate design was provided by General Atomics and manufactured by Aerojet. All other components were designed and constructed by Aerojet or their subcontractors.

The basic nuclear design and core geometry follow General Atomics TRIGA reactor design characteristics. The original core was comprised of all aluminum clad fuel elements; however, new fuel elements are stainless steel clad. The original fuel elements are enriched to 8 weight % U-235 and the stainless steel clad fuel elements are 12 weight % U-235. As of June 2004, twenty of the stainless steel clad elements have been added.

The principal characteristics of the ARRR are as follow:

(1) Fuel: < 20% enriched U-235.

(2) Moderator: zirconium hydride and water.

(3) Reflector: demineralized water and graphite.

(4) Coolant: demineralized water.

(5) Control: 1 safety rod, 1 shim rod, 1 regulating rod, all boron carbide.

(6) Structural material: aluminum and stainless steel.

(7) Shield (principal materials): demineralized water, concrete, lead, and wood.

(8) Active core dimensions: [redacted] diameter (nominal) by [redacted] high for aluminum clad elements and [redacted] high for stainless steel elements.

It should be noted that the reactivity worth of all rods is dependent on the core configuration. Under no circumstances will the reactor be operated when the safety rod is worth less than 2.19% Δk/k. The total worth of all rods is about 6%Δk/k. Thus, if the shim rod should stick, the safety rod is fully capable of shutting down the reactor.

1.3.3 Reactor Coolant System

The ARRR cooling system is comprised of three basic parts: the reactor water tank, the cooling system, and the demineralizer system. The pool water provides convection cooling, neutron and gamma shielding and neutron moderation. The cooling system provides heat removal to a cooling tower via a heat exchanger. The demineralizer system maintains the purity of the water. A fourth component is the reactor water tank makeup system. This system is comprised of a small ion exchange column and redundant level control valves for the reactor water tank.

1.3.4 Instrumentation and Control

The ARRR has four neutron measuring control channels. These channels provide neutron flux information over the entire range of reactor power.

The reactor console uses a log count ratemeter for its Channel 1. The detector is a ^{10}B lined proportional counter connected to a preamplifier whose output drives a ratemeter with a logarithmic element in the output amplifier.

Channels 2, 3, and 4 all use current measuring instruments with ion chambers as the current averaging or integrating element. The Channel 2 instrument also uses a logarithmic diode at its input which results in a logarithmic output. It is capable of responding to a current of 10^{-12} amperes to 10^{-4} amperes with no switching. Channels 3 and 4 use current measuring instruments that require switching as the power level changes.

Three boron carbide control rods operate in perforated aluminum guide tubes. Each control rod drive assembly consists of a motor and reduction gear driving a rack and pinion. In the event of a scram signal or power failure, the control rod magnets are de-energized, decoupling the control rods from the drive assembly and the control rods fall into the core.

1.3.5 Radioactive Waste and Radiation Protection

The ARRR generates very little radioactive waste. Most of the induced radioactivity is short half life material and can be stored until the radioactivity decays.

The basic policy at Aerotest Operations, Inc. is to limit personnel exposures as low as reasonably achievable (ALARA). Under no circumstances will operations be permitted which would result in personal radiation exposures in excess of the radiation protection standards. In addition, any exposure or environmental measurement in excess of the control guides will be investigated and recorded.

1.4 COMPARISON WITH SIMILAR FACILITIES

The design of the ARRR fuel is similar to those of approximately 40 TRIGA fueled reactors currently operating world-wide with half in the United States. Twenty-four of these reactors are of the same approximate age as the ARRR. Since a large number of these reactors have been in operation for many years, considerable operational information is available and their characteristics are well documented.

There are 6 other TRIGA reactors approved for steady state operation at 250 kW and an additional 7 that can be pulsed. Of the former, 3 are in the United States and are located at Argonne National Laboratory - West (constructed in 1977), University of Maryland (1974), and Reed College (1968). There are 23 TRIGA reactors approved for steady state operation between 300 and 14,000 kW.

The ARRR reactor fuel, instrumentation, and control systems are of proven design, based on past operating experience of systems with the same or similar designs. The reactor fuel, control rod drives, control rods, and experimental systems are similar to many other systems used throughout the United States. These items have well-established operating experience. The potential for and consequences of an accident at the ARRR are no greater than those of other similar reactors using the same fuel systems.

Table 1.6-1
Comparisons of Typical Principal Design Parameters

Parameter	ARRR (R-98)	Argonne	Maryland (R-70)	Reed (R-112)
Maximum power (kW)	250	250	250	250
Fuel elements	[redacted]	[redacted]	[redacted]	[redacted]
Control rods	3	3	3	3
Total reactivity (Δk/k)	0.03	0.045	0.0827	0.03
U-235 enrichment	20 %	70%	20%	20%
Pool temperature (°F)	70	89	80	68
Reactor Water Tank (feet)	[redacted]	[redacted]	[redacted]	[redacted]
Rod speed (inches/minute)	[redacted]	[redacted]	[redacted]	[redacted]

* 12 inches per minute for the safety and shim rod and 20 inches per minute for the regulating rod.

1.5 SUMMARY OF OPERATIONS

The ARRR provides a neutron source for research and development and services, mainly neutron radiology. Irradiation services for activation analyses have included: crude oil and hydrocarbon samples for oil companies; plastic slides impregnated with microscopic quantities of fissionable materials; ocean silt samples for the Bureau of Mines; and, silver iodide in snow samples from cloud seeding. Other irradiation services have included: calibration of power reactor fission detectors; radiation damage effects studies of solid state electronic components; detection of gunshot residue in paraffin; lattice deformation studies in ammonium perchlorate; and, spallation experiments with uranium dioxide.

Currently, the ARRR is operated for an eight hour shift five days a week, plus four hours every Saturday. Figure 1-2 shows the annual operating hours for the ARRR from 1966 to 2004.

1.6 COMPLIANCE WITH THE NUCLEAR WASTE POLICY ACT OF 1982

Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982 provides that the NRC may require, as a precondition to issuing or renewing an operating license for a research or test reactor, that the applicant shall have entered into an agreement with the Department of Energy (DOE) for the disposal of high-level radioactive waste and spent nuclear fuel. Aerotest Operations, Inc has a fully executed contract with DOE (DOE Contract: DE-CR01-83NE44484, dated July 14, 1983) (Reference 3) that provides that DOE retain title to the fuel and that DOE is obligated to take the spent fuel and/or high-level waste for storage or reprocessing when the ARRR facility is decommissioned. Because the ARRR has entered into such a contract with DOE, the applicable requirements of the Waste Policy Act of 1982 have been satisfied.

1.7 FACILITY HISTORY AND MODIFICATIONS

Date:	Event:
April 16, 1965	Construction Permit No. CPRR-86 issued by the Atomic Energy Commission to Aerojet-General Nucleonics for the construction of a 250 kW pool-type reactor at the Aerojet-General Nucleonics plant near San Ramon, California.
July 2, 1965	Facility License number R-98 is issued for the Aerojet General Nuclear Industrial Reactor (AGNIR)
July 9, 1965	AGNIR achieves initial criticality.

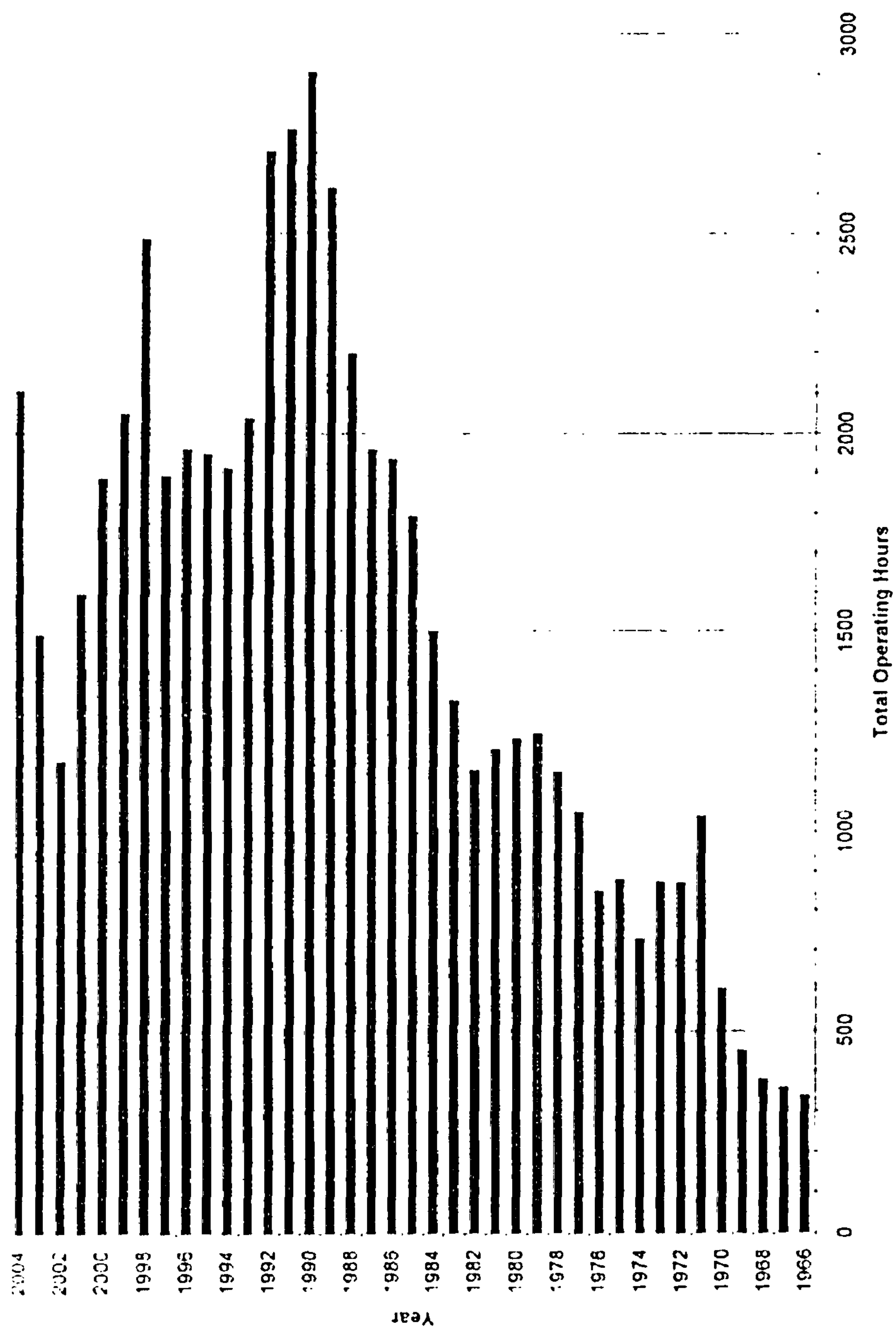
March 17, 1966	**Tech Spec Change No. 1:** Changed operational procedures and instrumentation requirements to reflect actual operational experience. (The number of changes are in parenthesis): Containment (1) Reactor core (6) Reactor safety systems (1) Experimental facilities (7) Nuclear instrumentation (5) Safety system functions (2)
Aug. 1, 1966	**Tech Spec Change No. 2:** Authorized the loading of [redacted] active fuel elements into the core in addition to the [redacted] elements then in use.
May 16, 1967	**Tech Spec Change No. 3:** Authorized irradiation of small amounts of ammonium perchlorate; and, Relocated gas sampler intake from roof vent to above the reactor pool.
Aug. 15, 1968	**Tech Spec Change No. 4:** Authorized installation of N-ray facility for class C explosives; and, Revised the method of monitoring high water radioactivity at the surface of the reactor pool.
April 20, 1970	**Tech Spec Change No. 5:** The main Aerojet Nuclear Division Plant shuts down, necessitating AGNIR to operate as a separate entity; and, A new Emergency Plan is submitted.
July 2,1970	**Tech Spec Change No. 6:** Authorized a new neutron radiography location; and, Authorized new radiation exposure limitations for explosive devices.
June 24,1971	**Tech Spec Change No. 7:** Addition of a second neutron radiography facility. Changes the limitations of explosive devices that may be radiographed. Deletes in-core irradiation of explosives.

Oct. 22, 1974	**License Amendment No. 1 and Tech Spec Change No. 8:** Transfers license from Aerojet-General Corp. to Aerotest Operations, Inc. and re-designates the reactor as the ARRR.
Feb. 24, 1977	**License Amendment No. 2:** Revised the security plan.
July 18, 1979	**License Amendment No. 3:** Revised the security plan.
Jan. 28, 1981	**License Amendment No. 4:** Rewords Tech Spec paragraph 11.2 regarding in-tank fuel storage facilities.

Figure 1-1
ARRR Facility Layout

[redacted]

Figure 1-2
ARRR Annual Operating Hours
(1966 – 2004)



1.8 REFERENCES:

1.8.1 Aerojet-General Nucleonics (AGN) Industrial Reactor: Hazards Summary Report (AN-1193), R. L. Newacheck, Project Engineer et al, September 1964.

1.8.2 Aerojet-General Nucleonics Industrial Reactor (AGNIR) Reactor Physics Tests (AN-1527), R. L. Tomlinson, August 1966.

1.8.3 Letter, Department of Energy (T. S. Keefe, Office of Procurement Operations) to R. L. Newacheck, Aerotest Operations, Re: Contract DE-CR01-83NE44484, July 14, 1983.

1.8.4 Directory of Operating Research, Training and Test Reactors in the United States of America, United States Department of Energy, 4th Edition, 1977.

2.0 SITE CHARACTERISTICS

2.1 GEOGRAPHY AND DEMOGRAPHY

The ARRR facility is located inside a [redacted] perimeter fence with a [redacted] gate that surrounds the ARRR facility and forms both the ARRR restricted area, as defined in 10 CFR 20, and the ARRR exclusion area, as defined in 10 CFR 100. The minimum distance from the center of the reactor pool to the boundary of the exclusion area fencing is [redacted]. [redacted]

The ARRR site is located in the town of Danville in Contra Costa County, California. The site is in the San Ramon Valley, about 23 miles east of San Francisco and 10 miles east of Oakland. It is separated from the East Bay urban complex by a series of ridges and hills up to 1600 ft high. The site is located approximately 9 miles south of Walnut Creek. Interstate 680 passes approximately 0.4 miles west of the site. Access to the ARRR site is via an extension to the site access road, Fostoria Way. The north end of the ARRR facility faces Fostoria Way with industrial buildings located on the opposite side of the street. The ARRR site is surrounded on the other three sides by Pacific Gas and Electric (PG&E) which uses the property for testing and research.

Although the ARRR site was originally in an area of sparse population, the AGNIR Hazards Summary Report (Reference 2.6.1) noted that agricultural land parcels in the immediate vicinity were for sale and development was expected. Currently, approximately 50% of the area within a one mile radius of the ARRR facility has been developed into suburban communities of single family homes and condominiums including a housing development located on the golf course of the Crow Canyon Country Club and portions of a housing development located on the Canyon Lakes Golf Course. Numerous light industrial, retail and community facilities are located within the one mile radius including the San Ramon Regional Medical Center which is approximately one-half mile from the ARRR facility. Approximately 35% of the area within a three mile radius of the ARRR facility has been similarly developed and includes a third golf course, The Bridges at Gale Ranch. The residential population within the three mile radius of ARRR facility is conservatively bounded at 44,076 which is the US Census 2000 population for the 94583 zip code in which the facility is located.

Despite the development since the ARRR facility was constructed, the surrounding area within the one mile radius remains significantly less populated than the areas surrounding similar TRIGA reactors in the United States which are typically located on university campuses and in hospitals with surrounding highly populated areas.

Locating TRIGA reactors in highly populated areas has been evaluated as acceptable because, as described in Reference 2.6.2, the radiological consequences for the maximum credible accident for a typical TRIGA reactor could be expected to affect only those personnel within the facility and would have no significant impact on the public heath and safety. Based on the reasons described in USAR Chapter 13, the consequences of the maximum credible accident for the ARRR are very conservatively bounded by the results described in Reference 2.6.2 for the typical TRIGA reactor.

Figure 2-1 is an aerial photograph that shows the ARRR facility buildings and perimeter and the buildings and perimeters of the adjacent properties.

Figure 2-2 is a street map of the area surrounding the ARRR facility.

Figure 2.3 is a topographic map of the area surrounding the ARRR facility.

Figure 2-1
Aerial Photograph of the ARRR Facility

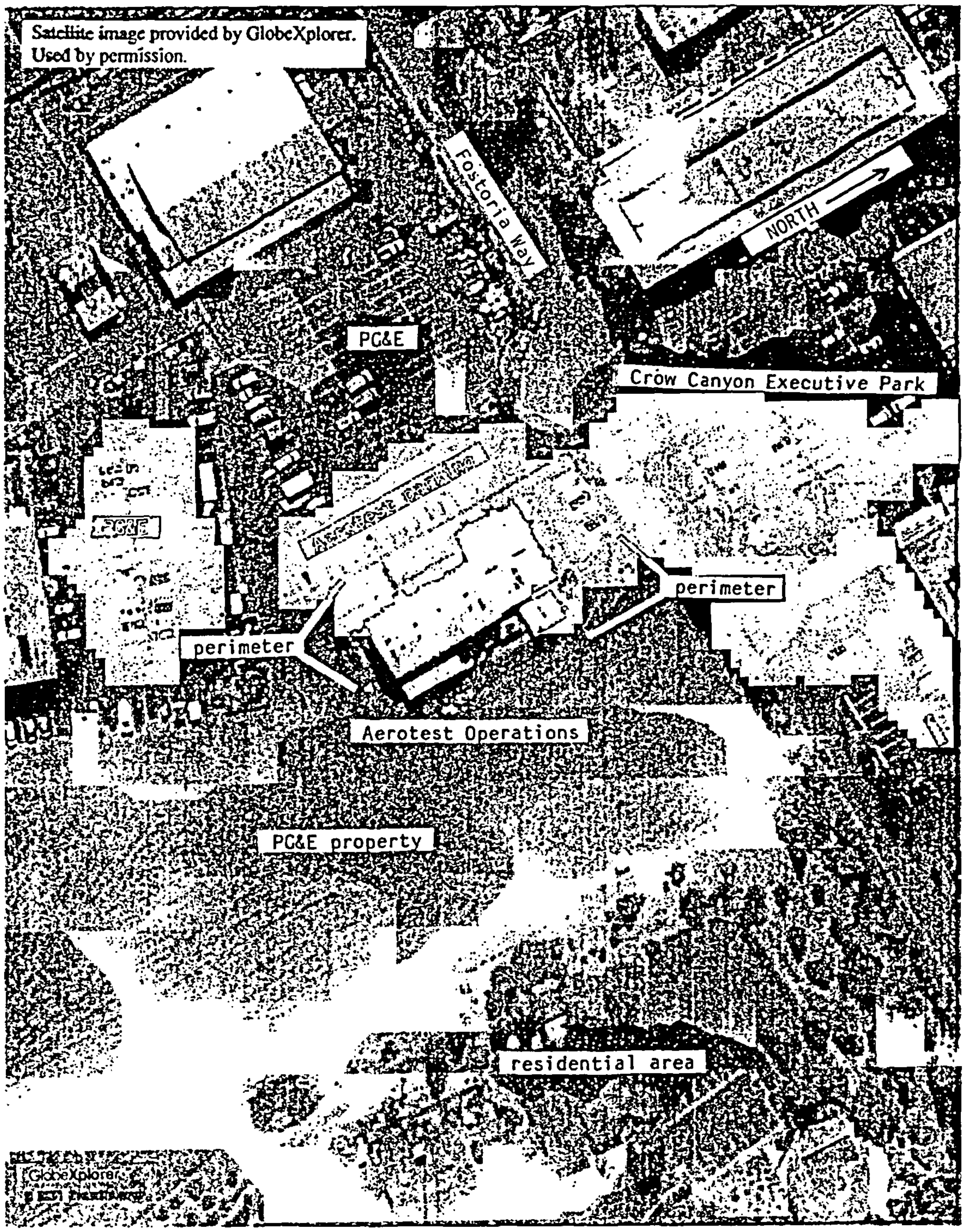


Figure 2.2
Street Map of the Area Surrounding the ARRR Facility

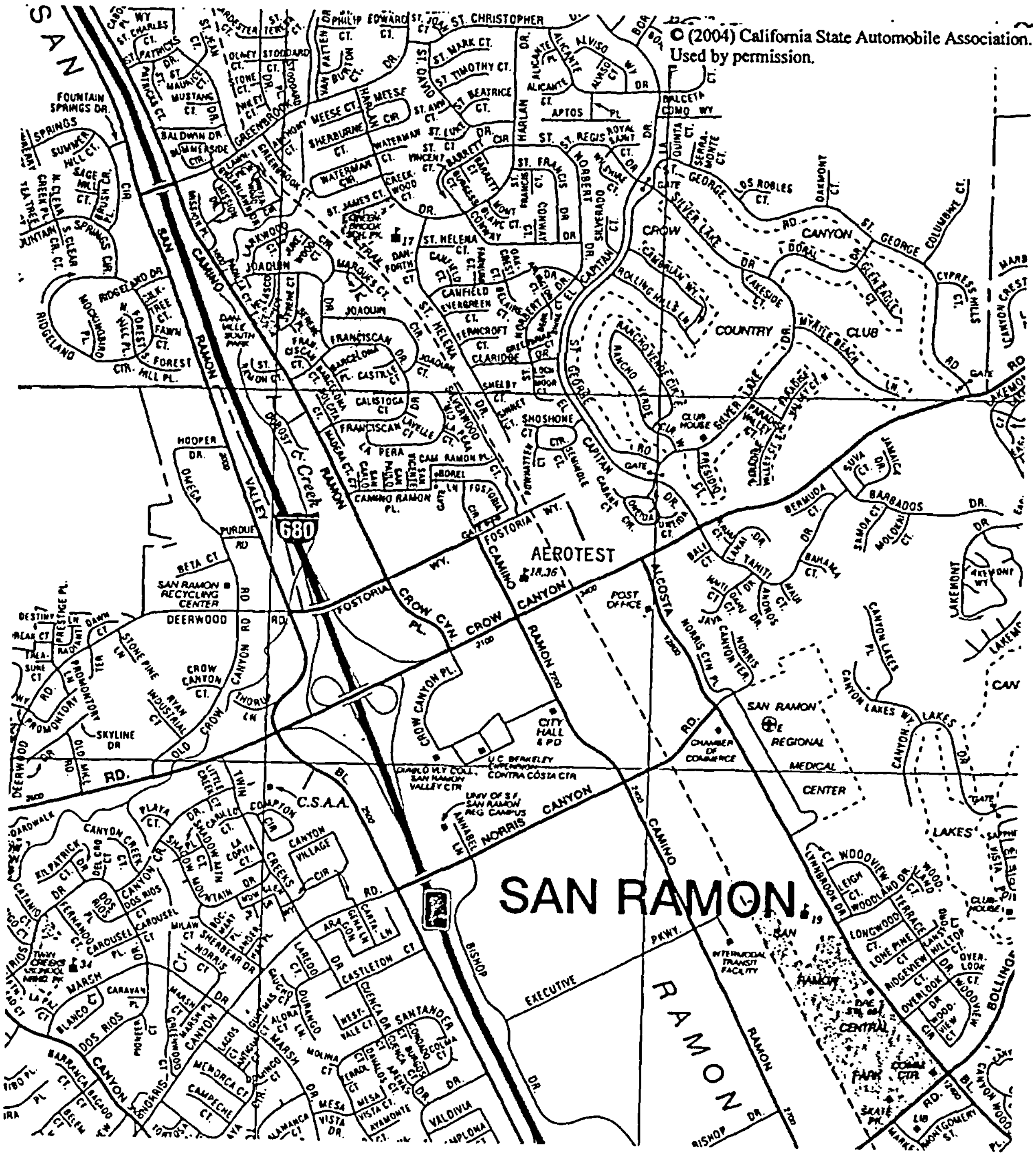
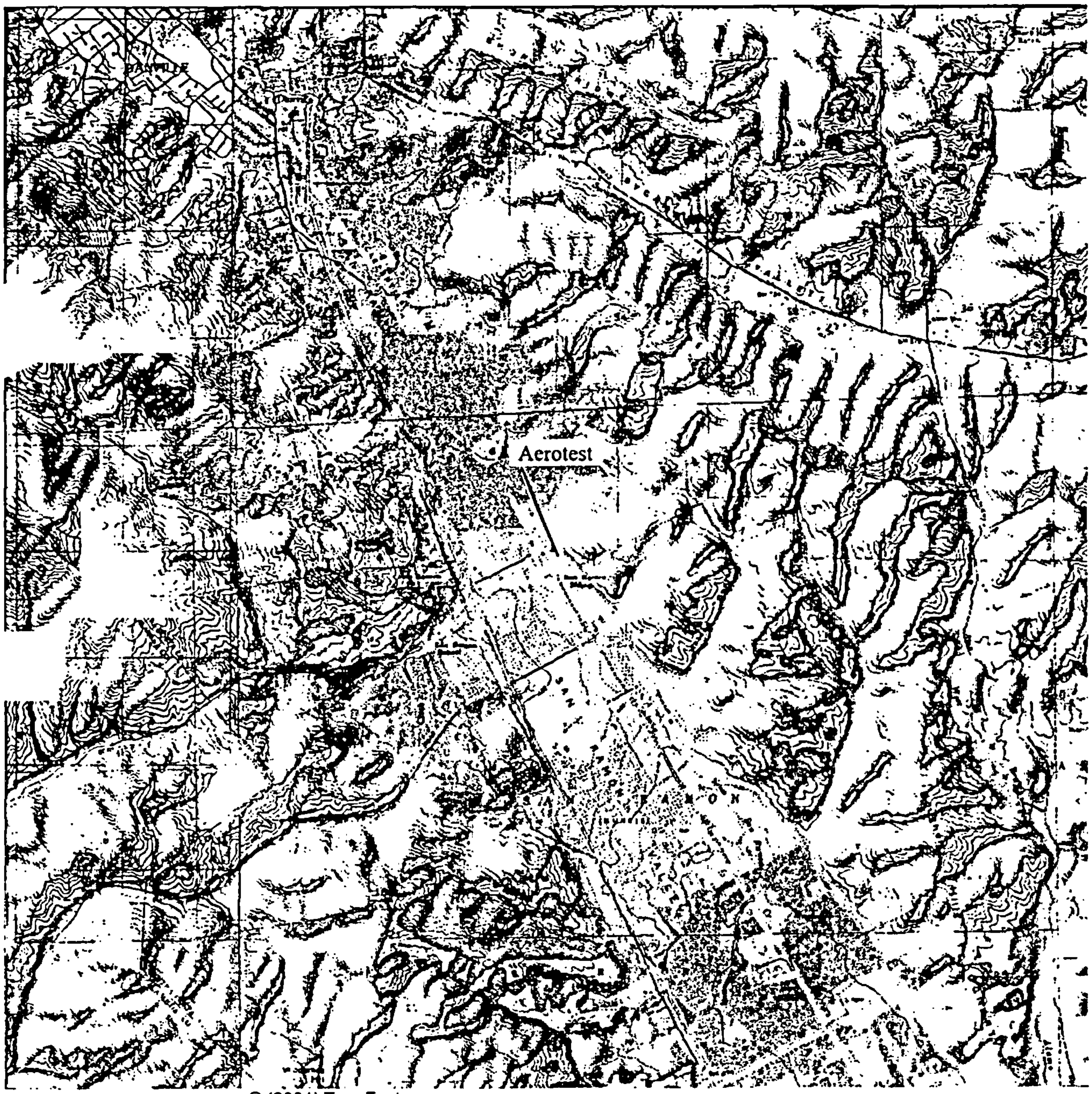


Figure 2.3
Topographic Map of the Area Surrounding the ARRR Facility



2.2 NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY FACILITIES

The ARRR site is located on property zoned for light industrial use. As stated earlier, the site is surrounded on three sides by a PG&E facility used for testing and research. The facilities located opposite the ARRR site on Fostoria Way include Electro-Test, Inc., Integrated Security Professionals, EKC Technology Inc., and CENCAL Insurance Services, Inc. Suburban communities of single family homes and condominiums, numerous small retail outlets and 3 golf courses account for most of the occupied land within both the one mile and three mile radii of the ARRR site. Some of the original agricultural use and designated open spaces account for the remainder of the land in the immediate vicinity of the site.

2.2.1 Locations and Routes

The principal transportation route near the ARRR site is Interstate 680 which was built after the ARRR facility was constructed and passes approximately 0.4 miles west of the site. San Ramon Valley Boulevard runs parallel to Interstate 680 approximately 0.5 miles west of the site. Crow Canyon Road is approximately 0.2 miles south of the site. The ARRR building is set back approximately 80 feet from the nearest public road, Fostoria Way.

2.2.2 Air Traffic

Airports in the vicinity of the ARRR site include: Livermore Municipal Airport, a small commercial airport, located approximately 10 miles southeast of the ARRR site; Oakland International Airport located approximately 14 miles west of the ARRR site; and, Hayward Air Terminal located approximately 11 miles southwest of the ARRR site.

2.2.3 Analysis of Potential Accidents at Facilities

There are no industrial, transportation or military facilities within the vicinity of the reactor site that have the potential for accidents with consequences significant enough to affect reactor operation or safety. Air traffic density and patterns in the vicinity of the ARRR site are typical of a suburban community and the probability of an aircraft accident involving the ARRR facility is extremely low. Additionally, the ARRR reactor water tank is located below ground level making damage to the reactor core from aircraft very unlikely. Additionally, as stated in USAR Chapter 6, Engineered Safety Features, and Chapter 8, Electrical Power Systems, there are no accidents or transients that depend on the availability of electrical power and no structures, systems or components (SSC) are assumed to be operable for the mitigation of any accident or the protection of the public health and safety. Finally, in the unlikely event of a transportation accident involving hazardous material that affected the ARRR site and the surrounding suburban housing developments, the ARRR reactor could be shut down promptly and the facility evacuated.

2.3 METEOROLOGY: GENERAL AND LOCAL CLIMATE

The San Ramon area has a temperate climate and high winds and severe storms are abnormal and infrequent. Hurricanes, tsunamis, and flooding do not occur in the San Ramon area. The highest monthly average local wind speeds are 9.6 mph for Livermore and 10.5 mph for Oakland. The highest recorded wind speed was 67 mph recorded in Oakland in 1938. The ARRR building is designed to withstand the area wind loads.

The average annual precipitation is about 15 inches, varying between 7 and 25 inches for very dry and very wet years, respectively. Most of the precipitation (virtually all of which is rainfall) occurs during the winter months in connection with Pacific storms, which occasionally bring enough rain to cause water to stand in some fields. Thunderstorms are rare, occurring on the average only one day a year, usually in January.

Flooding is not expected at the ARRR site. However, even if flooding occurred, reactor safety would not be an issue since the core is located in a water pool. Negligible amounts of solid precipitation (snow, sleet, or hail) occur, since the surface temperature is below freezing only about 1% of the time.

Table 2.3.1
San Ramon Area Weather

	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec
Average Temperature (°F)	47.8	50.0	51.4	55.2	59.8	65.8	70.9	71.0	69.0	63.0	53.3	48.2
Average High Temperature (°F)	56.2	58.9	61.1	66.4	71.8	78.7	84.2	84.2	81.9	74.9	63.0	56.8
Average Low Temperature (°F)	39.3	41.1	41.7	44.0	47.9	52.8	57.6	57.7	56.1	51.1	43.5	39.4
Average Precipitation (inches)	4.5	4.2	3.6	1.4	0.7	0.1	0.0	0.1	0.3	1.3	3.1	3.2
Average Number of Days with Precipitation.	10	9	9	5	3	1	0	0	1	3	7	9
Average Wind Speed (mph)	7.0	7.2	8.3	8.5	9.0	9.5	8.8	8.3	7.3	6.3	5.9	6.3

A detailed analysis of the weather patterns in the area at the time the ARRR facility was constructed is presented in Appendix A of Reference 2.6.1.

2.4 HYDROLOGY

San Ramon lies at the southern end of the San Ramon Valley, which drains into the mouth of the Sacramento River (Suisun Bay) and eventually into the Pacific Ocean. San Ramon is bordered on the south by the Amador Valley, on the east by the Sherburne

Hills, and on the west by Las Trampas Ridge. The ARRR site was selected to be above any potential flood level.

The principal watercourse in the vicinity of the ARRR site is San Ramon Creek. During the wet season, any run-off from the ARRR site would enter San Ramon Creek and would proceed northward into Walnut Creek and into Suisun Bay. During the dry season, which consists of two-thirds of the typical year, any release of water from the ARRR plant would percolate underground (Reference 2.6.1).

The ARRR facility waste water is discharged into underground pipelines and is collected and cleaned by the Central Contra Costa Sanitary District (CCCSD). To ensure compliance with discharge requirements, CCCSD samples and analyzes Aerotest photo processing waste water semiannually. Holding tank waste water is analyzed prior to release to the sanitary system. Aerotest Operations is a recent recipient of CCCSD's Annual Pollution Prevention Award, which recognizes outstanding efforts to protect the local water environment by minimizing or eliminating harmful discharges into the sewer or storm drain systems.

2.5 GEOLOGY, SEISMOLOGY, AND GEOTECHNICAL ENGINEERING

San Ramon is located in the Southern Coast Range, to the southwest of Mount Diablo. The elevation of the ARRR plant site is about 460 ft above sea level. The basic subsoil structure is sandstone over 200 ft thick. The loam-like surface soil varies from 15 to 30 feet in depth and is sufficiently porous to allow percolation to the highly absorbent subsoil structure. According to local well drillers, the true water table is reached at a depth of approximately 200 feet (Reference 2.6.1).

The area is underlain by a basement of Franciscan rocks belonging to the Northern Franciscan Area. This structure is characterized by the ability to yield under deforming forces even when covered by sedimentary rocks, as is the case for the ARRR site.

2.5.1 Site Topography

The site is in the San Ramon Valley, about 23 miles due east of San Francisco and 10 miles east of Oakland. It is separated from the East Bay urban complex by a series of ridges and hills up to 1600 feet high. A topographical map of the area is shown in Figure 2.4.

2.5.2 Seismic Characteristics

The ARRR site was selected at a location where no earth fractures were known to exist. Slight tremors are fairly common in the San Ramon area. Although severe quakes in this area are uncommon, the ARRR facility is designed to tolerate the maximum credible earthquake (MCE) without significant damage to the ARRR facility and without significant impact on the public health and safety. The seismology of the ARRR site was evaluated in three separate studies:

(1) ARRR Site Seismic Evaluation (1964)

Reference 2.6.1, Appendix B, provides details of a study of the area with regard to the hazards due to earthquake and describes the site seismic evaluation performed prior to the construction. This study consisted of an evaluation of each of the known seismic faults in the East Bay area. This study identified that the ARRR site is in proximity to the Sunol Fault. The exact position of the fault in the vicinity of San Ramon is uncertain, but is probably 2000 feet west of the ARRR site. Additionally, the fault is of limited activity. Large earthquakes and actual fault displacement at the site are not to be expected along this line of faulting.

(2) ARRR Site Seismic Evaluation (1973)

Reference 2.6.3, Geotechnical Investigation for Aerotest Facility Addition, Peter Kaldveer and Associates, Inc., Geotechnical Consultants reports on a detailed engineering evaluation of the ARRR site to support an addition to the building. The purposes of the evaluation were (1) to determine if an active fault underlies the building addition site, and (2) to evaluate the foundation soils and provide recommendations concerning the soil and foundation engineering aspects of the project. The evaluation consisted of the following:

(a) A detailed review of previous studies performed in the general area of the site.

(b) A study of black and white aerial photographs of the site by a consulting engineering geologist.

(c) A detailed reconnaissance of the site by a consulting engineering geologist.

(d) Excavation and logging of a trench approximately 12 feet deep and 150 feet long by consulting engineering geologist.

(e) Drilling and logging of three borings adjacent to the existing building and proposed addition to a maximum depth of 15 feet by a soil engineer.

(f) Laboratory testing of the soils encountered in the borings.

(g) Geotechnical evaluation and analyses of the data obtained during the study.

(h) Preparation of the geotechnical investigation report summarizing conclusions.

Based on the geologic reconnaissance, air photo study, exploratory borings and trench, and review of previous geotechnical reports and other geologic information for the ARRR site area, this study concluded that there is no evidence for faulting within the building site area. At least one moderate to severe earthquake is likely during the design life of the project. During such an earthquake, strong ground shaking of the property will occur, however, the likelihood of fault offset through the site is slight.

(3) ARRR Site Seismic Evaluation (1993)

"Evaluation of Seismic Potential at the Aerotest Facility from an MCE on the Calaveras Fault," (Reference 2.6.4) is an evaluation performed by Bechtel Corporation intended to ensure that the safety analyses performed for the ARRR in 1964 remain valid. The Bechtel evaluation included a review of all current knowledge regarding the Northern Calaveras fault, review of on-site soils and foundation conditions, examination of plans and "as-built" status of facilities and structures on the site, and an assessment of the consequences of the Maximum Credible Earthquake to the ARRR facility using the criteria given in NRC Regulatory Guide 1.60, Design Response Spectra for Seismic Design of Nuclear Power Plants (Reference 2.6.5). The evaluation included both the original design and design changes that have occurred since 1964.

The Bechtel study concluded that the ARRR facility site is not intersected by active faults. It is about 1.4 km east-northeast of the Calaveras fault which has been depicted in recent studies as the potential site of a moderate to large earthquake. The largest ground motion expected from a reasonably expectable maximum earthquake on the segment of the Calaveras fault closest to the ARRR facility can be adequately modeled within guidelines in NRC Regulatory Guide 1.60 (Reference 2.6.5) by a design response spectrum anchored at a high frequency asymptote of 0.50 g. ARRR components are designed for lateral acceleration in excess of 1.0 g. Note that the ARRR Reactor Protection System will initiate a reactor scram in the event of a seismic event with an average peak acceleration of 0.015 g to 0.02 g to ensure the reactor is shut down at the initiation of a major earthquake.

The Bechtel study noted that the foundation materials at the ARRR, stiff clays overlying weathered rock, would be expected to remain stable under, and not to modify, the above ground motions. Therefore, this study concluded that the reinforced concrete, aluminum-lined tank forming the pool provides a leak-tight tank sufficient to resist the forces of the earthquake. Although the potential exists for a significant loss of water from the reactor water tank during an earthquake due to sloshing, the loss of water inventory from the 10 foot diameter tank is not expected to uncover the core which is positioned more than 18 feet below the upper rim of the reactor water tank. Water lost from the reactor tank due to sloshing would be collected in the trench surrounding the tank and would not create any significant hazards. As described in USAR Chapter 13, Accident Analyses, even an instantaneous loss of all cooling water will not result in a significant threat to the public health and safety because this event will not result in fuel temperatures that could cause the cladding to melt or a phase change in the zirconium hydride in the fuel in either the aluminum or stainless steel clad fuel.

The Bechtel study recommended additional calculations to confirm that the trolley system, large perimeter concrete blocks, and the heavy structural timbers above the pool will behave acceptably during a seismic event. However, Reference 2.6.6 notes that these recommendations contained in the Bechtel report relate to questions of potential on-site property loss, and not to public health and safety.

2.5.3 Maximum Earthquake Potential

As stated above, the largest ground motion expected from a reasonably expectable maximum earthquake on the segment of the Calaveras fault closest to the ARRR facility can be adequately modeled within guidelines in NRC Regulatory Guide 1.60 (Reference 2.6.5) by a design response spectrum anchored at a high frequency asymptote of 0.50 g. ARRR components are designed for lateral acceleration in excess of 1.0 g. Therefore, the ARRR design is sufficiently conservative to assure that the reactor can be safely shut down in the event of a major earthquake and that the potential for core damage that might result in the release of fission products is very small.

2.6 REFERENCES

2.6.1 "Aerojet-General Nucleonics Industrial Reactor (AGNIR): Hazards Summary Report (AN-1193)," R. L. Newacheck, Project Engineer et al, September 1964.

2.6.2 NUREG/CR 2387, "Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors," S.C. Hawley and R.L. Kathren, US Nuclear Regulatory Commission, April 1982.

2.6.3 "Geotechnical Investigation for Aerotest Facility Addition," Peter Kaldveer and Associates, Inc., Geotechnical Consultants, November 1979.

2.6.4 "Evaluation of Seismic Potential at the Aerotest Facility from an MCE on the Calaveras Fault," Bechtel Corporation, April 1993.

2.6.5 NRC Regulatory Guide 1.60, Design Response Spectra for Seismic Design of Nuclear Power Plants, Revision 1, 1973.

2.6.6 Letter, R.R. Tsukimura (Aerotest Operations, Inc.) to R.J. Pate (NRC), "Seismic Potential of the Northern Calaveras Fault," May 5, 1993.

3.0 DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS

The ARRR (originally AGNIR) is a TRIGA Mark I open pool reactor that was designed and constructed by the Nuclear Division of Aerojet-General in 1964. The TRIGA reactor fuel elements, reflector elements, control rods, control rod drive mechanisms, control rod drive controls, and reactor protection system were purchased from General Atomics and were incorporated into the ARRR without any significant changes.

TRIGA reactors were conceived and developed by the General Atomic Division of General Dynamics Corporation with the objective of providing a training, research, and isotope production reactor containing intrinsic safety features designed to significantly reduce the probability and consequences of a nuclear accident. To achieve this goal, General Atomic designed the TRIGA reactor fuel to have intrinsic physical properties that shut down the reactor or limit its power to a safe value in the event of an inadvertent addition of positive reactivity. General Atomic also designed the TRIGA reactor fuel element such that fuel clad temperature will remain significantly less than 660°C and well below the melting point of both the aluminum and stainless steel fuel cladding following an instantaneous loss of all cooling water following extended full power operation (References 3.3.1 and 3.3.3).

The accident analyses for the ARRR described in USAR Chapter 13, Accident Analyses, and NUREG/CR 2387, "Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors," (Reference 3.3.3), are based entirely on the design of the TRIGA fuel elements and the low source term present in a small research reactor. No specific structures, systems, or components (SSC) are assumed to be operable for the prevention or mitigation of any accident or the protection of the public health and safety. Because the ARRR accident analyses do not rely on any specific features provided in the equipment and building that surround the reactor, there were no design criteria for the protection of ARRR SSC from meteorological damage, water damage or seismic damage except that all facility structures were constructed to the applicable industrial building codes in existence at the time the facility was designed.

Although not relied upon in the USAR Chapter 13, Accident Analyses, certain ARRR SSC are intended to minimize the potential for challenges to fuel cladding integrity or to mitigate the consequences of events that could challenge fuel cladding integrity. These include the following:

(1) The Reactor Water Tank, Reactor Bridge Structure, Control Rods and Control Rod Drive Mechanisms which are addressed in USAR Chapter 4, Reactor.

(2) The Reactor Protection System and Reactor Monitoring System which are addressed in USAR Chapter 7, Instrumentation and Control Systems.

(3) The Neutron Radiography Facility and Other Experimental Facilities which are addressed in USAR Chapter 10, Experimental Facilities and Explosives.

(4) The Reactor Building and Reactor Building and Control Room Ventilation Systems which are addressed below.

The mechanical and electrical systems important to safety listed above are readily accessible for visual inspection, testing, and maintenance. A preventive maintenance program has been in effect for many years at the ARRR to ensure that operability of these systems is in conformance with the performance requirements of the Technical Specifications. Additionally, the operating history of the ARRR includes few significant malfunctions of electrical or electromechanical systems and no persistent malfunction of any one component.

3.1 REACTOR BUILDING

The Aerotest Operations building is constructed of steel with internal rooms built of fire resistant framing with sheetrock covering. It is designed to contain the main reactor room, the control room, a laboratory area, and the other areas shown in Figures 3-1 and 3-2. The building is designed to meet local building codes for earthquake-resistant buildings. An automatic [redacted] system covers the entire building and [redacted] are strategically located throughout the building. [redacted]

The reactor and neutron radiography facility are enclosed in the 40 X 80 foot high bay area. This area encloses the reactor water tank which is embedded in the floor, extending [redacted] below the surface and one foot above the floor. A [redacted] thick by [redacted] high block wall made of normal density concrete encloses the reactor area above the floor level and includes the neutron radiography facility. The reactor and reactor water tank are described in USAR Chapter 4, Reactor. The neutron radiography facility is described in USAR Chapter 10, Experimental Facilities and Explosives.

[redacted]

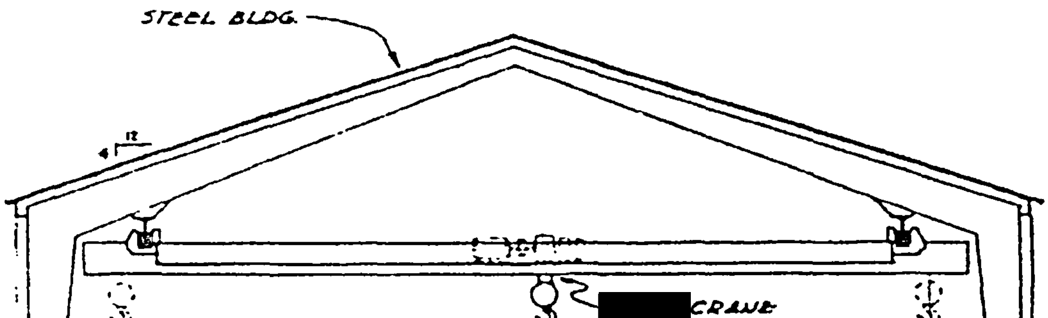
A [redacted] bridge crane serves most of the work area of the reactor high bay area, including the fuel storage pits. The bridge crane is used for general work, including the handling of heavy shields.

The Aerotest Operations building is not assumed to act as either a containment or confinement and is not assumed to mitigate the release of radioactivity following a reactor accident. In fact, the analysis of the maximum fission product release following a fuel element failure (Appendix C of Reference 3.3.1) assumes 0.5 to 2.0 air changes per hour in the building and that all fission product gases leak out of the building within one hour. This assumption is very conservative because the original building included large evaporative coolers which purged the building with up to 9,000 cfm of air. These coolers were removed in 1970.

Figure 3-1
Plan Layout of the ARRR Building

[redacted]

Figure 3-2
Cross Section ARRR Reactor Building
Showing Reactor High Bay Area and Bridge Crane



[redacted]

3.2 REFERENCES:

3.2.1 "Aerojet-General Nucleonics Industrial Reactor (AGNIR): Hazards Summary Report (AN-1193)," R. L. Newacheck, Project Engineer et al, September 1964.

3.2.2 "Aerojet-General Nucleonics Industrial Reactor (AGNIR): Reactor Physics Tests (AN-1527)," R. L. Tomlinson, August 1966.

3.2.3 NUREG/CR 2387, "Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors," S.C. Hawley and R.L. Kathren, US Nuclear Regulatory Commission, April 1982.

4.0 REACTOR

4.1 SUMMARY DESCRIPTION

The ARRR (originally AGNIR) is a modified TRIGA Mark I reactor. It is an open pool type reactor with the pool (i.e., reactor tank) located below ground level (Figure 4.1-1). The ARRR was designed and constructed by the Nuclear Division of Aerojet-General in 1964. The reactor achieved initial criticality on July 9, 1965 with a licensed steady-state thermal power limit of 250 kW. There is no pulsing capability which is sometimes a design feature for similar reactors. The "Aerojet-General Nucleonics Industrial Reactor (AGNIR): Hazards Summary Report (AN-1193)," (Reference 4.9.1) summarizes the safety analysis used to license the ARRR.

4.1.1 TRIGA Reactor – General Description

The ARRR is a light water cooled and moderated reactor with the water also serving as a neutron reflector and biological shielding. Water, moving by natural convection, is employed as the primary reactor coolant. The coolant serves to remove the heat of fission from the fuel. The ARRR also includes a flow path through a heat exchanger if forced convection cooling of the reactor is desired. The primary circulating system draws water from near the top of the pool and pumps it through a heat exchanger where a secondary loop removes heat from the primary coolant. The secondary water is pumped through a forced air cooling tower which cools the secondary water by evaporation. The pool water is returned to the bottom of the pool where it is discharged tangentially to cause a spiraling of the cooling water. A demineralizer system circulates the reactor coolant through a filter and demineralizer to maintain required water chemistry.

The reactor fuel elements, reflector elements, control rods, control rod drive mechanisms, and control rod drive controls were purchased from the General Atomics Company and were incorporated into the ARRR without any significant changes. The standard "G" ring core grid plate design was provided by General Atomics and manufactured by Aerojet. All other components were designed and constructed by Aerojet or their subcontractors.

The reactor also contains the following facilities for experiments which are described in USAR Chapter 10, Experimental Facilities and Explosives:

(1) Neutron Radiography Facility;

(2) Graphite Thermal Column;

(3) Glory Hole Facility;

(4) Vertical Tubes;

(5) Central Core Irradiation Facility;

(6) Triangular Incore Irradiation Facilities;

(7) Incore Irradiation Capsules;

(8) Large Component Irradiation Box (Use prohibited by Technical Specifications);

(9) Pneumatic Transfer Facility (Use prohibited by Technical Specifications); and

(10) Beam Port (Not Installed).

4.1.2 TRIGA Reactors - Intrinsic Safety Features

By 1956, the year General Atomic was formed, considerable progress had already been made in devising safety systems to minimize the possibility of reactor accidents. All nuclear reactors had some type of mechanical device to shut down the reactor if the power level became excessive. These usually employed circuits that would electronically measure the level of radiation received by a detector, and if the radiation level or its rate of increase exceeded preset levels, then the safety circuits would initiate a reactor shutdown or "scram" by inserting the control rods. In addition to these safety circuits, most reactors have a shutdown mechanism that is an intrinsic property of their cores called the delayed negative temperature (bath) reactivity coefficient. This negative temperature coefficient, however, provides a shutdown mechanism only after a relatively long time, thus the reactor may be damaged before power is reduced by this mechanism.

TRIGA reactors were conceived and developed by the General Atomic Division of General Dynamics Corporation with the objective of providing a training, research, and isotope production reactor containing intrinsic safety features designed to significantly reduce the probability and consequences of a nuclear accident. Rather than depend entirely on electronic circuitry, moving parts, and the delayed negative temperature coefficient, General Atomic felt that it would be desirable to incorporate some intrinsic physical property of the fuel itself that would shut down the reactor or limit its power level to a safe value. As its initial contribution to reactor technology, General Atomic conceived and developed a unique fuel element consisting of a solid homogeneous alloy of uranium fuel and zirconium-hydride moderator (U-ZrH). These unique fuel elements provide the TRIGA reactor with a large prompt negative temperature coefficient. This means that any increase in power heats both the fuel and the moderating material in the fuel element simultaneously. Therefore, the temperature increase immediately causes the fuel to become less effective by decreasing the reactivity of the core. This causes the reactor to return automatically to normal operating levels within milliseconds.

The intrinsic self-regulating characteristic is basic to the TRIGA reactor core, and permits safe, steady-state operation and pulsing to high power levels without the use of any mechanical or electronic control devices to return the reactor to a safe power level. The ARRR is not pulsed; however the instantaneous introduction of a sufficient amount of excess reactivity to cause a large power increase (approximately 1000 fold) will not result in any damage to the fuel elements.

Figure 4.1-1
Cutaway Drawing of the ARRR Reactor

[redacted]

4.2 REACTOR CORE

The ARRR reactor core consists of standard TRIGA fuel elements, graphite reflector elements, 3 control rods and guide tubes, a neutron source, and irradiation facilities (Figure 4.2-1). The core forms a right circular cylinder and consists of a lattice of cylindrical fuel elements and graphite dummy elements immersed in water. The elements are spaced so that 33% of the core volume is occupied by water. The fuel-to-water ratio in the core was selected because calculations show that it gives nearly the minimum critical mass. The basic nuclear design and core geometry follow General Atomics TRIGA reactor design characteristics.

4.2.1 Fuel Element Inventory

The fuel elements are supported by upper and lower grid plates (discussed below) with a total of [redacted] grid positions ([redacted] in [redacted] concentric rings around a central hole) available for core components (Figure 4.2-1) including fuel elements, graphite dummy elements, control rods, a neutron startup source and a removable glory hole facility (USAR Chapter 10, Experimental Facilities and Explosives).

At initial criticality, the ARRR core included [redacted] aluminum clad TRIGA fuel elements. The design was intended to allow for more fuel elements to be added, as necessary, to compensate for fuel burn up. Because aluminum clad TRIGA fuel elements are no longer manufactured, most of the fuel elements added to the ARRR are stainless steel clad. General Atomic has been using a mixed core of stainless steel and aluminum clad fuel in TRIGA reactors since 1960 when they were first authorized to use a limited number of stainless steel clad together with aluminum clad elements. As explained in USAR Chapter 13, Accident Analyses, use of stainless steel clad fuel elements results in additional margin in the accident analyses. As of 2004, the ARRR contains [redacted] active fuel elements consisting of [redacted] aluminum clad elements and [redacted] stainless steel clad elements Figure 4.2-2 shows a typical arrangement of the ARRR core. [redacted] core lattice positions are occupied by graphite dummy elements, 3 by control rods, one by a neutron startup source, and one by a removable glory hole facility.

The addition of new fuel elements is limited by the ARRR license which currently specifies that the core configuration be maintained within the following limits:

(1) The reactor core lattice shall contain [redacted] TRIGA type fuel elements.

(2) The reactor core lattice shall be loaded with [redacted] of U-235.

4.2.2 Fuel-Moderator Elements

The fuel elements are standard aluminum or stainless steel clad TRIGA fuel elements. The standard TRIGA fuel elements used in the ARRR had the following characteristics at the time of fabrication:

Fuel alloy:	zirconium-hydride moderator, homogeneously combined with uranium fuel
Enrichment:	≤ 20 wt % U-235 nominal
Cladding:	Aluminum elements: [redacted] thick Stainless steel elements: [redacted] thick
Hydrogen-to-zirconium atom ratio in the ZrH_x:	Aluminum elements: 92 wt % $ZrH_{1.1}$ Stainless steel elements: 88 wt % $ZrH_{1.6}$
Fuel loading (U-235): (nominal)	Aluminum elements: [redacted] Stainless steel elements: [redacted]
Active fuel length	Aluminum elements: [redacted] Stainless steel elements: [redacted]
Moderator:	Zirconium hydride and water
Reflector:	Demineralized water: (∞) Graphite : sides [redacted] top & bottom: [redacted] (aluminum) top & bottom: [redacted] (stainless)
Core (active) Dimensions:	Diameter: [redacted] Height: [redacted] (nominal)

The fuel portion of each fuel element contains a zirconium-hydride moderator, homogeneously combined with the enriched uranium fuel. The original aluminum clad fuel elements also contained Samarium, a burnable poison, to lengthen their lifetime (probably no longer effective). The graphite slugs at the top and bottom of each fuel element act as reflectors.

A typical aluminum clad fuel element is shown in Figure 4.2-3 and a typical stainless steel clad fuel element is shown in Figure 4.2-4. In each case, the fuel is contained by aluminum or stainless steel cladding which is welded to the top and bottom end fittings. The top end fitting is grooved and shaped to fit and lock into the fuel handling tools. The top end fitting also incorporates a triangular spacer block of the aluminum clad elements or three radial fins on stainless steel clad elements that positions the top of the element in the top grid and provides passages for cooling water through the grid plate. When properly installed in the core, the top of the triangular spacer block or triangular fins is level with the top of the top grid plate. The bottom end fits into the counter-sunk holes of the bottom grid plate and supports the entire weight of the element. Serial numbers on the top end fixture or spacer block are used to uniquely identify each fuel element.

Graphite reflector elements occupy many of the outer grid positions not used by fuel elements or other core components. The graphite reflector elements are clad in aluminum and have aluminum end fixtures and spacer blocks. These elements are of the same dimensions as the fuel elements, but are filled entirely with graphite. Originally, the spacer blocks had a blue, anodized finish to make the graphite reflector elements easily distinguishable from fuel elements. Currently, however, graphite reflector elements can only be segregated from the fuel elements by weight or the fuel element serial number. When properly installed in the core, the top of the triangular spacer block is about level with the top of the top grid plate.

4.2.3 Core Support Structure

The core structure, shown in Figure 4.2-5, consists of a cylindrical shroud and top and bottom grid plates. The core shroud, an aluminum right circular cylinder, is about [redacted] high and has an inside diameter of about [redacted]. [redacted] The center of the [redacted] to [redacted] active fuel region is approximately [redacted] above the bottom of the reactor pool. [redacted]

The elements are supported and spaced by upper and lower grid plates made of 6061 aluminum. The grid plates are [redacted]. As stated above, the upper and lower grid have a total of [redacted] grid positions ([redacted] in [redacted] concentric rings around a central hole) available for core components (Figure 4.2-2) including fuel elements, graphite dummy elements, control rods, a neutron startup source and a removable glory hole. Each grid position is identified by a letter and number combination using a letter for each concentric ring and a number for each position in that ring.

The upper grid plate is used to position the fuel elements and control rods using holes approximately [redacted] in diameter in the aluminum upper grid plate. The upper grid plate has a thickness of [redacted]. The top grid plate does not support any of the weight of the elements. The holes serve only to determine the lateral position of the elements and to permit their withdrawal from the core. Triangular spacer blocks or radial fins on the upper ends of the fuel elements are positioned in the upper grid plate holes in a way that allows water to flow out of the core through the upper grid plate. Small holes at various positions in the top and bottom grid plates permit insertion of wires into the core to obtain flux measurements.

[redacted]

The bottom grid plate holes receive the lower-end fixtures of the fuel elements. The bottom grid plate has a thickness of [redacted]. A shoulder is provided on the end fixture of each fuel element and the hole in the bottom grid plate is countersunk by a corresponding amount. The weight of the fuel element rests on this shoulder, not on the bottom of the end fixture of the fuel element, which is used only to position the element as it is being put into place.

The fuel is cooled by natural circulation of water in the pool which flows through the core from bottom to top. Spaces for passage of the cooling water through the bottom grid plate are provided by special holes and through the top grid plate by the gap between the triangular spacer blocks or triangular fins on the fuel elements and the round grid hole. Cooling water enters the core region through large access holes cut through the shroud between the upper and lower grid plates and through holes in the bottom grid plate. Water enters the region below the core through large openings in the lower portion of the shroud.

Instrument guide tubes (Figure 4.2-1) are provided to hold the four nuclear instruments for the four channels of instrumentation discussed in Chapter 7, Instrumentation and Control. These guide tubes are welded to the side of the shroud and are supported at the bottom by the T beam.

4.2.4 Neutron Startup Source

A specially designed source holder (Figure 4.2-6) occupies one of the core grid positions and contains the neutron source that is used to ensure the presence of the minimum neutron count rate needed to ensure a controlled reactor startup.

(1) Neutron Source

The current ARRR neutron source uses a mixture of beryllium (Be) and americium (Am), a radioactive alpha emitter, to produces neutrons by an alpha-neutron (α,n) reaction:

$$^{9}Be + \alpha \rightarrow {}^{13}C \rightarrow {}^{12}C + n$$

The advantage of americium-241 is its long half-life (458 years) and the emission of energetic alphas for interaction with the beryllium. The gamma activity associated with the decay of americium is of low energy and causes no real problems with neutron detectors when used for calibration. The ARRR neutron source had an original strength of 2 curies which gave about 5.1×10^6 neutrons per second. However, the americium source strength has increased slightly due to exposure to neutrons in the reactor. Americium-242 is produced in the source due to the high neutron cross section of americium-241. Americium-242 is a short-lived beta emitter that decays into Curium-242. In turn, Curium-242 has a half life of 163 days and is a more energetic alpha emitter than americium-241. Therefore, the neutron source becomes stronger when the reactor is operated.

(2) Neutron Source Holder

The neutron source holder (Figure 4.2-6) is a cylindrical aluminum tube. Although the dimensions of the holder permit it to be installed in any of the grid positions in the core, it generally occupies one of the [redacted]. When loaded into the holder, a neutron source (the small slug shown in Figure 4.2-6) is contained in a cavity in the lower half of the assembly. This positions the source approximately at the [redacted] of the core. A narrow shoulder at the top of the source holder supports the assembly on the upper grid plate, the holder itself extending down into the core region. The assembly does not rest on the lower grid plate. The upper and lower portions of the holder are screwed together. A soft aluminum ring seals the cavity against water leakage.

[redacted]

4.2.5 Control Rods and Drive Mechanisms

ARRR reactivity control is achieved by the operation of three control rods (1 safety rod, 1 shim rod and 1 regulating rod) and their associated motor driven control rod drive (CRD) assemblies. The three control rods are used to control reactor power level during normal operation and are designed for rapid insertion for emergency reactor shutdown. Each of the three control rod assemblies consist of a control rod, control rod guide tube and a CRD assembly (Figures 4.2-7 and 4.2-8).

(1) Control Rods

Each of the three control rods is a sealed aluminum tube containing several sintered, boron carbide cylinders as the neutron poison. The control rods are approximately [redacted] long with outside diameters of [redacted] for the regulating rod and [redacted] for the shim and safety rods. The upper end of the control rod screws into an assembly extension tube that connects it to the control rod drive assembly that is used to withdraw and insert the control rod. The vertical travel of the control rods is approximately [redacted].

Each of the three control rods can be inserted into any of the [redacted] grid positions in the core. To accommodate a control rod, one fuel element is replaced by a control rod guide tube that has the same outer diameter as a fuel element. The current position of the control rods in the reactor core grid is shown in Figure 4.2-2.

The reactivity worth of each rod is dependent on the core configuration. Reactivity worth of the control rods must be maintained to ensure that shutdown margin (SDM) of ≥ 0.365% $\Delta k/k$ (≥ \$0.50) is maintained as required by Technical Specification 3.1.3, Shutdown Margin.

Furthermore, in conjunction with the Technical Specification requirement for an interlock that prevent the simultaneous withdrawal of multiple control rods, individual control rod speed and control rod worth must be maintained so that the maximum reactivity addition rate of +0.08% Δk/k/second (+$0.11/second), as required by Technical Specification 3.2.3, Reactivity Insertion Rate, is not exceeded. Technical Specifications require verification that SDM and control rod worth are within limits annually and following significant changes to the core or any control rod.

In the core configuration as of 2004, the reactivity worth of each control rod is as follows: the regulating control rod worth is approximately 1.10 % Δk/k; the shim control rod worth is approximately 2.40 % Δk/k; and, the safety control rod worth is approximately 3.41 % Δk/k.

(2) Control Rod Guide Tubes

A control rod guide tube allows a control rod to be substituted for selected fuel elements or reflector elements and can be inserted into any of the [redacted] grid positions in the core. Each guide tube assembly is made of aluminum, anodized to increase resistance to wear and corrosion. The guide tubes have an outside diameter of [redacted] The guide tubes are supported by the lower grid plate and are fixed in place in the lower grid plate by a locking pin in the lower end fixture. The guide tubes include a large number of holes evenly distributed over the entire length of the tube. These guide tube holes allow water displaced by control rod movement inside the tube to flow into or out of the guide tube during rod movement.

(3) Control Rod Drives (CRD)

Each control rod is attached to a control rod drive mechanism that performs two functions: the CRD uses a motor in combination with a rack and pinion to withdraw and insert the control rods at a slow, constant rate for reactor power level adjustment; and, the CRD uses an electromagnet coupling between the CRD and the control rod to provide rapid reactor shutdown (scram) by decoupling the control rod from the CRD allowing the control rods to drop into the core following a scram signal or loss of power.

(a) Connecting Rod, Armature, Piston and Tubular Barrel

Each control rod is attached to a connecting rod that includes an armature and a dashpot piston (Figure 4.2-7). The control rod is connected to the CRD using an electromagnet that, when energized, holds the connecting rod armature in contact with the CRD. A scram signal or loss of power de-energizes the electromagnet allowing the connecting rod and control rod to drop toward the core.

The CRD has an extension tube barrel (Figure 4.2-8) that extends below the surface of the water and acts as a guide for the connecting rod. The extension tube barrel also has a mechanical stop to limit the downward travel of the control rod assembly. Part-way down the upper portion of the connecting rod (i.e., just below the armature) is a piston that travels within the barrel assembly. Since the upper portion of the barrel is well ventilated by large slotted openings, the piston moves freely in this range; but when the piston is within [redacted] of the bottom of its travel, its movement is restrained by the dashpot action of the graded vents in the lower end of the barrel. This dashpot action provides a smooth deceleration and reduces the impact when the rods are dropped during a scram.

(b) Drive Motor

The CRD assemblies for the three control rods are mounted on the I-beam portion of the reactor bridge support assembly. Each CRD consists of a motor and reduction gear driving a rack and pinion that is used to raise or lower the control rod. The CRD motor is non-synchronous, single phase AC motor that is instantly reversible. This motor is capable or inserting or withdrawing the safety control rod and the shim control rod at a rate of approximately 12 inches per minute and the regulating control rod at about 20 inches per minute. Electrical dynamic and static braking on the motor are used for fast stops. Micro-switches provided rod motion interlocks and provide input for control rod status lights on the control panel. USAR Chapter 7, Instrumentation and Control Systems, provides a more detailed description of the CRD motor and control circuits.

(c) CRD Limit Switches

Limit switches (micro-switches), mounted on the drive assembly, provide control rod position signals used for the CRD motor control circuits and operator console control rod status lights and enforce the control rod interlocks required by Technical Specifications. Specifically, these limits switches provide the following interlocks and indication:

(i) Prevent withdrawal of the safety rod unless all of the following conditions exist:
- (a) The master switch is in the ON position;
- (b) The safety system has been reset;
- (c) All four nuclear instrument channels are in the OPERATE mode; and
- (d) The neutron flux channel 1 count rate is ≤ 120 counts per minute unless bypassed when channel 2 exceeds a fixed setting of approximately 1×10^{-10} amps.

(ii) Prevent withdrawal of the shim rod and the regulating rod unless the safety rod is withdrawn to its upper limit.

(iii) Prevent simultaneous withdrawal of the shim rod and the regulating rod.

(iv) Provide operator console indication of the following:
(a) Control rod magnet in the UP position.
(b) Control rod magnet in the DOWN position.
(c) Control rod magnet in contact with the armature.

USAR Chapter 7, Instrumentation and Control Systems, provides a more detailed description of the operation of the CRD limit switches and interlocks.

(d) Control Rod Position Indication

A helipot connected to the pinion generates control rod position indication.

(e) Control Rods Drive (CRD) Operation

A pinion mounted on the CRD motor shaft drives a rack attached to the draw tube. Attached to the lower end of the draw tube is an electromagnet which, when energized, magnetically couples with an armature attached to the connecting rod and control rod, enabling CRD motion to be transferred to the control rod. Depending on the direction of rotation of the CRD motor, the control rod will be raised or lowered in the core within the limits established by the control rod position limit switches and control rod interlocks.

In the event of a power failure or scram signal, the control rod magnets are de-energized and the armature is released. The armature, connecting rod, and control rod drop by gravitational force, reinserting the neutron poison into the reactor core. The electromagnet carriage is automatically driven to the down position when the armature is released from the magnet.

Figure 4.2-1
ARRR Core Configuration

[redacted]

Figure 4.2-2
Typical ARRR Core Arrangement

[redacted]

STANDARD FUEL ELEMENT
SOURCE DUMMY ELEMENT
GRAPHITE DUMMY ELEMENT
REMOVABLE GLORY HOLE
CONTROL ROD

Figure 4.2-3
ARRR Aluminum Clad Fuel Element

[redacted]

Figure 4.2-4
ARRR Stainless Steel Clad Fuel Element

[redacted]

Figure 4.2-5
ARRR Core and Support Structure

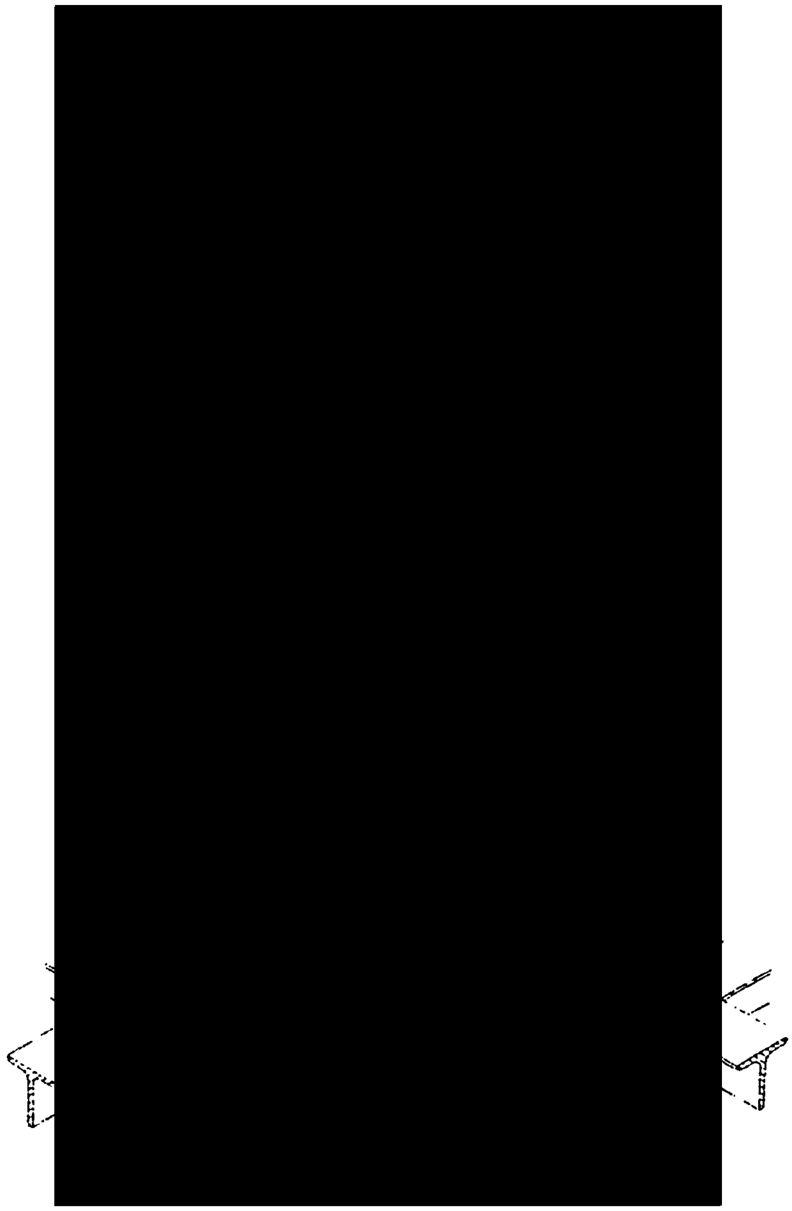


Figure 4.2-6
ARRR Neutron Source Holder

[redacted]

Figure 4.2-7
ARRR Control Rod Drive Connecting Rod and Barrel Assembly

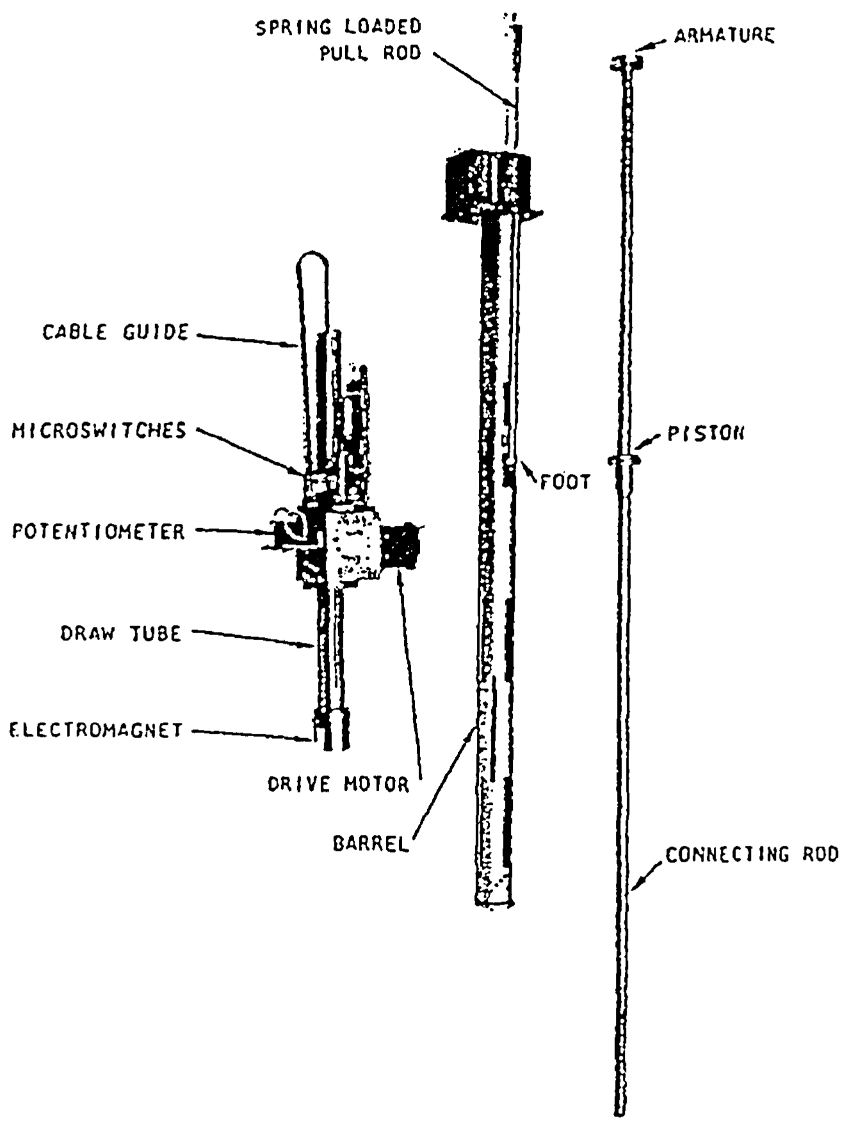
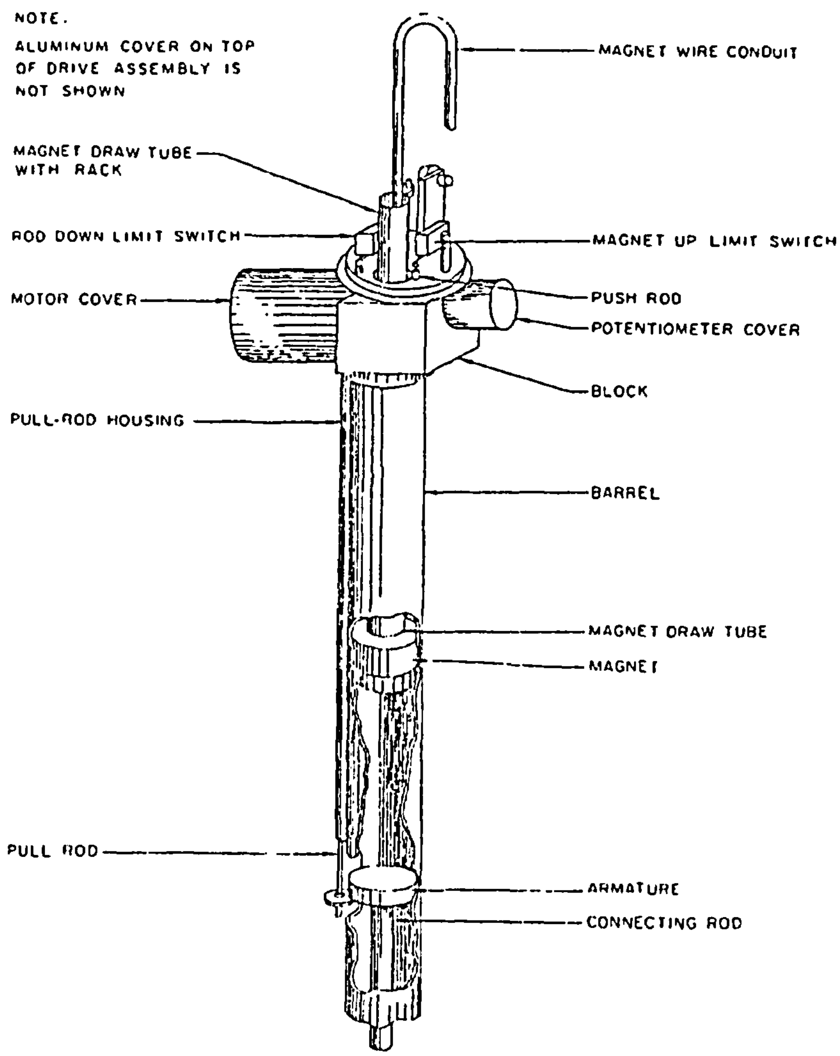


Figure 4.2-8
ARRR Control Rod Drive Assembly



4.3 REACTOR TANK

4.3.1 Reactor Tank Design

The reactor tank is an aluminum cylinder [redacted] in diameter and [redacted] deep that extends [redacted] below the floor level of the reactor building (Figures 4.2-1 and 4.3-1). The tank is open on the top with no openings below the water surface. The tank walls are [redacted] thick and the tank is set in concrete, which adds greatly to the mechanical integrity. The exterior of the tank is treated to minimize corrosion.

The reactor tank's concrete embedment includes one penetration consisting of a [redacted] outside diameter pipe about [redacted] that butts up against the outside of the reactor tank on the center line of the core. However, the tank wall is not cut open at this location. The pipe sleeve is provided so that a horizontal beam port may be installed at a later date, if desired, without having to break through the concrete around the tank. An amendment to the ARRR operating license would be required before installation of the beam port. This structure makes a rupture of the reactor tank very unlikely because it is supported on the bottom and sides by reinforced concrete and the tank is open to the atmosphere and cannot be pressurized.

The tank is surrounded by a concrete-lined trench, approximately [redacted] wide by [redacted] deep, to permit experimental and control cables and pipes to be located under the floor and thus protected from physical damage. The trench is partially covered by steel cover-plates. Radial trenches extend from the reactor tank in three directions. One radial trench extends into the control room and houses the reactor control cabling. A second trench extends to the east side of the building to provide under floor routing of the demineralizer and cooling water loops. The third radial trench extends to the south and houses a fan that cools the control rod magnets. A drain is provided in the trenches and, in addition, moisture-sensitive cables and equipment are supported off the trench floor.

4.3.2 Reactor Tank Requirements

Reactor tank water is used as the reactor core moderator, as the principal [redacted], and for reactor core cooling. Technical Specifications require that tank level is >16 feet above the top of the active core whenever fuel is in the pool and the reactor is not in the secured condition. This limit ensures adequate shielding for personnel working above the reactor during reactor operation. The reactor tank water level is normally maintained approximately at floor level which is [redacted] above the top of the core. This level corresponds to a water volume of approximately 13,000 gallons.

Rupture of the reactor tank is very unlikely because it is supported on the bottom and sides by reinforced concrete and the tank is open to the atmosphere and cannot be pressurized. Leaks can occur in the primary coolant system, or in the demineralizer loop. When leaks occur, the liquid drains to the reactor trench and then to the liquid waste hold-up tanks. In order to minimize waste volume, a moisture detector is installed in the trench. This detector automatically shuts-down both the primary and demineralizer pumps whenever water flows to the trench. This automatic system must be manually reset [redacted]

Reactor tank water quality is maintained within the Technical Specification limits for pH and conductivity by the demineralizer system. A conductivity meter provides continuous readout to the control room. The demineralizer system provides a convenient location to monitor radioactivity of the primary water. The water monitor provides a continuous readout to the control room and alarms at the reactor control console. Requirements that pH be maintained within Technical Specification limits are designed to minimize corrosion and the potential of leaks.

Reactor tank water temperature is maintained within the Technical Specification limits by the primary cooling loop. Temperature is maintained below 130 °F to prevent damage to ion exchanger resin in the demineralizer system. A temperature switch [redacted] initiates a scram prior to temperature reaching this level.

USAR Chapter 7, Instrumentation and Control Systems, describes the equipment used for monitoring reactor tank water quality and temperature including the associated trips and alarms.

4.3.3 Reactor Bridge Support Structure

The reactor bridge provides the structure above the reactor tank that mounts the control rods and glory hole assembly and provides a working platform [redacted]. The top edge of the reactor tank has a flange that supports the reactor bridge (Figure 4.1-1) and adds stiffening to the reactor tank wall. At each point where the bridge structure rests on the top flange, a vertical structural member is welded to the tank wall and extends into the concrete liner. The load of the bridge is thus transmitted directly into the concrete liner rather than being supported by the tank wall.

The reactor bridge itself is a support structure of [redacted] I-beams. Two of these beams support the [redacted] aluminum control rod drive mounting plate. This plate locates the control rods and glory hole. The reactor bridge is secured at one position on the tank flange. Leveling shims are used to provide accurate alignment between the bridge and the core support (grid) plate. A loading of [redacted], in addition to the normal weight of the control rods and equipment, will result in a maximum I-beam deflection of [redacted]. Therefore, the reactor bridge provides adequate support to ensure proper alignment of the control rod drive mechanisms. Different plates can be used if relocation of the control rods or the glory hole is desired. Some variations are possible with the existing mounting plate by using alternate mounting holes.

The area above the bridge structural members that is not covered by the mounting plate is covered with closed-tread plate. This plate provides a working platform giving access to the control rods and instrument tubes for maintenance.

4.4 BIOLOGICAL SHIELD

The reactor tank structure and reactor tank water provide the primary biological shield. The sunken reactor tank is surrounded by a concrete bulk shield structure and soil. Biological shielding above the reactor core is provided primarily by the required minimum of 16 feet of water above the top of the core. Additionally, the top of the reactor tank is also shielded by a combination of concrete blocks [redacted] thick and wooden (fir) beams [redacted] thick. This combination of structures provides radiation shielding for Aerotest personnel as well as excellent protection for the reactor core against natural phenomena.

4.5 NUCLEAR DESIGN

The ARRR nuclear design and accident analyses are based on the design features and analyses of the TRIGA fuel element as presented in References 4.7.4 through 4.7.10. These analyses are applicable to the ARRR because the ARRR uses standard TRIGA fuel elements that are essentially identical to the fuel elements used in more than 50 TRIGA and TRIGA-fueled reactors that have been in operation since 1958. TRIGA fuel elements fall into three basic types: aluminum clad; stainless steel clad; and Fuel Lifetime Improvement Program (FLIP). The general design of the TRIGA fuel elements is the same irrespective of type. Additionally, because all TRIGA reactors use fuel elements with very similar design and construction, TRIGA reactors have limited variability in core arrangement and lattice spacing regardless of the rated thermal power of the reactor. As indicated in Reference 4.7.3, there is no impact on accident analyses between different TRIGA reactors as a result of the small variations in core arrangement and lattice spacing or the small variations in fuel element end-plug design, dimensions, or the inclusion of burnable poisons.

At initial criticality, the ARRR core included [redacted] aluminum clad TRIGA fuel elements. The design was intended to allow for more fuel elements to be added, as necessary, to compensate for fuel burn up. Because aluminum clad TRIGA fuel elements are no longer manufactured, most of the fuel elements added to the ARRR are stainless steel clad. As of 2004, the ARRR reactor contains [redacted] active fuel elements consisting of [redacted] aluminum clad elements and [redacted] stainless steel clad elements as shown in Figure 4.2-2. The ARRR does not use FLIP fuel.

USAR Chapter 13, Accident Analyses, provides a description of how the ARRR is conservatively bounded by nuclear design and accident analyses presented in References 4.7.1 and 4.7.3. This discussion includes the general characteristics of the various types of TRIGA fuel element, the ARRR's use of both aluminum and stainless steel clad TRIGA fuel elements, bases for limits on peak fuel temperature, number of TRIGA fuel elements used in the ARRR core, ARRR Technical Specification limits on rated thermal power and excess reactivity, and the ARRR operating history.

4.6 THERMAL-HYDRAULIC DESIGN

The ARRR is designed for natural convection cooling. Cooling of the reactor tank water is not required by Technical Specifications during reactor operation. As demonstrated during startup physics testing (Reference 4.7.2), the ARRR can operate at 250 kW without external cooling of the reactor tank water for more than 6 hours before exceeding the Technical Specification reactor water temperature limit of 130 °F when starting with an initial water temperature of approximately 70 °F. The Technical Specification limit for a maximum reactor water temperature of ≤ 130 °F and an automatic reactor scram prior to the temperature reaching this limit and on low water level in the reactor tank ensure loss of cooling during normal operation will never result in exceeding the Safety Limit that the temperature in any fuel element in the ARRR reactor shall not exceed 500 °C.

Loss of coolant accidents for the ARRR are bounded by the instantaneous loss of all cooling water. Reference 4.7.3 summarizes the results of a loss of coolant analysis for the Reed College TRIGA reactor, a typical TRIGA reactor with aluminum clad elements. The results indicated that the maximum fuel temperature would be less than 150°C after the infinite operation at 250 kW which was terminated by the instantaneous loss of water. USAR Chapter 13, Accident Analyses, provides a description of the ARRR response to an instantaneous loss of all cooling water and a justification of why the results in Reference 4.7.3 very conservatively bound the results expected at the ARRR.

4.7 REFERENCES AND BIBLIOGRAPHY:

4.7.1 "Aerojet-General Nucleonics Industrial Reactor (AGNIR): Hazards Summary Report (AN-1193)," R. L. Newacheck, Project Engineer et al, September 1964.

4.7.2 "Aerojet-General Nucleonics Industrial Reactor (AGNIR): Reactor Physics Tests (AN-1527)," R. L. Tomlinson, August 1966.

4.7.3 NUREG/CR 2387, "Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors," S.C. Hawley and R.L. Kathren, US Nuclear Regulatory Commission, April 1982.

4.7.4 "TRIGA Mark III Reactor Description (GA-4339)," General Atomic Division of General Dynamics, December 1963.

4.7.5 "Hazards Report for the 250 kW TRIGA Mark II Reactor (GA-2025)," General Atomic Division of General Dynamics, August 1961.

4.7.6 "TRIGA Reactor Types and General Information (GEN-9)," General Atomic, September 1964.

4.7.7 "TRIGA Mark II Reactor General Specifications and Description (GA-2627)," General Atomic Division of General Dynamics, March 1964.

4.7.8 "Metallurgy of TRIGA Fuel Elements (GA-1949)," General Atomic Division of General Dynamics, January 4, 1961.

4.7.9 "Technical foundations of TRIGA (GA-471)," General Atomic Division of General Dynamics, August 27, 1958.

4.7.10 "Transient Behavior of TRIGA – A Zirconium-Hydride Moderated Reactor (GA-757)," General Atomic Division of General Dynamics, June 12, 1959.

5.0 REACTOR COOLANT SYSTEMS

The ARRR cooling system is comprised of five basic parts:

1. The reactor tank water provides natural circulation convective cooling of the reactor core, acts as neutron moderator and reflector, and provides neutron and gamma shielding.

2. The primary cooling loop provides forced circulation for mixing of the reactor tank water and, when required for reactor water tank temperature control, removes heat from the reactor tank water via a heat exchanger to the secondary cooling loop.

3. The secondary cooling loop, when required for reactor water tank temperature control, removes heat from the primary loop heat exchanger to the environment via the main or auxiliary evaporative cooling tower.

4. The demineralizer system maintains purity of the reactor tank water, provides forced circulation of the reactor tank water for mixing, and provides a mechanism of monitoring representative samples of reactor tank water for radioactivity.

5. The reactor tank water make-up system provides demineralized water to maintain the level of the reactor tank water.

5.1 REACTOR TANK WATER

The reactor tank is an aluminum cylinder [redacted] in diameter and [redacted] deep (Figures 4.2-1 and 4.3-1) that is open on the top with no openings below the water surface. When filled to its normal operating level of [redacted] (i.e., [redacted] above the top of active fuel), the reactor tank contains about [redacted] gallons of water. Physical features of the reactor tank are described in USAR Chapter 4, Reactor.

The reactor tank water provides natural circulation convective cooling of the reactor core, acts as neutron moderator and reflector, and provides neutron and gamma shielding. When the reactor fuel is producing heat, the resulting temperature differential between the water in the core region and water above the level of the core causes natural circulation through the core. Water heated by the core rises and leaves the core region through the gaps between the circular holes in the upper grid plate and the triangular or fin shaped sections of the fuel elements. Cooler water enters the core through holes in the bottom grid plate. Large openings in the shroud located between the grid plates allow additional flow into and out of the core region.

Cooling of the reactor tank water is not required by Technical Specifications during reactor operation. As demonstrated during startup physics testing (Reference 5.7.2), the ARRR can operate at 250 kW without external cooling of the reactor tank water for more than 6 hours before exceeding the Technical Specification reactor water temperature limit of 130 °F when starting with an initial water temperature of approximately 70 °F (Figure 5-1). Therefore, Technical Specifications do not require operation of the primary or secondary cooling loops as long as reactor water temperature is maintained within the Technical Specification limit. Although licensed to operate at a rated thermal power (RTP) of 250 kW, ARRR has been operating at 180 kW or below since 1992 and at 150 kW or below since 2000. These lower steady state power levels provide convenient exposure times for radiography while minimizing personnel exposure in compliance with the ALARA program.

An instantaneous loss of all reactor core cooling due to a reactor tank rupture is considered very unlikely because the reactor tank is supported on the bottom and sides by reinforced concrete and the tank is open to the atmosphere and cannot be pressurized. However, as described in USAR Chapter 13, Accident Analyses, even an instantaneous loss of all cooling water will not result in fuel temperatures that could cause the cladding to melt or a phase change in the zirconium hydride in the fuel in either the aluminum or stainless steel clad fuel. This analysis does not assume any emergency core cooling following the loss of coolant accident.

A significant loss of reactor cooling water through the primary cooling loop is prevented by the location of the suction inlet [redacted] below the normal reactor water tank level. There are no penetrations of the reactor tank below the normal operating water level and lines that access the lower portion of the reactor tank use siphon breaks to prevent inadvertent draining of the tank. Although the potential exists for a significant loss of water from the reactor water tank during an earthquake due to sloshing, the loss of water inventory from the [redacted] diameter tank is not expected to uncover the core which is positioned more than [redacted] below the upper rim of the reactor water tank.

As explained in USAR Chapter 7, Instrumentation and Control Systems, an automatic reactor scram on "Low Pool Water Level" is initiated when the reactor tank level is below normal but still ≥ 16 feet above the top of the core. This protection ensures that a significant loss of water inventory is unlikely and ensures that the reactor will be shut down prior to any significant loss of water inventory. Additionally, the reactor scram prior to the loss of water inventory will ensure that the Safety Limit, the temperature in any fuel element shall not exceed 500 °C, is not violated.

A loss of reactor cooling resulting in high reactor water temperature is prevented by Technical Specifications that limit maximum reactor water temperature to ≤ 130 °F and that require an automatic reactor scram prior to reactor water temperature exceeding this limit. The Technical Specification limit of ≤ 130 °F for reactor water temperature is intended to minimize thermal degradation of the ion exchange resin in the demineralizer system and is not required for reactor protection.

As described in USAR Chapter 7, Instrumentation and Control Systems, reactor water temperature is monitored by a thermocouple which indicates the pool temperature in the control room and a separate temperature switch located in the pool that is used to initiate the reactor scram on a Reactor Water Temperature – High signal. The Reactor Water Temperature – High trip set point is currently set about 125 °F to allow a sizable margin to the Technical Specification limit for both the water temperature and the automatic scram. The reactor water temperature detectors are located under the bridge near the top of the pool.

The reactor tank water provides neutron moderation and reflection. Reactors that use water as a moderator are typically designed to be under moderated to provide a negative temperature coefficient of reactivity. Maintaining the reactor tank water temperature ≥ 60 °F, as required by Technical Specifications when the reactor is not in the reactor secured condition, satisfies an assumption that ensures the bath temperature coefficient of reactivity remains negative.

The reactor tank water provides neutron and gamma shielding. Maintaining the reactor tank water level >16 feet above the top of the active fuel (i.e., approximately [redacted]), as required by Technical Specifications when the reactor is not in the reactor secured condition, provides shielding of the reactor core which reduces personnel exposure.

5.2 PRIMARY COOLING LOOP

The reactor tank water can be cooled, when necessary, by operation of the primary cooling loop (Figure 5-2) which is a closed loop that uses a pump to circulate reactor tank water through a shell and tube heat exchanger that is cooled by the secondary cooling loop. The primary cooling loop takes suction near the top of the reactor pool through a check valve that prevents loss of prime when the primary pump is not operating. The water flows through a three inch stainless steel pipe to the primary pump located in the heat exchanger building. A three phase, 220 Vac, five horsepower pump rated at 171 gallons per minute provides flow through the shell side of the heat exchanger. The water is returned to the pool through a [redacted] stainless steel pipe. The outlet of the return is at the bottom of the reactor tank where it is discharged in a circumferential direction to provide a swirling effect. Thermometers and pressure gages measure the inlet and outlet temperatures and pressures of the heat exchanger to allow performance of the heat exchanger and pump to be monitored locally.

As demonstrated during startup physics testing (Reference 5.7.2), the heat exchanger (853,000 BTU per hour (250 kW) at rated conditions) is capable of reducing reactor water temperature while the reactor continues to operate at the 250 kW licensed rated thermal power when starting from the reactor water upper temperature limit (Figure 5-1).

A significant loss of reactor cooling water through the primary cooling loop is prevented by the location of the suction inlet just one foot below the normal reactor water tank level. This prevents pumping water out of the reactor tank if the tank level falls below 16 feet above the active core.

All of the primary cooling loop piping connections are located in the reactor tank trench or in the heat exchanger building where drainage is provided to the waste storage tanks. Any leakage of the pump packing gland or fittings is collected in the liquid waste holding tanks. Leakage detection is provided by a moisture sensor located in the trench that will shut off the primary cooling loop pump and the demineralizer system pump and activate the demineralizer system low flow alarm in the control room if leakage is detected.

As stated earlier, cooling of the reactor tank water is not required by Technical Specifications during reactor operation except as needed to maintain reactor water temperature within Technical Specification limits. Primary cooling loop flow can be initiated manually or cycled automatically to control reactor water temperature depending on the position of the primary cooling loop control switch. When the control switch is placed in "Automatic," a temperature sensor (located in the demineralized loop flow path) will start the primary cooling loop pump when the water temperature rises above a preset temperature and stop the primary cooling loop pump when the reactor water cools to a lower preset temperature. When the control switch is placed in "Hand" position, the primary cooling loop pump operates continuously. The OFF position on the cooling switch prevents the operation of the system under all conditions.

When the reactor is operating, the primary cooling pump is operated continuously as soon as reactor power reaches the point where Neutron Flux Channel 1 is bypassed. Continuous circulation of the reactor water is maintained to increase the time required for induced radioactivity, particularly ^{16}N, to reach the top of the tank as explained later in this section. When the reactor is shut down, the reactor water temperature switch is used to turn off the primary cooling pump when the reactor tank water cools to a preset temperature.

5.3 SECONDARY COOLING LOOP

The secondary cooling loop is an open loop system used to remove heat from the primary cooling loop heat exchanger and serves to isolate the reactor water from the environment. The secondary cooling loop water removes the heat from the primary cooling loop heat exchanger and transports it to the top of the main cooling tower located outside the reactor building as shown in Figure 5-2. The auxiliary cooling tower has been retired in-place and is isolated from the secondary cooling loop by closed valves and a blank flange.

A three phase, 220 Vac, five horsepower pump rated at 175 gallons per minute provides flow through the tube side of the primary cooling loop heat exchanger. The secondary cooling pump is activated whenever the primary pump is on. When the secondary cooling pump is on, the secondary cooling loop water is then evenly distributed to cascade down over diffusers in the cooling tower where forced air is drawn through the water. A three phase, 220 Vac, two horsepower fan draws air into one side of the cooling tower and vents it out the opposite side. A temperature switch in the cooling tower pool activates the fan when the temperature is above a preset point (approximately 70 °F). The cooling tower fan operates only when the pumps are on but will cycle as required by the water temperature sensor in the basin. Note that certain tests may require that the circulation or heat removal not be functional. For example, during a calorimetric power calibration, the secondary cooling pump is locked out to prevent heat removal and the primary cooling pump is left on to retain good mixing in the pool.

The main cooling tower sump is treated to minimize scale and is operated with continuous blow down through a centrifugal dirt separator to reduce total dissolved solids. Additionally, the sump is supplied with H_2O_2 to inhibit water borne biological growth and control the acidity.

5.4 DEMINERALIZER SYSTEM

The ARRR includes a demineralizer system that maintains the reactor water purity (Figure 5-3). Additionally, the demineralizer system is used to provide a re-circulating sampling system that provides continuous indication and alarms for reactor water radioactivity. The combination of demineralizer and monitoring provides the following:

(1) Minimization of corrosion of the fuel element cladding and the reactor water tank;

(2) Minimization of dissolved materials subject to neutron activation and,

(3) Early detection of the release of radioactive materials to the coolant (e.g., fuel clad defects or defects in experiments) before such releases become significant.

The demineralizer system takes suction from the reactor tank through a skimmer at the water surface to remove floating material from the water surface. This design also ensures that a leak in the loop could not decrease the level of the reactor tank more than a few inches before the pump would lose prime.

The reactor water conductivity monitor is located near the top of the pool, under the bridge, and provides continuous indication at the reactor control cabinet. Elevated conductivity levels in the reactor water indicate the presence of corrosion products and promote more corrosion. Experience with water quality control at many reactor facilities, including the ARRR, has shown that maintaining water conductivity and pH within limits provides acceptable corrosion control. Maintaining low levels of dissolved electrolytes also reduces the amount of induced radioactivity which in turn decreases the exposure of personnel to radiation.

The demineralizer system flow is exposed to the temperature switch that activates the primary and secondary cooling pumps when the pool temperature reaches the upper temperature set point discussed earlier. The temperature monitor is located in the suction line to minimize time for cooling after the water leaves the reactor water tank. The demineralizer system flow is exposed to a radiation monitor that provides indication in the control room of the reactor water radioactivity level and provides annunciation and an alarm if the radioactivity level exceeds the set point specified in Technical Specifications. The radiation monitor is located in the suction line so that the radioactivity monitor is located upstream from the particle filter and mixed bed demineralizer. The radioactivity monitor is located in the [redacted] and is shielded to ensure it is isolated from any direct radiation field. The range of the reactor water radiation monitor is 0.1 to 100 mrem/hour.

The demineralizer system pump is a single phase, 220 Vac, one horsepower pump rated at 20 gallons per minute. Downstream of the pump, water flows through a 100 micron string filter followed by a two cubic foot capacity mixed bed demineralizer with a 20 gpm flow rate capacity. Large particles are removed with a filter prior to the inlet to the demineralizer to minimize clogging the resin and extend the lifetime. The demineralizer can be bypassed for maintenance while the demineralizer system remains in operation.

Operation of the demineralizer is sufficient to maintain a reactor water conductivity and pH within the Technical Specifications limits of ≤ 5 μmho/cm and ≤ 7.5, respectively. With these conditions, the reactor tank water is very clear and allows an excellent view of the reactor components in the pool.

The demineralizer system uses a flow switch to monitor coolant flow. This flow switch is the last device in the loop and therefore verifies that the water is returning to the reactor water tank. Leakage upstream of the flow meter would reduce the flow and send an indication of "Flow-Low" to the reactor control console. As required by Technical specifications, an alarm signal is generated prior to the demineralizer system flow rate falling below 4 gallons per minute which is indicative of a pump failure or a clogged demineralizer. The demineralizer flow meter is a Proteus Model #155 with a minimum capacity of 4 gallons per minute.

The demineralizer system must be in operation whenever the reactor is operating at any power level because the radioactivity monitor is located in the demineralizer loop. Typically, the demineralizer system is operated continuously, regardless of reactor operational status. However, there is no Technical Specification requirement that the system operate when the reactor is in the reactor secured condition.

Continuous operation of the demineralizer system after reactor shutdown provides automatic temperature control for the cooling system and allows the demineralizer to continue purifying the water while the reactor is not operating. A rotary vane flow switch provides an alarm if the demineralizer system flow rate falls below the minimum required by Technical Specifications.

5.5 PRIMARY COOLANT MAKEUP WATER SYSTEM

Water evaporating from the surface of the pool is replaced by a simple water makeup system. Typical evaporation rates for the reactor water tank were determined during the startup physics tests (Reference 5.7.2). The reactor tank water evaporation rate was found to be about 0.5 gallons per hour with a pool temperature of 70°F and approximately 5 gallons per hour with a pool temperature of 125 °F.

The water used for makeup to the reactor water tank is from the municipal water supply, East Bay water, which supplies the ARRR site with relatively pure water. The reactor water tank makeup water is passed through a de-ionizer cartridge prior to entering the pool. There is no license requirement to de-ionize the makeup water; however, use of the de-ionizer reduces the pool water radioactivity and greatly increases the time between changes of the resin in the demineralizer in the demineralizer system.

The pool is maintained at a constant level, ± [redacted] inches, by two float valves which drip makeup water into the reactor water tank. The two float valves are piped in series such that either valve will shut off the flow independently in case one valve should fail. The makeup water discharge spout is about [redacted] inches higher than the reactor tank water level and there is a drain hole through the reactor tank wall above the normal water level but below the level of the spout. These features eliminate any potential for backflow from the reactor water tank to the municipal water supply.

5.6 POOL WATER RADIOACTIVITY

5.6.1 Nitrogen-16 Control

Nitrogen 16 (^{16}N), is a gamma emitting isotope with a 7.1 second half-life, that is produced during reactor operation by the fast neutron irradiation of oxygen in the reactor tank water (^{16}O (n, p) ^{16}N). Radiation levels, and thus personnel exposure, can be reduced by increasing the amount of time that it takes for induced radioactivity, particularly ^{16}N, to reach the top of the pool. ARRR experience indicates that operating the primary cooling loop increases the transit time for ^{16}N to reach the water surface by removing warm water from near the surface of the tank and returning the cooled water to near the bottom of the tank in a swirling motion. The injection of primary cooling loop water into the bottom of the tank in a swirling motion breaks up the thermal plume that rises from the core under quiescent conditions. This thermal plume would otherwise provide a relatively rapid transport of warmed water from the center of the core to the pool surface. Because ^{16}N has a 7.1 second half-life, the increase in transit time is sufficient to decreases the amount of induced radioactivity, particularly ^{16}N, released through the surface of the water of the reactor tank. The difference in radiation levels measured by the air monitor, with its intake just above the water surface, with and without the primary cooling loop in operation confirms the effectiveness of the primary cooling loop in reducing radiation levels from decay of ^{16}N above the reactor. To reduce radiation exposure to ALARA, ARRR operates the primary cooling loop continuously as soon as the reactor exceeds the point of adding heat (i.e., a few watts power) whether or not the reactor water temperature is above the automatic temperature control setting.

5.6.2 Argon Activation

Another potential source of radiation exposure above the reactor is from activation of argon dissolved in the reactor water. ARRR operating experience is consistent with the results of the analysis in Appendix E of Reference 5.7.1 that radiation exposure from activation of argon dissolved in the reactor water does not pose a significant risk to operating personnel. Control of argon activation during experiments, particularly during the use of a glory hole or the vertical tube, is addressed in USAR 10, Experimental Facilities and Explosives.

Figure 5-1
Reactor Water Tank Heating and Cooling
Temperature versus Time
(Reference 5.7.2)

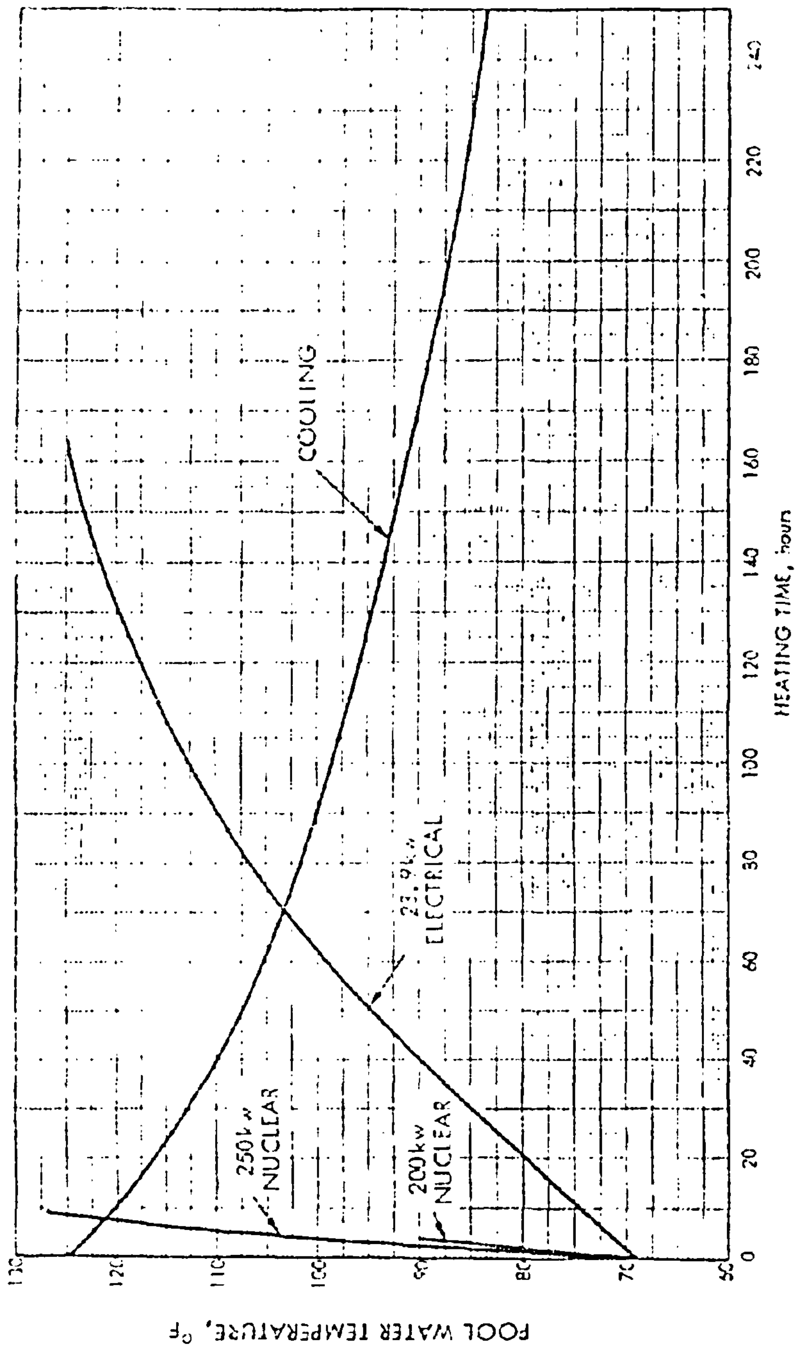


Figure 5-2
Primary and Secondary Cooling Loops

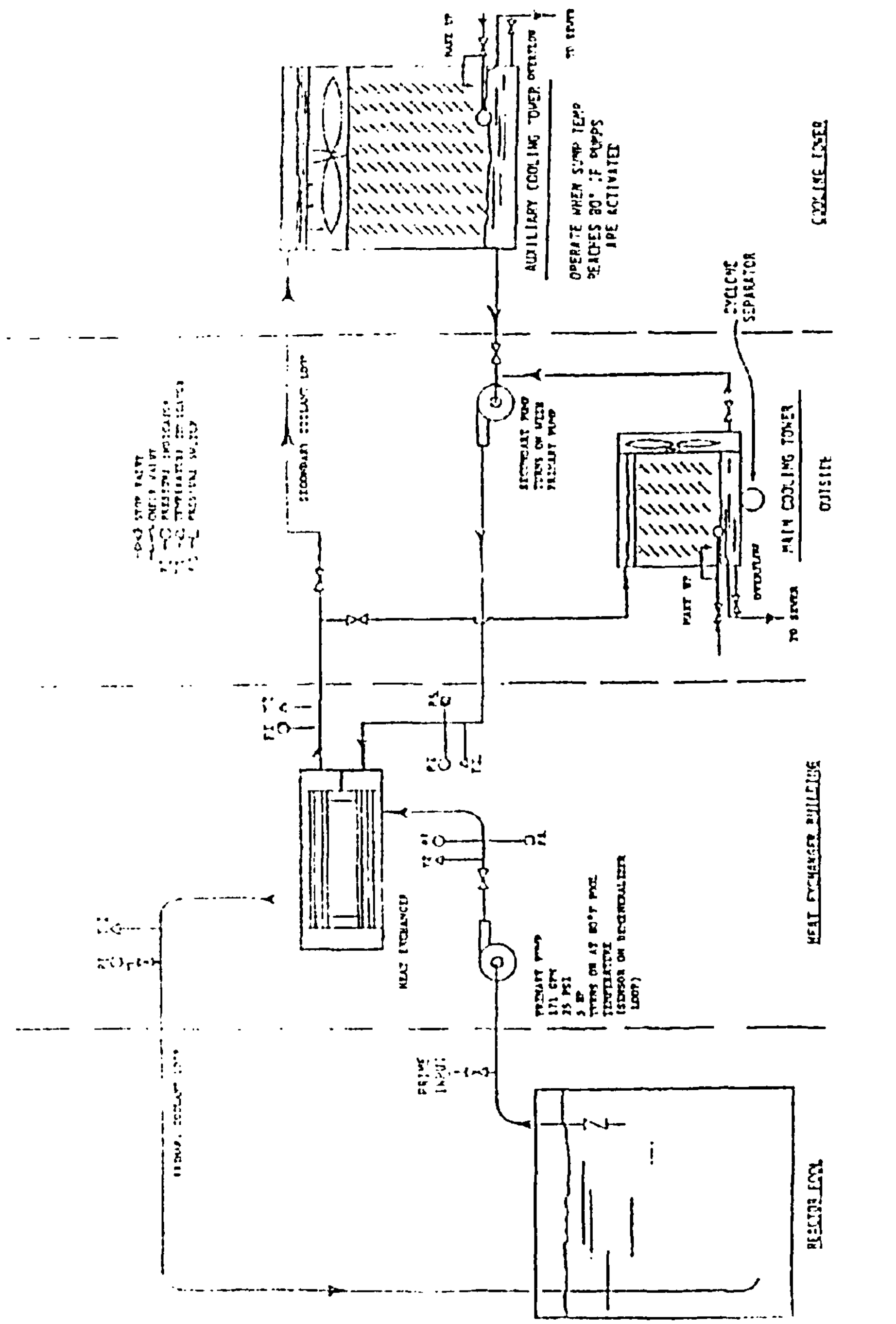
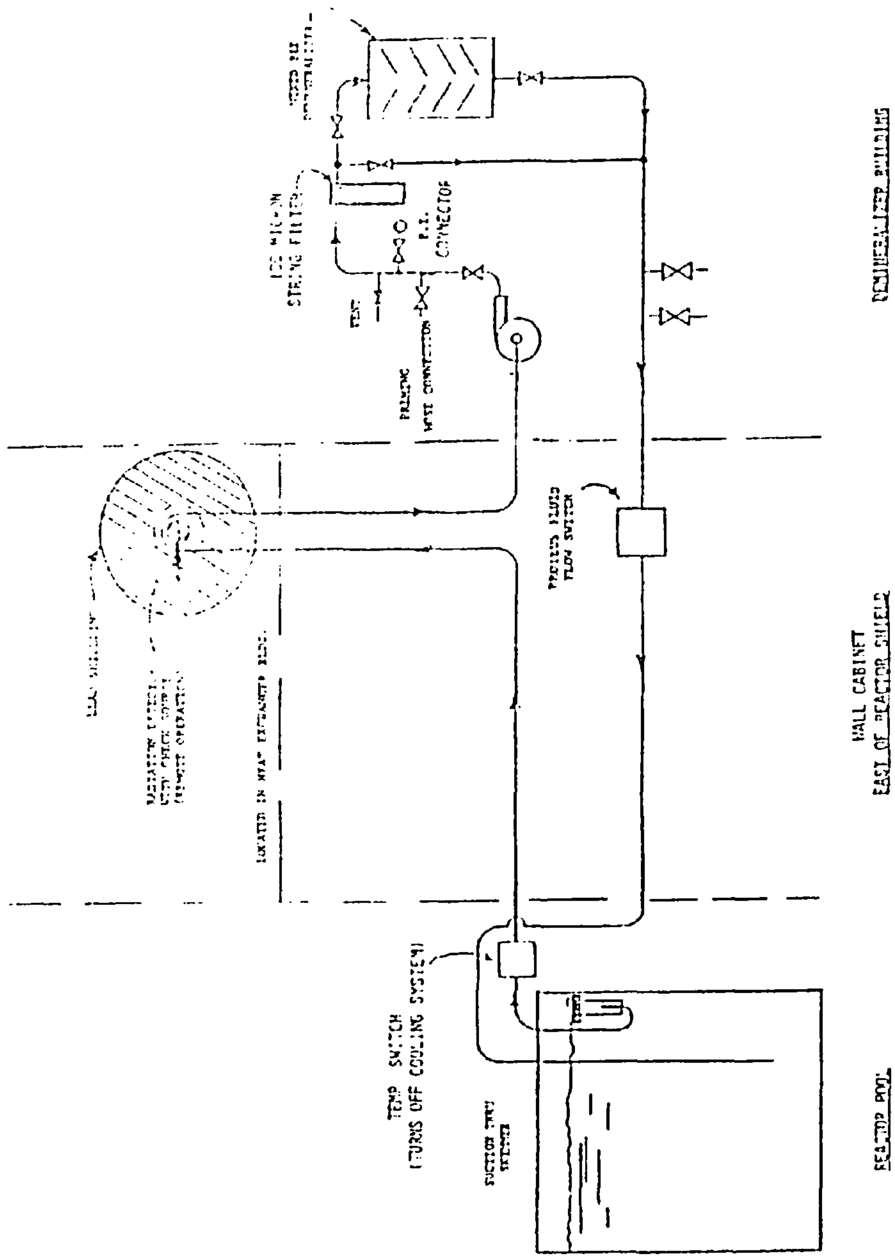


Figure 5-3
Demineralizer System



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6.0 ENGINEERED SAFETY FEATURES

Engineered Safety Features (ESF) are active or passive features designed to mitigate the consequences of accidents and to keep radiological exposures to the public, the facility staff, and the environment within acceptable values.

The accident analysis for the ARRR described in USAR Chapter 13, Accident Analyses, is based on the ARRR Hazards Summary Report (Reference 3.3.1) and NUREG/CR 2387 (Reference 3.3.2), relies entirely on the design of the TRIGA fuel elements for the prevention and mitigation of any accidents. The design of the TRIGA fuel and Technical Specification imposed operating limits allow the ARRR to respond to an uncontrolled rod withdrawal, step addition of the maximum available excess reactivity, complete loss of all reactor coolant, and all other credible events with no hazard to the public without reliance on any engineered safety features. No specific structures, systems, or components (SSC) are assumed to be operable for the mitigation of any accident or the protection of the public health and safety.

Technical Specification limits for rated thermal power (250 kW) and the amount of excess reactivity available in the reactor, with or without experiments installed, ensure that the ARRR is operated within the assumptions used to determine that TRIGA fuel elements have the ability to tolerate credible events without damage. Additionally, the Reactor Protection System and Reactor Monitoring Systems initiate either automatic or operator initiated reactor shutdown at conservative levels in the event of high reactor power level, short reactor period, seismic disturbance, high reactor water temperature, low reactor water tank level, high reactor water conductivity, or high reactor water radioactivity.

6.1 CONTAINMENT AND CONFINEMENT

The effectiveness of the TRIGA reactor fuel and fuel elements at minimizing the consequences of reactor accidents is demonstrated in the accident analysis (USAR Chapter 13, Accident Analyses) which does not assume that the ARRR building acts as either a containment or confinement to mitigate the release of radioactivity following a reactor accident. In fact, the analysis of the maximum fission product release following a fuel element failure (Appendix C of Reference 3.3.1) assumes 0.5 to 2.0 air changes per hour in the building and that all fission product gases leak out of the building within one hour. This assumption is very conservative because the original building included large evaporative coolers which purged the building with up to 9,000 cfm of air. These coolers were removed in 1970.

As described in USAR Chapter 9, Auxiliary Systems, the ARRR building ventilation system is designed to maintain pressure differentials between selected areas within the ARRR building to achieve two objectives: confinement of airborne radioactive material to the reactor high bay area; and, exclusion of airborne radioactive material from inhabited areas, especially the control room.

As described in USAR Section 9.1, confinement of airborne radioactive material to the reactor high bay area is enhanced by a ventilation system design that does not provide any makeup of outside air directly into the reactor high bay area. The doors and window to the control room that form part of the high bay area boundary are maintained closed except for momentary passage. Doors to the adjacent radiography setup room are closed by procedure during any radiological event. Areas adjacent to the high bay area are supplied with outside air and are at a pressure that is slightly positive relative to the high bay area. Any inleakage into the high bay area is directed out of the building through three gravity ventilators in the roof over the high bay area. This ventilation strategy prevents air from entering the reactor building through the reactor high bay area which minimizes the potential for spreading airborne radiation or contamination from the high bay area to other parts of the ARRR building.

During any radiological event that could spread contamination or airborne radiation within the building or release it to the environment, the reactor high bay ventilation system, chemical laboratory hood blower, and rest room vent fans are shut off to reduce the potential for spread of contamination and airborne radiation. Ventilation systems that maintain areas adjacent to the reactor high bay area at a pressure slightly higher than the high bay area are not shut off. Confinement of airborne radioactive material to either the ARRR building or the reactor high bay area following an accident is not an assumption used in the accident analysis.

Exclusion of airborne radioactive material from inhabited areas, especially the control room, is accomplished by maintaining the areas at a positive pressure relative to the reactor high bay area. The air flows throughout the facility have been designed so that the control room, the lunch room, the office complex, the rest rooms, and the machine shop are at positive pressure with respect to the reactor high bay area. The positive pressure is maintained by an oversized ventilation system with a high fraction of make-up air from the outside. Technical Specifications require that the control room is at a positive pressure with respect to the reactor high bay area whenever the reactor is not in a secured condition. Exclusion of airborne radioactive material from the control room following an accident is not assumed in the accident analysis because control room habitability is not assumed.

6.2 EMERGENCY CORE COOLING

The effectiveness of the TRIGA reactor fuel and fuel elements at minimizing the consequences of reactor accidents is described in the accident analysis. As described in USAR Chapter 13, Accident Analyses, even an instantaneous loss of all cooling water will not result in fuel temperatures that could cause the cladding to melt or a phase change in the zirconium hydride in the fuel in either the aluminum or stainless steel clad fuel. This analysis does not assume any emergency core cooling following the loss of coolant accident.

As described in USAR Chapter 4, Reactor Coolant Systems, the core is cooled by natural circulation convective cooling by the [redacted] of water in the reactor water tank. As demonstrated during startup physics testing (Reference 6.3.3), the ARRR can operate at 250 kW without external cooling of the reactor tank water for more than 6 hours before exceeding the TS reactor water temperature limit. Emergency core cooling is not assumed in the accident analysis for a loss of coolant flow accident.

6.3 REFERENCES

6.3.1 "Aerojet-General Nucleonics Industrial Reactor (AGNIR): Hazards Summary Report (AN-1193)," R. L. Newacheck, Project Engineer et al, September 1964.

6.3.2 NUREG/CR 2387, "Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors," S.C. Hawley and R.L. Kathren, US Nuclear Regulatory Commission, April 1982.

6.3.3 "Aerojet-General Nucleonics Industrial Reactor (AGNIR): Reactor Physics Tests (AN-1527)," R. L. Tomlinson, August 1966.

7.0 INSTRUMENTATION AND CONTROL SYSTEMS

The ARRR instrumentation and control (I&C) systems provide the following:

1. Continuous indication of important reactor operating parameters.
2. Control of insertion and withdrawal of control rods for reactor startup, shutdown, and control of reactor power level.
3. Automatic shutdown (scram) of the reactor if important operating limits are exceeded, if I&C electrical power is lost or if initiated by the operator.
4. Interlocks that enforce design assumptions regarding reactivity addition and personnel protection.
5. Indication of radiation levels and radioactive gaseous effluents in the reactor building with alarms if pre-set limits are exceeded.

These objectives are accomplished using four systems:

Rod Control System is used to control withdrawal and insertion of the control rods during reactor startup and shutdown and to control power level during normal operation. The rod control system also works in conjunction with reactor protection systems to provide rapid insertion of control rods if reactor protection system limits are exceeded, I&C power is lost, or if a reactor scram is manually initiated by the operator. The rod control system also works in conjunction with other I&C systems to enforce rod movement interlocks that limit the addition of reactivity consistent with design and accident analysis assumptions.

Reactor Protection System (RPS) is used to monitor important reactor operating conditions (e.g., reactor power level, reactor tank water level) and initiate automatic reactor shutdown via the rod control system if pre-set limits are exceeded or required conditions are not met.

Reactor Monitoring Systems (RMS) are used to monitor primary coolant conductivity, primary coolant cleanup (demineralizer) loop flow, reactor water radioactivity and bridge crane location and provide audible and visual indication to the operator if pre-set limits are exceeded.

Criticality Alarm, Radiation and Radioactive Gaseous Effluent Monitoring Systems are used to monitor area radiation levels and gaseous radiation levels in the area above the reactor and provide audible and visual indication to the operator if pre-set limits are exceeded.

The control and instrumentation (I&C) systems for the ARRR are similar to those used in other TRIGA reactors. The design objectives of the RPS were that the system would be fail-safe (i.e., de-energizing shall cause a scram) and that no single component failure or circuit fault could simultaneously disable both the automatic and manual scram circuits. ARRR protection and control systems are hardwired analog systems that use analog displays with the exception of the semi-conductor bi-stable trip cards used in the RPS and RMS. RPS Channel 1 Neutron Flux is a solid state system. The instruments normally monitored by an operator or that may require operator action are located in a [redacted]. Additional power supplies and process instrumentation chassis are located in a separate instrumentation rack placed [redacted].

Technical Specification Table 3.2-1, Reactor Protection System Instrumentation, Table 3.2-2, Reactor Monitoring System Instrumentation, and Table 3.5-1, Criticality Alarm, and Area and Effluent Radiation Monitoring Instrumentation, and the associated Bases list the I&C functions, minimum operability requirements, and set points for each of the I&C systems, with descriptions later in the chapter.

7.1 ROD CONTROL SYSTEM

ARRR reactivity control is achieved by the operation of three control rods (1 safety rod, 1 shim rod and 1 regulating rod) and their associated motor driven control rod drive (CRD) assemblies. The three CRDs and their associated control circuits have two primary functions: the CRD reactor control function (i.e., controlled withdrawal and insertion of the control rods for reactor startup, shutdown and changing reactor power level); and, the CRD emergency shutdown (Scram) function (i.e., rapid reactor shutdown by decoupling the control rod from the CRD allowing the control rods to drop into the core following an RPS scram signal or loss of power.

Each of the three control rod assemblies consist of a control rod, control rod guide tube and a control rod drive assembly (Figures 4.2-7 and 4.2-8). USAR Chapter 4 provides a description of each of the following: the control rods; the control rod guide tubes; control rod drives (CRD); the connecting rod, armature, piston and tubular barrel; drive motor; and CRD limit switches.

7.1.1 CRD Reactor Control Function

The reactor control function performed by the CRD uses a motor in combination with a rack and pinion to withdraw and insert the control rods at a slow, constant rate for reactor power level adjustment. As described in USAR Chapter 4, when the electromagnet on the CRD is energized, it is magnetically coupled to the connecting rod armature and the connecting rod and control rod move with the rack and pinion on the CRD. Depending on the direction of rotation of the CRD motor, the control rod will be raised or lowered in the core within the limits established by the control rod position limit switches and control rod interlocks. In the event of a power failure or scram signal, the control rod magnets are de-energized and the armature is released. The armature, connecting rod, and control rod drop by gravitational force, reinserting the neutron poison into the reactor core.

The CRD motor is a non-synchronous, singe phase AC motor that is instantly reversible. This motor is capable of inserting or withdrawing the safety control rod and the shim control rod at a rate of approximately 12 inches per minute and the regulating control rod at about 20 inches per minute.

[redacted] allow the operator to control the CRDs and provide indication of the status of the CRD and associated control rod. Limit switches (micro-switches), [redacted], provide control rod position signals used for the CRD motor control circuits and operator console control rod status lights. These limits switches also enforce the control rod interlocks required by Technical Specifications. [redacted]. One light indicates the CRD is at the down limit, a second indicates the CRD is at the up limit, and the third light is a split indicator: one half for CRD magnet contact status; and, the other half for CRD magnet current. When lighted, the [redacted] indicate current to the CRD magnet and that the magnet is in contact with the rod armature. When the [redacted] are depressed, the magnet current will be interrupted and the ON lights will be extinguished. If a drive is above the down limit, the rod will fall back into the core and the magnet contact light will be extinguished. When the armature is released from the magnet, the electromagnet carriage is automatically driven to the down position until it makes contact with the rod again. Releasing the button will close the magnet circuit, and magnet current will be reset.

Figure 7.1-1 shows a simplified CRD motor-control circuit. This circuit is designed for maximum reliability by minimizing the number of switch contacts required and eliminating the need for relays. The diagram shows that both CRD motor windings (for the reversible motor) will receive 115 Vac line power through the normally closed up (UP) and down (DN) pushbuttons. This feature provides the motor with dynamic braking. Depressing the UP button opens the line to one of the windings. This essentially permits the line current to flow only through the DN button to the other winding and will affect the phase-shifting capacitor. The phase difference at the motor windings causes the motor to rotate. Opposite motor rotation is obtained when the DN button is depressed (Reference 7.5.1).

Figure 7.1-2 is a schematic diagram combining the standard drive motor control circuit with the drive limit and indicating circuits. All switches are shown as they would be if the control drive and magnet were both fully down. Under these conditions, the magnet down (DOWN) lamp and the contact (CONT) lamp would both be on, whereas the magnet up (UP) lamp would be off. Reference 7.5.2 provides a more detailed description of the operation of the control rod drive limit switches and indication circuits. There are four interlocks enforced by the CRD reactor control function:

(1) All control rod cables must be plugged into the rear of the control chassis and connected to the control rods or no power can be applied to any of the rod drive motors.

(2) CRD Interlocks prevent withdrawal of the safety rod unless all of the following conditions exist:

    (a) The master switch is in the ON position;

    (b) The safety system has been reset;

(c) All four nuclear instrument channels are in the OPERATE mode; and,

(d) The neutron flux channel 1 count rate is ≤120 counts per minute unless bypassed when channel 2 exceeds a fixed setting of approximately 1 x 10^{-10} amps.

(3) CRD Interlocks prevent withdrawal of the shim rod and the regulating rod unless the safety rod is withdrawn to its upper limit. This interlock ensures that an RPS scram will insert sufficient reactivity to shut down the reactor.

(4) CRD Interlocks prevent simultaneous withdrawal of the shim rod and the regulating rod. This interlock ensures the maximum rate of reactivity addition will remain within the limits specified in the Technical Specifications.

7.1.2 CRD Emergency Shutdown (Scram) Function

The emergency shutdown (scram) function performed by the CRD is to provide rapid reactor shutdown (scram) by decoupling the control rod from the CRD allowing the control rods to drop into the core by gravity following an RPS scram signal or loss of power. Each control rod is attached to a connecting rod that includes an armature and a dashpot piston (Figure 4.2-7). The control rod is connected to the CRD using an electromagnet that, when energized, holds the connecting rod armature in contact with the CRD. The power supply to the CRD electromagnets is controlled by the scram bus which must be energized for power to be supplied to the CRD electromagnets. A scram signal or loss of power de-energizes the electromagnet allowing the connecting rod and control rod to drop toward the core. The electromagnet carriage is automatically driven to the down position when the armature is released from the magnet.

Figure 7.1-1
Simplified Control Rod Drive Motor-Control Circuit

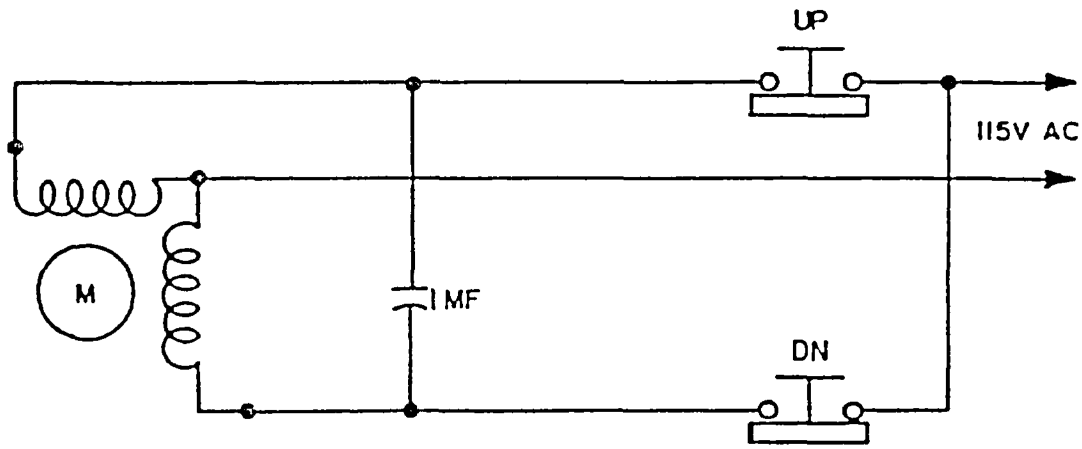
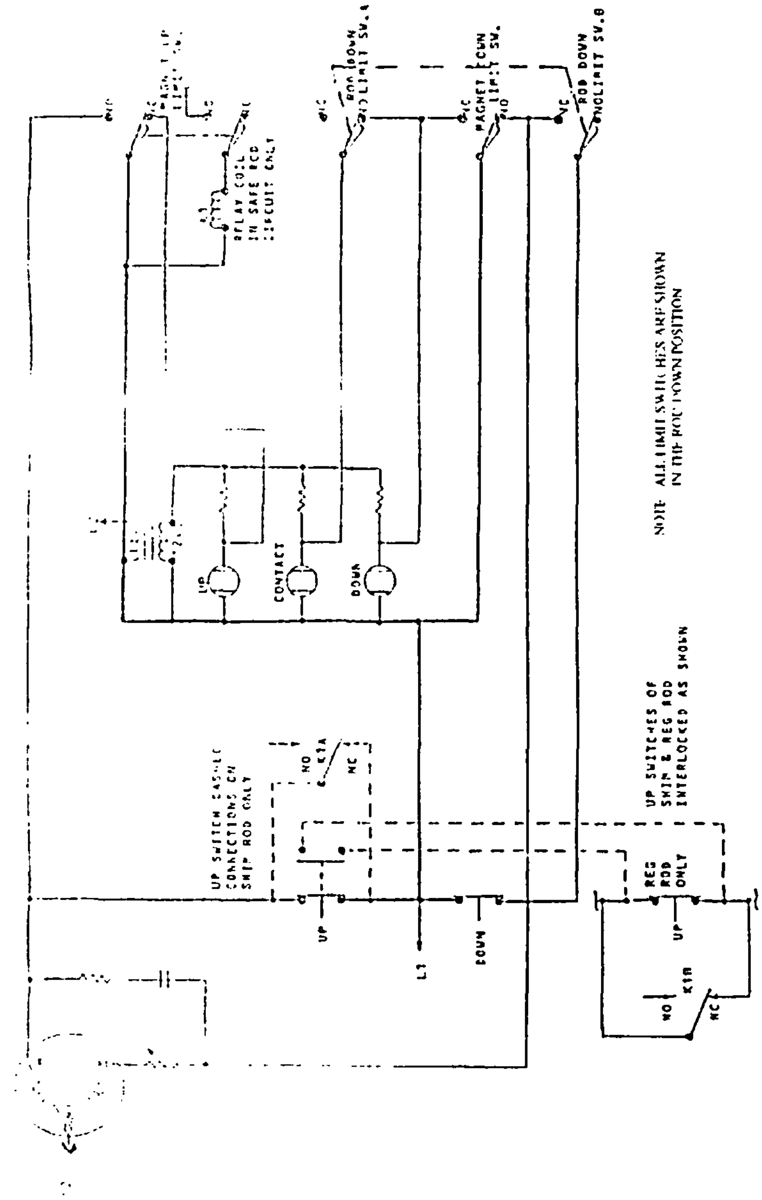


Figure 7.1-2
Control Rod Drive Motor Control Circuit
with the Drive Limit and Indicating Circuits



7.2 REACTOR PROTECTION SYSTEM (RPS)

The purpose of the RPS is to monitor important reactor operating conditions and initiate automatic reactor scram via the rod control system if pre-set limits are exceeded, or required conditions are not met, or if initiated by operator action. The RPS consists of sensing devices (four channels of nuclear instruments, a seismic monitor, and reactor tank level and temperature sensors), a 115 Vac power supply, high voltage power supplies for the nuclear instruments, amplifiers, associated circuits, semi-conductor trip cards, and relays that allow the scram bus (i.e., the circuit that supplies power to the CRD electromagnets) to remain energized if required conditions (i.e., interlock conditions) are met and pre-set limits are not exceeded. If any required condition is not met or pre-set limit exceeded, the scram bus is immediately de-energized which in turn de-energizes the CRD electromagnets and allows the control rods to drop into the reactor core. These RPS sensor signals are also used to provide continuous indication of reactor status and as inputs to the annunciator and alarm instrumentation.

As defined in the Technical Specifications, a channel is the combination of sensor, line, amplifier, and output device which are connected for the purpose of measuring the value of a parameter. In the case of the RPS, the output device is the relay that de-energizes the scram bus when the associated sensor does not meet the required limit or condition. Figure 7.2-1 shows a block diagram of the nuclear instrumentation portion of the reactor protection system. Figure 7.2-2 shows a logic diagram of the reactor protection system that includes the seismic monitor and reactor tank water level and temperature. Figure 7.2-3 shows a schematic diagram of the scram bus and CRD electro magnet circuit. Figure 7.2-4 shows the operating range of each of the nuclear instruments.

Each RPS channel or function includes a bi-stable trip card designed to give an output of either 12 volts or 0 volts depending on the output from the associated sensor. Depending upon the normal (i.e., reactor operating within specified limits) output received from the associated sensor (i.e., continual signal or no signal), the relays associated with that trip card are kept energized by proper choice of the trip card output connection. If the sensor output exceeds the specified limit, the bi-stable trip card output changes state and the relays associated with that trip card are de-energized which de-energizes the scram bus. The trip cards and relays are mounted separately so that removal of a trip card will automatically scram the reactor by releasing the relays, all of which are energized during normal operation. This assures fail-safe conditions for trip card removal and for electronic power supply failure as well. All scram signals remain in the tripped condition until they are manually reset after the scram condition has been cleared.

[redacted]

Requirements for periodic channel checks, channel tests, and channel calibrations for each of the RPS functions are specified in the Technical Specifications.

The fifteen channels that will initiate a reactor scram are divided into nine functions based on the sensor as follows:

Function 1.a. Channel 1 Neutron Flux: Short Period.
Function 1.b. Channel 1 Neutron Flux: Low Source Level.
Function 2.a. Channel 2 Neutron Flux: Short Period.
Function 2.b. Channel 2 Neutron Flux: Loss of Instrument Power.
Function 2.c. Channel 2 Neutron Flux: Low Detector Voltage.
Function 3.a. Channel 3 Neutron Flux: High Neutron Flux.
Function 3.b. Channel 3 Neutron Flux: Low Neutron Flux.
Function 3.c. Channel 3 Neutron Flux: Low Detector Voltage.
Function 4.a. Channel 4 Neutron Flux: High Neutron Flux.
Function 4.b. Channel 4 Neutron Flux: Low Neutron Flux.
Function 4.c. Channel 4 Neutron Flux: Low Detector Voltage.
Function 5. Low Reactor Tank Water Level.
Function 6. Seismic Disturbance.
Function 7. High Reactor Tank Water Temperature.
Function 8. Manual Scram Bar.
Function 9. [redacted]

7.2.1 Neutron Flux Channel 1 (Function 1):

Neutron Flux Channel 1 is used to measure neutron flux levels in the source range and provides indication of both the neutron flux level (count rate) and the rate of change of the neutron flux (called the reactor period). The reactor period, T, is defined as the time required for the neutron flux to change by a factor of e (2.718).

The sensor or detector for Neutron Flux Channel 1 is a boron lined proportional counter that uses ^{10}B as the neutron sensitive material. A proportional counter is used in the source range because of the high sensitivity of this type of detector and the ability to discriminate between neutron and gamma radiations in the source range. Above the source range, channel 1 is bypassed. The design minimum sensitivity of the ARRR channel 1 detector is 4.5 counts per second per neutron per square centimeter-second (cps/n/cm^2-sec). A detailed description of the operation of a proportional counter, including how a neutron pulse or count is generated and an explanation of how a proportional counter discriminates between neutron and gamma radiation, is contained in reference 7.5.2.

The Neutron Flux Channel 1 detector is designed to operate in the source range from the minimum detectable count to approximately 1 watt. To provide neutron count rate over this extended range, the neutron flux channel 1 detector output is connected to a preamplifier whose output drives a ratemeter with a logarithmic element (thermistor) in its output amplifier. The neutron detector output pulses are amplified and then rate-averaged by a logarithmic count ratemeter (10^1 to 10^7 counts per minute) to provide the neutron flux level. The channel 1 console instrument's log circuitry allows the entire count rate range to be displayed without changing the scale. In addition to count rate indication on the reactor console, this signal is also used as an input to the bi-stable trip card that provides a low source level scram and prevents withdrawal of the safety rod below a minimum count rate.

The count ratemeter output is amplified and used to drive a period circuit. The rate of change measurement is accomplished by coupling the changing DC output of the ratemeter through a capacitor to another amplifier. The capacitor blocks the DC component so that only the changing portion is seen by the period amplifier. Since the signal is derived from a logarithmic amplifier, the voltage at the capacitor input is proportional to the log of the neutron flux. The rate of change produces a voltage through the capacitor proportional to 1/T. The amplifier's output meter is calibrated in seconds (-30 to +3 seconds). In addition to the indication of reactor period on the reactor console, this signal is also used as an input to the bi-stable trip card that monitors short reactor period.

In addition to indication of source range count rate and reactor period, there are two RPS scram functions associated with neutron flux channel 1: short period (function 1.a); and, low source level (function 1.b).

<u>Neutron Flux Channel 1 Short Period (Function 1.a)</u> provides a reactor scram signal if reactor period (time required for the neutron flux to change by a factor of e (2.718)) is less than the Technical Specification Limiting Safety System Setting (LSSS) of 3 seconds. This reactor scram function limits reactivity addition to a rate that ensures increases in reactor power level can be observed and controlled.

<u>Neutron Flux Channel 1 Low Source Level (Function 1.b)</u> provides a reactor scram signal if the measured neutron count rate is not greater than the Technical Specification Limiting Safety System Setting of 120 counts per minute (cpm). This reactor scram function ensures the presence of a measurable neutron source prior to allowing control rod withdrawal for reactor startup. The presence of a measurable neutron source prior to control rod withdrawal enables the operator to measure and control reactivity additions during the reactor startup, ensures that the RPS system can detect and respond to a short reactor period and limits the peak power that would be achieved during an uncontrolled rod withdrawal from the source range.

Bi-stable trip cards associated with the low source level and short period Channel 1 functions change state and interrupt the scram bus continuity when the pre-set limits are exceeded. The pre-set limits are set more conservative than the LSSS specified in the Technical Specifications.

Operating the channel 1 proportional counter at high flux levels will damage the detector. Therefore, when neutron flux levels are high enough to be measured on Neutron Flux Channel 2, a solid state relay on channel 2, similar to the scram trip circuits, automatically provides a bypass around (i.e., disables) the channel 1 contacts on the scram bus for short period (Function 1.a) and low source level (Function 1.b), removes the high voltage power supply to the channel 1 detector, and establishes a short circuit across the de-energized channel 1 detector. On increasing power, channel 1 is automatically removed from service when channel 2 is on scale and indicates approximately 1 x 10^{-10} amps. On decreasing power, channel 1 is automatically restored to service when channel 2 indicates approximately 9 x 10^{-11} amps to ensure channel 1 is operating before channel 2 reaches the lower end of its range.

7.2.2 Neutron Flux Channel 2 (Function 2):

Neutron Flux Channel 2 is used to measure neutron flux levels in the intermediate and power range (10^{-2} watts to 120% Rated Thermal Power (RTP)) to provide indication of both the power level and the rate of change of the power level (called the reactor period). The reactor period, T, is defined as the time required for the neutron flux to change by a factor of e (2.718). A channel 2 bi-stable trip card is used to automatically disable the scram for short period (Function 1.a) and low source level (Function 1.b) and to de-energize the channel 1 detector, when neutron flux levels are high enough to be measured on channel 2. This channel 2 bi-stable trip card also controls a bypass around the low neutron flux scrams (Function 3.b and 4.b) provided by channels 3 and 4. This feature ensures that the Functions 3.b and 4.b are bypassed until neutron flux levels are high enough to be measured on channel 2 (i.e. until neutron flux levels are above the Function 3.b and 4.b low flux scram set points).

The sensor or detector for Neutron Flux Channel 2 is a compensated ionization chamber. A compensated ionization chamber is used as the intermediate range channel 2 detector because of the ability to measure neutron flux in the low and medium power ranges by discriminating between neutron and gamma radiation and the ability to operate in high neutron flux without being damaged. When a high voltage gradient is applied across the ionization chamber, the current in the detector circuit is a function of the amount of ionizing radiation that enters the detector. A compensated ion chamber is essentially two differentially connected detecting volumes enclosed in a single case. One of the two nitrogen-filled volumes is coated with boron and is sensitive to both neutron and gamma radiation. The other, uncoated, is sensitive only to gamma activity. The difference between the signals is then a function of the neutron flux incident on the chamber. Gamma compensation is used to extend the lower operating range of an ionization chamber by as much as two decades. At the higher power levels, where the current due to neutron flux is larger than the current produced by gamma radiation, gamma compensation becomes less important. The design minimum sensitivity of the ARRR channel 2 detector is 4 x 10^{-14} amps per neutron per centimeter^2 – second (amps/n/cm^2-sec). A detailed description of the operation of a compensated ionization chamber, including how a compensated ionization chamber discriminates between neutron and gamma radiation, is contained in reference 7.5.2.

The channel 2 detector produces a current that is a function of the neutron flux. The output of the detector is measured with a wide range logarithmic current meter covering the range of 10^{-12} to 10^{-4} amperes (8 decades). Before reactor startup (i.e., when neutron flux is negligible), current from the detector is adjusted to approximately 10^{-11} amps. This procedure adjusts channel 2 compensation voltage to provide overlap between channels 1 and 2. This ensures that channel 2, including the period and flux level measuring indications, are on-scale before channel 1 reaches its upper limit of operation.

In addition to providing indication of neutron flux levels and reactor period and controlling bypasses for channel 1,3 and 4 features, there are three RPS scram functions associated with neutron flux channel 2: short period (function 2.a); loss of instrument power (function 2.b) and, Low Detector Voltage (function 2.c).

<u>Neutron Flux Channel 2 Short Period (Function 2.a)</u> provides a reactor scram signal if reactor period is less than the Technical Specification Limiting Safety System Setting (LSSS) of 3 seconds. This reactor scram function limits reactivity addition to a rate that ensures increases in reactor power level can be observed and controlled.

<u>Channel 2 Neutron Flux: Loss of Instrument Power (Function 2.b)</u> initiates a scram on loss of power to the instrument channel because, on loss of instrument power, channel 2 level indication will fail low while the period indication will read infinity, neither of which will cause a scram. However, channel 2 level indication failing low will, after a short delay, re-energize the channel 1 detector which, if neutron flux level is high, will damage the channel 1 detector. This feature, which protects the channel 1 detector from damage, is not required for reactor safety.

<u>Channel 2 Neutron Flux: Low Detector Voltage (Function 2.c)</u> provides a reactor scram signal on low voltage from the detector high voltage power supply because a reduction in the detector voltage could cause the indicated neutron flux level to be lower than the actual flux level. This RPS function ensures that low detector voltage is a fail safe condition.

Channel 2 is also designed to initiate a scram on loss of power to the instrument channel because, on loss of instrument power, channel 2 level indication will fail low while the period indication will read infinity, neither of which will cause a scram. However, channel 2 level indication failing low will, after a short delay, re-energize the channel 1 detector which, if neutron flux level is high, will damage the channel 1 detector. This feature, which protects the channel 1 detector from damage, is not required for reactor safety.

Bi-stable trip cards associated with Short Period (Function 2.a) and Low Detector Voltage (Function 2.c) change state and interrupt the scram bus continuity when the pre-set limits are exceeded in the same manner as described for channel 1. The pre-set limits are set more conservative than the LSSS specified in the Technical Specifications.

7.2.3 Neutron Flux Channel 3 (Function 3)

Neutron Flux Channel 3 is used to measure neutron flux levels in the power range (approximately 30 watts to 300 watts which is 120% Rated Thermal Power (RTP)). The detector for Neutron Flux Channel 3 is an uncompensated ionization chamber. When a high voltage gradient is applied across the ionization chamber, the current in the detector circuit is a function of the amount of ionizing radiation that enters the detector. An uncompensated ionization chamber is used because, at power levels of a few watts, the neutron induced current is sufficiently higher than that due to gammas so that, at steady state or slowly increasing power, the channel 3 detector will accurately indicate the power related current on the linear current meter. However, a rapid reduction of reactor power will not result in a commensurate rapid reduction of current from the detector because short and intermediate lived fission products produced at high power will continue to cause ionization in the detector for a period of time after the neutron flux levels have decreased. Slow response to a rapid reduction in power level is not a safety concern because channel 3 indicated power will be higher than actual power and channels 2 and 4 provide accurate indications of neutron flux levels. The lack of compensation on channel 3, however, makes for a simpler and more reliable detector and requires one less power supply for its operation. The design minimum sensitivity of the ARRR channel 3 detector is 4.4×10^{-14} amps per neutron per centimeter^2 – second (amps/n/cm^2-sec). A detailed description of the operation of an uncompensated ionization chamber is contained in reference 7.5.2.

The channel 3 detector produces a current that is a function of the neutron flux. The output of the detector is measured by a linear micro-micro ammeter. The linear ammeter is designed to monitor a ten decade range of reactor power (1×10^{-13} amps to 10×10^{-4} amps) using a manually operated twenty position range switch. The operator must carefully monitor reactor power when changing power levels and adjust the range switches to keep the reactor power indication on scale.

In addition to indication of reactor power level, there are three RPS scram functions associated with neutron flux channel 3: High Neutron Flux (Function 3.a); Low Neutron Flux (Function 3.b); and Low Detector Voltage (Function 3.c).

<u>Channel 3 Neutron Flux: High Neutron Flux (Function 3.a</u>) provides a reactor scram signal before reactor power exceeds 98% of full scale on each range and before reactor power exceeds 120% RTP. This reactor scram function ensures that the reactor operator changes the instrument range prior to reactor power going off scale high and terminates a power excursion that exceeds 120% RTP. Additionally, the scram at 98% of full scale on each range serve as a backup to

the channel 2 reactor scram on short period by initiating a scram within approximately 1/3 decade when the reactor period is too short for the operator to adjust the range switches to keep indicated reactor power on-scale. Technical Specifications require that the reactor scram signal be set ≤ 98% of full scale on each range but the set point must also be high enough on the range to allow an operator to changes ranges without initiating a scram signal. The actual set point is typically 70% of full scale.

Channel 3 Neutron Flux: Low Neutron Flux (Function 3.b) provides a reactor scram signal before reactor power falls below 5% of full scale on each range. This reactor scram function ensures that the reactor operator changes the instrument range prior to reactor power going off scale low. Technical Specifications require that the reactor scram signal be set ≥ 5% of full scale but the set point must also be low enough on the range to provide time for the operator to change ranges when decreasing power. The actual set point is approximately 7% of full scale. This scram function also provides a scram on loss of instrument power on channel 3 because the channel will fail low on loss of instrument power.

Channel 3 Neutron Flux: Low Detector Voltage (Function 3.c) provides a reactor scram signal on low voltage from the detector high voltage power supply because a reduction in the detector voltage could cause the indicated neutron flux level to be lower than the actual flux level. Technical Specifications require that the reactor scram signal be set ≥ 500 volts. The actual set point is approximately 700 volts. This RPS function ensures that low detector voltage is a fail safe condition.

Bi-stable trip cards associated with High Neutron Flux (Function 3.a), Low Neutron Flux (Function 3.b), and Detector Voltage (Function 3.c) change state and interrupt the scram bus continuity when the pre-set limits are exceeded in the same manner as described for channel 1.

7.2.4 Neutron Flux Channel 4 (Function 4)

Neutron Flux Channel 4 is identical to channel 3 with the exception that channel 4 uses a compensated ionization chamber (identical to the one used in channel 2) instead of the uncompensated ionization chamber used for channel 3. Therefore, channel 4 has the ability to discriminate between neutron flux and gamma flux. This difference enables channel 4 to accurately track reactor power when decreasing power. Additionally, by adjusting the compensating voltage, it is possible to use channel 4 to monitor neutron flux levels when operating near the source range.

Channel 4 Neutron Flux: High Neutron Flux (Function 4.a) provides a reactor scram signal before reactor power exceeds 98% of full scale on each range and before reactor power exceeds 120% RTP. This reactor scram function operates identically to Function 3.a.

Channel 4 Neutron Flux: Low Neutron Flux (Function 4.b) provides a reactor scram signal before reactor power falls below 5% of full scale on each range. . This reactor scram function operates identically to Function 3.b.

Channel 4 Neutron Flux: Low Detector Voltage (Function 4.c) provides a reactor scram signal on low voltage from the detector high voltage power supply because a reduction in the detector voltage could cause the indicated neutron flux level to be lower than the actual flux level. This reactor scram function operates identically to Function 3.c.

7.2.5 Low Reactor Tank Water Level (Function 5)

A reactor tank water level float switch provides an alarm and a reactor scram signal before the reactor tank water level falls to a level 16 feet above the top of the reactor core. Maintaining a minimum of 16 feet depth of pool water above the top of the core provides the necessary shielding to reduce personnel exposure. Activation of the float switch at the low level set point interrupts the continuity of the reactor scram bus and initiates a scram in the same manner as the neutron flux channels. USAR Chapter 5, Reactor Coolant, describes the design features that minimize the potential for a loss of water inventory from the reactor water tank. Loss of primary coolant cleanup (demineralizer) loop flow, which will occur when reactor water tank level falls below the level of the suction skimmer at the water surface, provides a diverse alarm that could be indicative of lower reactor tank water level.

7.2.6 Seismic Disturbance (Function 6)

A seismic disturbance (earthquake) monitor, located in the trench, east of the reactor enclosure, provides an alarm and a reactor scram signal during a seismic event with an intensity 'IV' on a Modified Mercalli scale. An intensity 'IV' seismic event has an average peak acceleration of 0.015 g to 0.02 g, where g is gravity = 9.8 meters/sec^2. ARRR components are designed for lateral acceleration in excess of 1.0 g. This corresponds to intensities above X on the Modified Mercalli scale. The purpose of this scram function is to ensure the reactor is shut down as soon as the initiation of a seismic event is detected. This design is sufficiently conservative to ensure that the reactor will be shut down in the event of a major earthquake. The Technical Specification LSSS for the seismic monitor is a seismic event with intensity 'IV' which is characterized by the following:

- During the day it is felt indoors by many, outdoors by a few.
- At night some are awakened.
- Dishes, windows, and doors are disturbed; walls make creaking sound.
- Sensation like heavy truck striking building.
- Standing automobiles rock noticeably.

7.2.7 High Reactor Tank Water Temperature (Function 7)

A temperature switch, located in the reactor water tank under the bridge near the top of the pool, monitors reactor water temperature and provides an alarm and a reactor scram signal before the reactor tank water temperature reaches 130 F. The TS limit of ≤ 130 °F for reactor water temperature is intended to minimize thermal degradation of the ion exchange resin in the demineralizer loop and is not required for reactor protection.

7.2.8 Manual Scram Bar (Function 8)

The Manual Scram Bar enables the operator to manually initiate a scram using a circuit that operates independently from the automatic scram circuit. This function satisfies the design requirement that no single component failure or circuit fault shall simultaneously disable both the automatic and manual scram circuits.

7.2.9 [redacted] (Function 9)

[redacted]

Figure 7.2-1
Block Diagram of the Nuclear Instrumentation

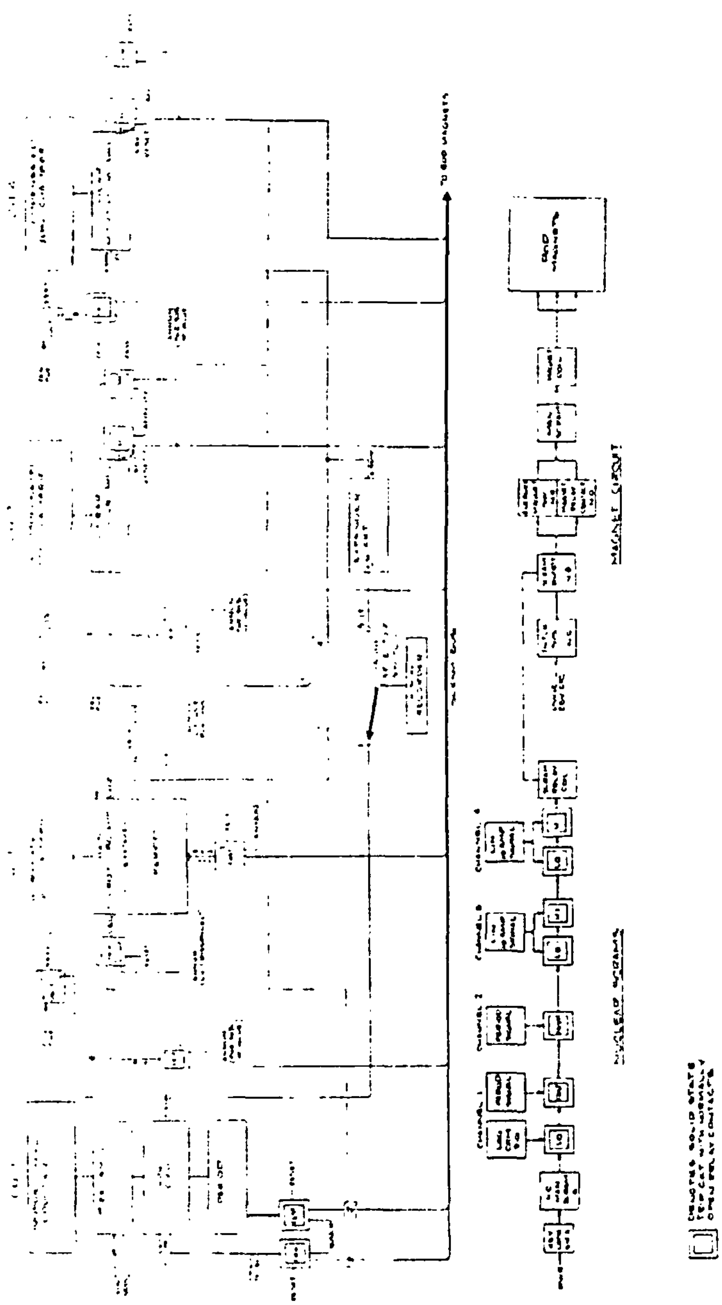


Figure 7.2-2
Logic Diagram of the Reactor Protection System

[redacted]

Figure 7.2-3
Schematic Diagram of the Scram Bus and CRD Electro Magnet Circuit

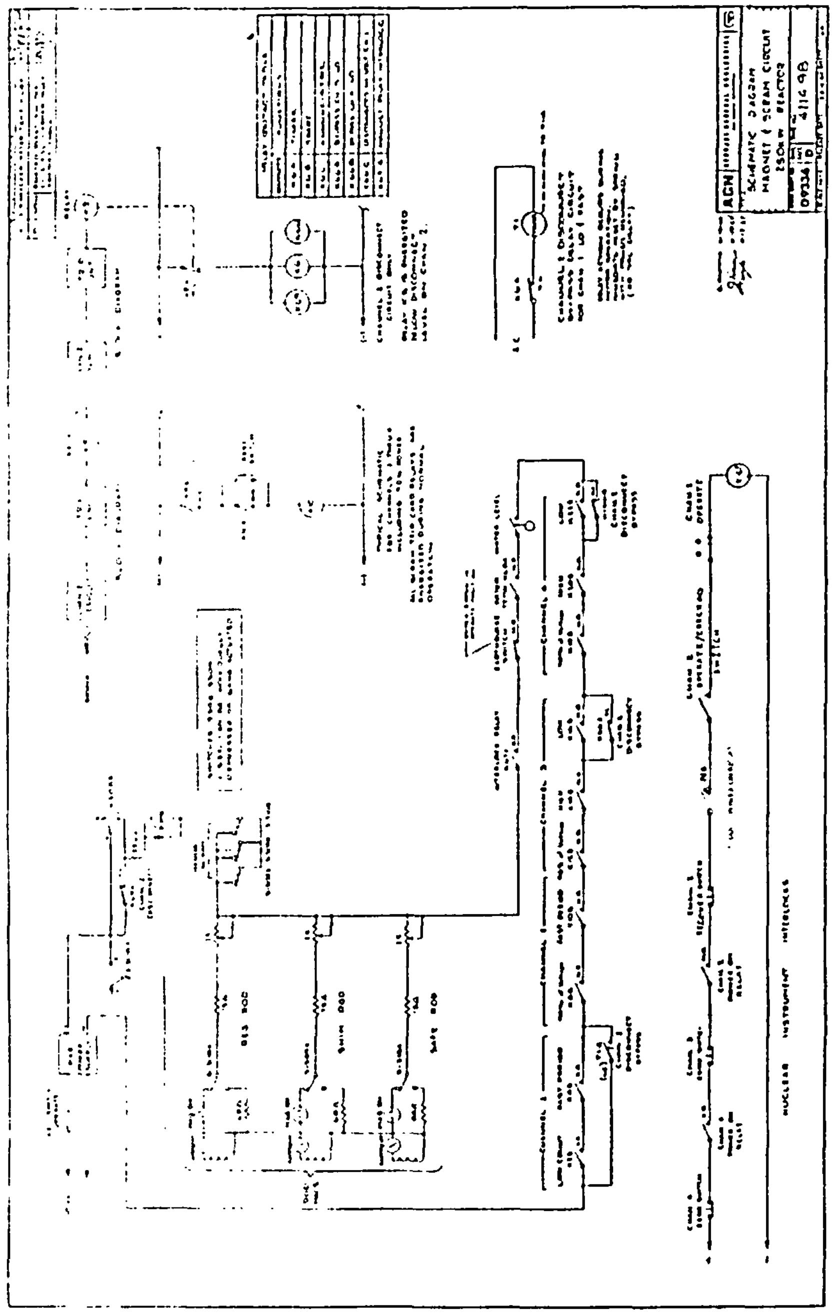
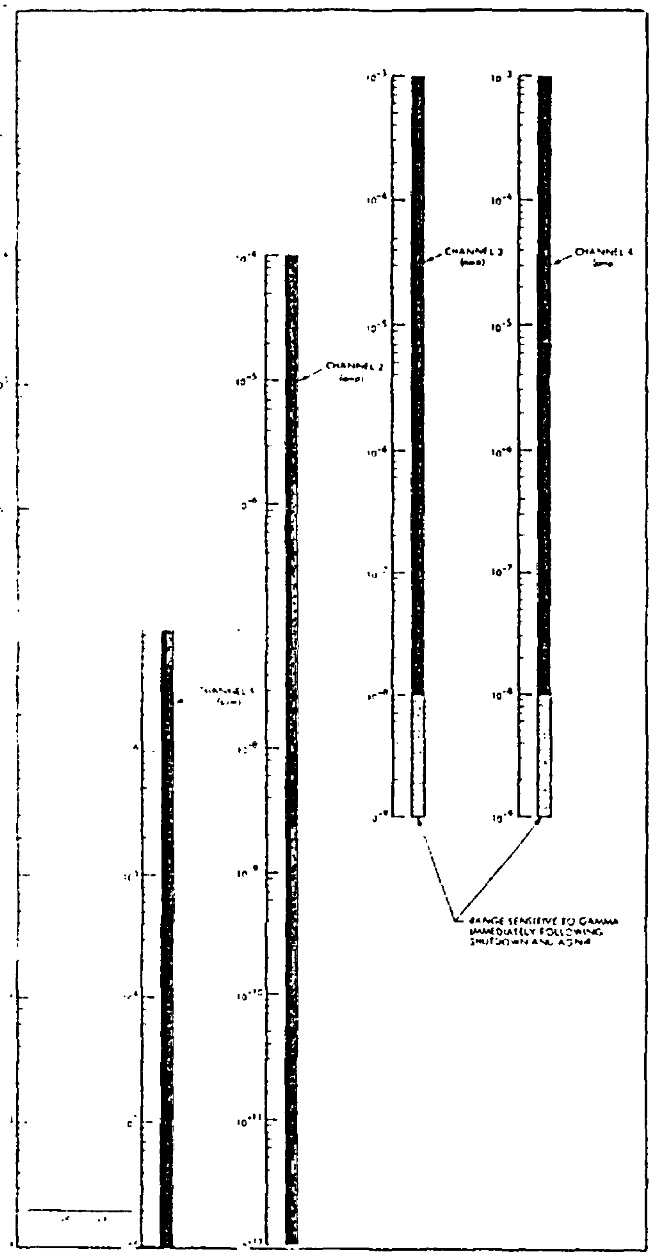


Figure 7.2-4
Normal Operating Range of the 4 Neutron Flux Detectors



7.3 REACTOR MONITORING SYSTEMS (RMS)

The Reactor Monitoring systems (RMS) consist of sensing devices and associated circuits which automatically actuate visual and audible alarms when pre-set limits are exceeded for bridge crane location, reactor water radioactivity, primary coolant conductivity, or primary coolant cleanup (demineralizer) loop flow. Requirements for periodic channel checks, channel tests, and channel calibrations for each of the RMS functions are specified in the Technical Specifications.

7.3.1 Bridge Crane Location

The bridge crane is normally stored approximately [redacted] north and above the pool tank and is maintained in the storage position when not in use. Bridge crane location is monitored by a [redacted] when the [redacted] bridge crane is moved off the storage position. The bridge crane is designed for lateral accelerations in excess of [redacted]. However, except when in use, the bridge crane is kept in the storage location as an added precaution against it falling into the reactor pool as a result of an earthquake. This alarm ensures that the console operator is cognizant of the location of the bride crane.

7.3.2 Primary Coolant Conductivity

Primary Coolant Conductivity is monitored by a conductivity cell which provides continuous indication of reactor water conductivity. Operation of the demineralizer is normally sufficient to maintain a reactor water conductivity and pH within the TS limits of ≤ 5 μmho/cm and ≤ 7.5, respectively.

7.3.3 Reactor Water Radioactivity

Reactor water radioactivity is monitored by a radiation monitor on the suction side of the primary coolant cleanup (demineralizer) loop which provides continuous indication of reactor water radioactivity and activates an alarm if a preset limit is exceeded. The radiation monitor is located in the suction line so that the radioactivity monitor is located upstream from the particle filter and mixed bed demineralizer. The radioactivity monitor is located away from the reactor (i.e., in the heat exchanger building) and is shielded to ensure it is isolated from any direct radiation field.

The reactor water radioactivity monitor uses a count ratemeter (linear) for measuring reactor water radioactivity. The monitor is a gamma scintillation detector which produces pulses proportional to the energy of the interacting gamma photons. These pulses are discriminated to eliminate the low energy scattered radiation always present and yet respond to important fission product energies which would be present in event of a fuel element rupture.

The range of the reactor water radiation monitor is 0.1 to 100 mrem/hour and can be extended to 10^{11} mrem/hour. The Technical Specification limit for this alarm,

≤ 20 mrem/hour, corresponds to a measurable concentration of approximately 5 x 10^{-2} μCurie/cubic centimeter in water. The radiation monitor and the associated alarm will provide an early indication of a fuel element cladding failure.

7.3.4 Reactor Water Demineralizer Water Flow

Reactor water flow through the primary coolant cleanup (demineralizer) loop is monitored by a flow switch that actuates an alarm if the flow rate falls below 4 gpm, indicating pump failure or a clogged demineralizer. This alarm is necessary because the primary coolant cleanup (demineralizer) loop must operate whenever the reactor is operating at any power level to support the water radiation monitor which is located in the demineralizer loop flow path. The alarm uses a Proteus Model #155 rotary vane flow switch rated at 20 gpm and a minimum capacity of 4 gpm. The alarm set point is established by the minimum sensitivity of the flow switch.

7.4 CRITICALITY ALARM, RADIATION AND RADIOACTIVE GASEOUS EFFLUENT MONITORING SYSTEMS

The radiation and radioactive gaseous effluent monitoring systems consist of sensing devices and associated circuits which automatically actuate visual and audible alarms when pre-set limits are exceeded for radiation levels and gaseous activity in the area above the reactor water tank. Requirements for periodic channel checks, channel tests, and channel calibrations for radiation and radioactive gaseous effluent functions are specified in the Technical Specifications.

7.4.1 Criticality Alarm and Area Radiation Monitor

The area above the reactor water tank is continuously monitored by a fixed gamma monitor employing Geiger tube detectors located on the wall connecting the control room and the reactor room. This monitor detects the presence and indicates the intensity of gamma radiation and serves as both an area radiation monitor and a criticality alarm. The monitor has a range of 0 to 20 mrem/hour with a Technical Specification requirement that the alarm actuate at ≤ 10 mrem/hour. The area radiation monitor actuates a siren within the reactor building on high radiation level that warns personnel to evacuate the reactor room. [redacted]

7.4.2 Building Gaseous Effluent Monitor

Building gaseous effluents are continuously monitored by sampling air from the area above reactor water tank. The air intake is either just above the reactor pool, which is the normal position, or in the ceiling of the reactor room. These locations ensure gaseous fission product release will be detected.

Radioactivity levels in the air are monitored using a gamma scintillation detector which produces pulses proportional to the energy of the interacting gamma photons. These pulses are discriminated to eliminate the low energy scattered radiation always present and yet respond to important fission product energies which would be present in event of a fuel element rupture.

Technical Specifications require that the air monitor alarm is set at ≤ 2 mrem/hour. This corresponds to 5900 cps as read on the monitor. The maximum gaseous fission product release, from xenon and krypton decay, after a single fuel element failure with the release entirely in the reactor room would result in an average dose rate of 59 mrem/hr to a person in the reactor room. One could therefore remain there for about 1.5 hours without exceeding the 100 mr/week permissible dose. Building gaseous effluent monitor alarm set point ensures early detection of a leaking fuel element and ensures that corrective action can be taken before the problem becomes too severe.

7.5 REFERENCES

7.5.1 "Aerojet-General Nucleonics Industrial Reactor (AGNIR): Hazards Summary Report (AN-1193)," R. L. Newacheck, Project Engineer et al, September 1964.

7.5.2 ARRR Reactor Operator Training Manual, Volume 2 – Reactor Design Features.

8.0 ELECTRICAL POWER SYSTEMS

8.1 NORMAL ELECTRICAL POWER SYSTEMS

The electrical power requirements of the ARRR facility are supplied from the Pacific Gas & Electric Company grid through a transformer that supplies 3 phase, 60 hertz power at 240 and 120 Vac.

8.2 EMERGENCY ELECTRICAL POWER SYSTEMS

There are no safety-related electrical power supplies because none are needed for preventing or mitigating an accident or for maintaining the facility in safe shutdown condition, even for extended periods of time. In the event of loss of electrical power to the control rod drive system or the reactor safety system, the control rods de-couple from the control rod drive system allowing the control rods to drop into the core by gravity, causing safe shutdown of the reactor. The ARRR is designed for natural convection cooling and the core is located in a [redacted] reactor water tank capable of removing decay heat by evaporation for extended periods of time. Additionally, a sudden and complete loss of reactor coolant will not result in fuel damage or release of fission products because the operating power level of ≤ 250 kW results in minimal decay heat. Therefore, there is no requirement either for an emergency core cooling system or for containment or confinement of the reactor following any accident. As explained in USAR Chapter 6, Engineered Safety Features, and in USAR Chapter 13, Accident Analyses, there are no accidents or transients that depend on the availability of electrical power to protect the public heath and safety.

8.3 BATTERY BACKUP ELECTRICAL POWER SYSTEMS

The ARRR Technical Specifications require a Criticality and Area Radiation Monitoring in the reactor building that actuates a siren within the reactor building on high radiation levels in the area above the reactor. [redacted]

The Criticality and Area Radiation Monitoring alarm system [redacted]. This auxiliary power supply is held at full charge by a low voltage rectifier circuit. The auxiliary power supply capacity is [redacted] under high limit alarm conditions and [redacted] under non-alarm conditions.

The ARRR Technical Specifications also require a site security alarm system that is monitored continuously to detect unauthorized entry into the reactor building. [redacted]

The ARRR reactor building is also equipped with [redacted] that activate when the building power fails to allow safe egress from the reactor building.

8.4 REFERENCES

There are no references for this section.

9.0 AUXILIARY SYSTEMS

9.1 HEATING, VENTILATION, AND AIR CONDITIONING SYSTEMS

The ARRR building ventilation systems are designed with two objectives: provide normal heating, cooling, and ventilation functions for personnel comfort and equipment cooling; and, to protect personnel from exposure to airborne radioactivity and prevent the spread of contamination. However, the accident analyses (USAR Chapter 13, Accident Analyses) do not assume that the building acts as either a containment or confinement to mitigate the release of radioactivity following a reactor accident. A plan view of the ARRR building ventilation systems is shown in Figure 9.1-1.

Normal heating, cooling, and ventilation is provided by the building air handling systems which include six different heating and air conditioning systems and 4 exhaust systems. They recirculate and condition a significant portion of the air from the areas they serve and receive makeup air from inside or outside the facility. Four of these systems are refrigeration-type air conditioners.

Personnel protection from exposure to airborne radioactivity and prevention of the spread of radioactive contamination is based on a ventilation strategy that assumes that leakage from the fuel elements or a spill of material from an irradiated experiment will be the primary source of airborne radioactivity and contamination. The ventilation system is designed to maintain pressure differentials between selected areas to achieve two objectives: confinement of airborne radioactive material to the reactor high bay area; and, exclusion of airborne radioactive material from inhabited areas, especially the control room.

Confinement of airborne radioactive material to the reactor high bay area is enhanced by a ventilation system that does not provide outside air directly into the reactor high bay area. Areas adjacent to the high bay area are supplied with outside air and are at a pressure that is slightly positive relative to the high bay area. Any inleakage into the high bay area is directed out of the building through three gravity ventilators in the roof over the high bay area. This prevents air from entering the reactor building through the reactor high bay area and minimizes the potential for spreading airborne radiation or contamination from the high bay area to other parts of the building.

The boundary of the reactor high bay area includes the following: [redacted]

During any radiological event that could spread contamination or airborne radiation within the building or release it to the environment, the reactor high bay ventilation system, setup room ventilation system, chemical laboratory hood blower, and rest room vent fans are shut off to reduce the potential for spread of contamination and airborne radiation. All of the ventilation equipment that should be shut down during a radiological event is controlled by an Air Handling Relay on a single circuit. [redacted]

Exclusion of airborne radioactive material from inhabited areas, especially the control room, is accomplished by maintaining the areas at a positive pressure relative to the reactor bay. The airflows throughout the facility have been designed so that the control room, the lunch room, the office complex, the rest rooms, and the machine shop are at positive pressure with respect to the reactor high bay area. The positive pressure is maintained by an oversized ventilation system with a high fraction of make-up air from the outside. Technical Specifications require that the ability to maintain a positive pressure in the control room with respect to the reactor high bay area and this capability is tested quarterly. ARRR policy requires periodic verification of the ability to maintain the lunch room and the office complex at a positive pressure relative to the reactor high bay area and has designated these areas as "clean areas." Other areas with fresh air intakes do not qualify for clean air rooms because the pressure is not measured routinely and documented. Such areas include the electronic lab, the storage shed, and the quality control complex (i.e., dark room, quality control room, visitors viewing room and the back office space.)

9.1.1 High Bay Area Ventilation and Air Conditioning

The reactor high bay area includes a water cooled, 20 ton air conditioner, located in the reactor bay. Additionally, natural gas fired heating units are supported from the ceiling. The air conditioner or the heaters recirculate air in the reactor high bay area for cooling or heating. Neither the air conditioner nor the heaters has a fresh air intake in order to prevent air from entering the reactor building through the reactor high bay area. This minimizes the potential for spreading airborne radiation or contamination from the high bay area to other parts of the building. Both the air conditioner and the heaters in the reactor high bay area are powered via the Air Handling Relay single circuit and can be shut down using a button in the control room.

9.1.2 Control Room Ventilation and Air Conditioning

Control room ventilation and air conditioning consists of a 5 ton air conditioner unit that is mounted above the control room. Electrical heating elements are located in the duct for heat.

The control room ventilation system was designed to minimize exposure of control room personnel to airborne contamination from the reactor high bay area. The control room ventilation system uses an oversized circulating fan aligned to inject a high percentage of make-up air from the outside into the control room. This system maintains the control room at a positive pressure relative to the reactor high bay area.

Technical Specifications require that the control room is maintained at a positive pressure with respect to the reactor high bay area whenever the reactor is not in the secured condition. This requires that the control room circulating fan is operating prior to taking the reactor out of the reactor secured condition. Additionally, Technical Specifications require that quarterly verification that the control room circulating fan maintains positive air pressure in the control room. The control room ventilation circulating fan is not included on Air Handling Relay single circuit because control room pressurization is maintained during any radiological event.

9.1.3 Office Area Ventilation and Air Conditioning

The office area ventilation and air conditioning system is a combination gas furnace and air conditioner located above the machine shop area. The air conditioning compressor and condenser are located outside the south end of the building. The office area ventilation system uses outside makeup air and, although not required by Technical Specifications, maintains the office area at a slight positive pressure relative to the reactor high bay area when the system is in operation. Normal procedure is to keep the office area ventilation fan in operation when the building is occupied.

The office area ventilation system fan is not included on Air Handling Relay single circuit because office area pressurization is desirable during any radiological event. The office area ventilation system fan is capable of maintaining a positive pressure in the control room office complex if the control room ventilation fan fails. The ability of the office area ventilation system fan to maintain the control room at a positive pressure is not required by Technical Specifications and this capability is not routinely tested.

9.1.4 Neutron Radiography Set-Up Room Ventilation and Air Conditioning

Neutron radiography setup room ventilation and air conditioning is provided by the reactor high bay area 20 ton air conditioner. The neutron radiography setup room uses a separate gas furnace, located above the quality control room, for heating. Air intake for the gas furnace is the top plenum of the high bay area 20 ton air conditioner. As subsystem of the reactor high bay area ventilation system, the Neutron radiography setup room ventilation system does not have a fresh air intake and is powered via the Air Handling Relay single circuit and can be shut down using the button in the control room.

9.1.5 Quality Control Complex Ventilation and Air Conditioning

The quality control complex ventilation and air conditioning consists of a combination air conditioner and gas heater located above the quality control room. This system uses mostly recirculated air with some makeup air from a roof vent. The quality control complex ventilation maintains the quality control complex at a slight positive pressure relative the to reactor high bay area. However, this feature is not required by Technical Specifications and this capability is not routinely tested. Therefore, the quality control complex is not approved for occupancy during an emergency. The quality control complex ventilation system fan is not included on Air Handling Relay single circuit because quality control complex pressurization is desirable during any radiological event.

9.1.6 Chemical Laboratory Hood

The chemical laboratory hood includes a 3000+ cfm blower that vents the hood to the outside through the south end of the building. The chemical laboratory hood blower ultimately takes input air from the reactor high bay area; therefore, the chemical laboratory hood must be shut off during a radiological event to prevent spreading contaminated air outside the building. The chemical laboratory hood blower is included on the Air Handling Relay single circuit.

9.1.7 Rest Room Vent Fan

The rest room vent fans are 100 cfm exhaust fans located in the rest rooms adjacent to the control room. The rest room vent fans are included on the Air Handling Relay single circuit because operation of these fans could interfere with the ability to maintain the control room at a positive pressure during a radiological event.

9.1.8 Film Processor Vents

The film processor vents for the film dryer exhausts hot air from the dryer to the outside through the roof of the building. The film processor vents are not included on Air Handling Relay single circuit because the air pressure in the quality control complex is slightly positive relative to the reactor high bay area.

9.1.9 Film Processing Chemical Room Vent

The film processing chemical room vent is a small fan that exhausts chemical fumes to the outside through the roof. The film processing chemical room vent is not included on Air Handling Relay single circuit because the quality control complex is at a slightly positive pressure relative to the reactor high bay area.

Figure 9.1-1
ARRR Building Ventilation Systems

[redacted]

9.2 FUEL STORAGE

9.2.1 Fuel Storage in the Reactor Tank

A fuel element storage rack with positions for [redacted] elements is located along the wall of the reactor tank. Fuel elements in this storage rack are centered about [redacted] above the floor of the reactor tank. Technical Specifications require that fuel elements in the reactor tank but not part of the reactor core shall be stored in a geometric array where keff ≤ 0.9 for all conditions of moderation and reflection using light water. The design of the fuel element storage rack ensures that this requirement is satisfied when storing any standard TRIGA fuel element.

9.2.2 Fuel Storage Pits

[redacted] fuel storage pits, located in the floor of the reactor room, are provided in the design of the reactor building (Figure 9.2-1). These holes are [redacted] in diameter and [redacted] deep and may be used to store fuel elements that are not in use, or to store any radioactive material. Each fuel storage pit can be flooded individually. When a fuel storage pit is flooded, the radiation level with a [redacted] source located [redacted] above the bottom of the hole will not exceed 2 mrem/hour at the floor surface.

Technical Specifications allow a maximum of [redacted] fuel elements [redacted] of U-235) in each of the [redacted] storage pits and allow the storage pits to be dry or flooded with water. Additionally, Technical Specifications require that fuel storage pits containing one or more fuel element [redacted]

9.2.3 Fuel Shipping Containers

A shielded fuel transfer cask is used for the transfer of highly radioactive fuel elements between the reactor pool and the fuel storage pits. When transferring fuel elements offsite, a fuel element must be contained in an approved fuel shipping container. In either case, Technical Specifications allow a maximum of one fuel element in the ARRR facility that is not either in storage or in the reactor core lattice.

Figure 9.2-1
Typical Fuel Element Storage Pit

[redacted]

9.3 [redacted]

[redacted]

9.3.1 [redacted]

[redacted]

9.3.2 [redacted]

[redacted]

[redacted]

9.3.3 [redacted]

[redacted]

Figure 9.3-1

[redacted]

[redacted]

Figure 9.3-2
[redacted]

[redacted]

9.4 FIRE PROTECTION SYSTEMS AND PROGRAMS

Fire protection at the ARRR facility is centered on an automatic sprinkler system that was installed in 1981. The ARRR building sprinkler system is a wet pipe system that is connected to the existing 6 inch city water main [redacted] An outside Post Indicator Control Valve, [redacted] is provided to allow isolation of the underground water supply to sprinkler system from outside of the building.

The ARRR building sprinkler system uses standard industrial [redacted] sprinklers with [redacted] Sprinkler locations are shown in Figure 9.4-1. The sprinkler system was installed in accordance with the National Fire Protection Association standards for sprinklers systems. The sprinkler system is inspected quarterly by Aerotest Operations personnel and annually by the San Ramon Valley Fire Protection District (SRVFPD). It is tested every 5 years by the National Fire Extinguisher Company.

A fire alarm pull box is located [redacted]

Fire extinguishers (Type ABC) are located throughout the ARRR facility as shown in Figure 9.4-2. The fire extinguishers are inspected monthly by Aerotest Operations personnel and tested annually by the National Fire Extinguisher Company. Annual training of Aerotest Operations personnel in use of fire extinguishers is provided by the National Fire Extinguisher Company.

[redacted]

Figure 9.4-1
Sprinkler Locations

[redacted]

Figure 9.4-2
Fire Extinguisher Locations

[redacted]

9.5 COMMUNICATION SYSTEMS

USAR Chapter 7, Instrumentation and Control Systems, describes the area radiation monitor [redacted]

An evacuation alarm (designated as "Radiation") can be actuated from [redacted]

[redacted]

[redacted]

[redacted]

9.6 POSSESSION AND USE OF BYPRODUCT, SOURCE, AND SPECIAL NUCLEAR MATERIAL

Requirements governing possession and use of byproduct, source, and special nuclear material are listed in the ARRR operating license. Specifically, the ARRR license specifies that ARRR is authorized to receive, possess, and use up to 5.0 kilograms of contained uranium 235 in connection with operation of the reactor in accordance with the Atomic Energy Act of 1954 and 10 CFR Part 70, "Special Nuclear Material." The operating license also authorizes ARRR: (1) to receive, possess, and use a 2 curie americium-beryllium neutron startup source, and (2) to possess, but not to separate, such byproduct material as may be produced by operation of the reactor consistent with the Atomic Energy Act of 1954 and 10 CFR Part 30, "Licensing of Byproduct Material." Special nuclear material and experiments undergoing irradiation within the limits specified in the Technical Specifications are considered part of the reactor's license.

The receipt, possession, and use of byproduct and special nuclear material as authorized by the operating license must be in accordance with the regulations in 10 CFR Parts 30 and 70, including Sections 30.33, 70.23, and 70.31.

9.7 REFERENCES

None

10.0 EXPERIMENTAL FACILITIES AND EXPLOSIVES

The ARRR provides a neutron source for research and development and services, mainly neutron radiology. Irradiation services for activation analyses have included: crude oil and hydrocarbon samples for oil companies; plastic slides impregnated with microscopic quantities of fissionable materials; ocean silt samples for the Bureau of Mines; and, silver iodide in snow samples from cloud seeding. Other irradiation services have included: calibration of power reactor fission detectors; radiation damage effects studies of solid state electronic components; detection of gunshot residue in paraffin; lattice deformation studies in ammonium perchlorate; and, spallation experiments with uranium dioxide.

The various pieces of equipment used to perform the research and development and services described above are referred to as the experimental facilities and the activities are referred to as experiments. Some of the experimental facilities used for these activities are normally installed and others can be installed as the need arises.

The Technical Specification define an experiment as "Any operation, hardware, or target (excluding devices such as detectors, foils, etc.), that is designed to investigate non-routine reactor characteristics or that is intended for irradiation within the pool, on or in a beamport or irradiation facility, and is not rigidly secured to a core or shield structure so as to be a part of their design." A moveable experiment is one where it is intended that all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating. The ARRR is equipped with the following experimental facilities:

- Neutron Radiography Facility
- Graphite Thermal Column
- Glory Hole Facility
- Vertical Tubes
- Central Core Irradiation Facility
- Triangular Incore Irradiation Facilities
- Incore Irradiation Capsules
- Large Component Irradiation Box (Use prohibited by Technical Specifications)
- Pneumatic Transfer Facility (Use prohibited by Technical Specifications)
- Beam Port (Not Installed)

10.1 EXPERIMENTAL FACILITIES

10.1.1 Neutron Radiography Facility

The neutron radiography facility (Figure 4.1-1) consists of two parts: a vertical beam tube and the radiography facility.

The vertical beam tube is a hollow sealed tube, located adjacent to the core on the [redacted] side of the reactor, which extends from the floor of the reactor tank to above the reactor tank water surface. This vertical beam tube, by providing a path that does not contain the water that acts as a shield, allows a beam of neutrons from the core to reach the radiography facility located above the reactor.

The vertical beam tube consists of a two-section tapered tube with rectangular cross section with the weight supported directly by the bottom of the reactor tank. The vertical beam tube has a total length of approximately [redacted]. The top of the beam tube terminates at the bottom of the reactor bridge structure. The external dimensions of the beam tube are about [redacted] by [redacted] near the base and [redacted] by [redacted] at the top. The top of the vertical beam tube is supported laterally at the top of the pool.

The lower [redacted] of the lower section of the vertical beam tube is filled with graphite for moderation of fast neutrons. The upper section of the vertical beam tube is filled with helium which is a better medium for collimated neutrons than air which scatters the neutrons. Both the upper and lower sections of the vertical beam tube are equipped with fill and drain lines that are used to remove water or purge the vertical beam tube.

The lower [redacted] of the vertical beam tube is covered with lead for gamma shielding. This lead shield is [redacted] thick on the reactor side and [redacted] thick on the other three sides. The lead is protected from the pool water by welded sheets of aluminum. All components contacting the pool water are fabricated from aluminum or stainless steel.

The vertical beam tube includes a pneumatically operated boral (boron carbide and aluminum) shutter mechanism that offers a selection of 5 aperture settings to provide varying depths of field or resolution. The aperture is located near the bottom of the vertical beam tube just above the graphite.

The neutron radiography facility is integrated into the shielded enclosure directly above the vertical beam tube. The shielded enclosure consists of [redacted] thick concrete block shielding stacked to a nominal height of [redacted] above the floor that surrounds the entire top of the reactor water tank. The neutron radiography facility is supported by [redacted] steel "I" beams that transmit the weight of the shielding to beams imbedded in the floor of the reactor building. This shielding supports the [redacted] thick wood (fir) beams that cover the reactor enclosure. This shielding provides operating personnel additional shielding.

This shielding enclosure is penetrated at the [redacted] side of the reactor by the neutron radiography facility. [redacted] The facility was designed to allow the neutron radiography of contained detonating cords having lengths in excess of 25 feet. Concrete blocks in the shield structure may be moved as necessary to accommodate taller objects.

The top of the vertical beam tube is covered by a movable radiation shield called the neutron beam catcher. The beam catcher shield is wood that is [redacted] by [redacted]. The beam catcher shield reduces the radiation level due to neutrons and gammas within the reactor room and at the exclusion area fencing.

The reactivity effect of rapid flooding of the lower section of the vertical beam tube is to add reactivity because the flooding will increase the volume of moderator available to the reactor. Although the reactivity effects from flooding the vertical beam tube were not directly evaluated, a reactivity addition of 0.0657 $\Delta k/k$ ($0.09) was measured (Reference 10.4.2) after flooding of the large component irradiation box. The reactivity effects from flooding the large component irradiation box are larger than the effects of flooding the vertical beam tube because irradiation box volume is greater than that of the portion of the neutron radiography tube in the vicinity of the core and the irradiation box was physically closer to the core than the vertical beam tube.

The potential for damage to the CRDs from the inadvertent detonation of the explosive material being radiographed is limited by strict Technical Specification limits on the amounts and types of explosives that can be radiographed. Additionally, the control rod drives are surrounded by a circular steel can having an outer diameter of about [redacted] and a height of about [redacted]. This can will protect the CRDs from the effects of the inadvertent detonation of any of the explosive devices allowed to be radiographed. The CRD can has a wall thickness of about [redacted]
The shielded enclosure is also designed to allow access to the CRDs for maintenance.

Fire protection in the area around the neutron radiography facility is provided by a [redacted] halon fire extinguisher that can be discharged into the radiography facility to extinguish a fire that occurs during radiography. The halon extinguisher is located on the [redacted]. The extinguisher is operated by twisting the key to break the nylon retaining cord, removing the key, and squeezing the handle. The halon is discharged onto the radiography facility which is shielded from personnel by the concrete enclosure. Additionally, portable dry chemical or halon fire extinguishers are available at selected locations to control any fires that may occur outside of the radiography facilities

10.1.2 Graphite Thermal Column

The graphite thermal column (Figure 4.1-1) is a large block of graphite, encased in aluminum, containing [redacted] rows of [redacted] vertical holes through the graphite block. The vertical holes allow specimens to be inserted into the graphite block for irradiation. The [redacted] rows of irradiation holes (A through E) are [redacted] apart with each row at an increasing radius from the core. The increasing radius of each row allows samples being irradiated to be subjected to different ratios of thermal to fast neutrons. Flux wire holders, located near the centerline of each of the [redacted] rows, were used during startup physics testing (Reference 10.4.2) to measure flux levels at various positions in the thermal column.

There are [redacted] irradiation holes in each row. The irradiation holes are [redacted] in diameter, similar to the reactor core grid plate, which allows the same capsules or devices that are inserted into the core to be inserted into the graphite thermal column.

The thermal column graphite block measures [redacted] along the radial axis of the core and is [redacted] wide and [redacted] deep. It is located on the south side of the reactor and positioned adjacent to the core. The thermal column is positioned using [redacted].

Two irradiation holes in the thermal column are configured for specific tasks:

(1) One irradiation position in the first ('A') row is fitted with an aluminum tube, identical in design to the glory hole (described later), that extends to the top of the reactor water tank above the wood (fir) block shield. Material to be irradiated is lowered through the tube into the thermal column. This tube has a rotating motor to slowly rotate the sample and thus provide an evenly distributed irradiation to the sample. A shield plug is placed in the top of the tube to reduce gamma scattering to acceptable levels.

(2) A second position in the first ('A') row is fitted with a detector calibration system. Small fission detectors can be calibrated against a standard in this facility.

(3) A third position includes a [redacted] diameter neutron beam tube which can be located between the thermal column and N-ray tube and which extends to the area above the top reactor shield. This tube is used for the source of neutrons for the N-gage device

When the thermal column was installed, the reactivity worth was measured (Reference 10.4.2) to be less than 0.007% $\Delta k/k$ (($0.01) with respect to the water it replaces. The small reactivity worth is due to the [redacted] gap between the reactor core structure and the thermal column which effectively separates the reactivity effects of the thermal column from that of the reactor core.

The thermal column structure is also used to mount other experimental facilities. [redacted] slotted beams, [redacted] on each side, are provided to allow experiments to be attached directly to the thermal column. Extensions of these beams allow experiments to be placed immediately adjacent to the reactor core.

10.1.3 Glory Hole Facility

A glory hole facility is an aluminum tube, [redacted] in diameter, which will fit into any fuel element hole. The hollow tube extends from the lower grid plate to above the top of the wood (fir) shield on top of the reactor water tank. The tube is not filled with water and is used to lower material to be irradiated through the tube into the core region. The glory hole will accept capsules with a maximum diameter of [redacted]. Use of a glory hole shortens the sample turnaround time for sample activation, increases the sensitivity of neutron activation analysis, and reduces the use of the dummy element irradiation capsule with its inherent handling problems.

Technical Specifications allow a maximum of one glory hole facility that may be installed in selected locations in any of the [redacted] rings in the core grid plates. At the ARRR, the glory hole is typically installed in the F-2 position of the core. Technical Specifications limit the amount of reactivity that may be introduced to the core by any experiment to ≤ 1.46% Δk/k (≤ $2.00). Therefore, if any experiment located in the glory hole is instantaneously removed from the core, the peak power will be less than that assumed in the accident analysis for an insertion of all excess reactivity (i.e., 2.19% Δk/k (≤ $3.00)).

The glory hole may be used with or without an internal shield plug that reduces the radiation streaming at the top of the reactor water pool in the vicinity of the CRDs. Technical Specifications require that the glory hole be purged with CO_2 to prevent the formation of excessive amounts of ^{41}Ar during reactor operation. When operated with a shield plug, the glory hole is purged prior to each insertion of the shield plug. When operated without a shield plug, the glory hole is purged continuously when the reactor is operating. Additionally, when operated without a shield plug, the installed gas sampling system must be selected to sample in the immediate vicinity of the glory hole so that corrective action can be taken to prevent the release of gaseous activity in excess of 10 CFR 20 limits.

10.1.4 Vertical Tubes

A vertical tube is a [redacted] diameter dry beam tube that is used for low flux dry hole experiments such as research on a new neutron detector. A vertical tube may be attached to any of the [redacted] slotted beams, [redacted] on each side of the thermal column assembly. The vertical tube is supported at the bottom by the thermal column support structure and at the top by the reactor bridge structure. Removable lead disks are inserted into the bottom of the tube to compensate for the buoyant force of the surrounding water. Extensions of this tube allow experiments to be placed immediately adjacent to the reactor core. When the vertical tube is not being used for an experiment, it is placed near the reactor water tank wall to prevent production of ^{41}Ar and radiation streaming.

Technical Specifications require that, when in use, the vertical tube be continuously purged with CO_2 to prevent the formation of excessive amounts of ^{41}Ar during reactor operation. Additionally, Technical Specifications require that the installed gas sampling system be selected to sample in the area above the reactor so that corrective action can be taken to prevent the release of gaseous activity in excess of 10 CFR 20 limits.

10.1.5 Central Core Irradiation Facility

The central core irradiation facility is a hexagonal section that can be removed from the center of the upper grid plate to allow insertion of specimens into the core region of highest flux. Use of the central core irradiation facility requires prior relocation of the central fuel element and the six elements from the B-ring. Technical Specifications limit the size of the central core irradiation facility to [redacted] The facility will accommodate specimens up to about [redacted].

10.1.6 Triangular In-Core Irradiation Facilities

The triangular in-core irradiation facilities consist of two sections cut out of the upper grid plate, each of which encompasses one D-ring and two E-ring holes. When fuel elements are placed in these locations, their lateral support is provided by special aluminum pieces. With the aluminum spacers removed, each of these triangular sections allows the insertion of circular experiments to a maximum of [redacted] diameter or triangular experiments to a maximum of [redacted] on a side.

10.1.7 In-Core Irradiation Capsules

In-core irradiation capsules (Figure 10-1), approximately the same size and shape of a fuel element, are used to irradiate samples and can be used in any open position of the core or in the graphite thermal column. The capsules are usually sealed at the top by a gasket and threaded fitting but also have provisions for bringing instrumented tubes to the surface.

Incore irradiation capsules are inserted and removed from the core using the fuel element handling tools. A shielded transfer cask is used to transport the capsule within the ARRR building.

The capsules are designed to have a maximum inner void volume of [redacted] in the active fuel region and the worth of the dummy irradiation capsule versus water in the reactor core was found to be identical in reactivity with the glory hole (Reference 10.4.2).

10.1.8 Large Component Irradiation Box

A large component irradiation box is an aluminum box with an internal volume of up to [redacted] that can be installed in the reactor water tank. The walls of the box are thin to minimize parasitic neutron absorption.

The large component irradiation box can be installed, as needed, by being lowered onto the movable table and bolted in place with remote handling equipment. The box is positioned on the movable table using tapered locating pins and bolted to the movable table that is, in turn, bolted to the bottom of the ARRR reactor water tank. To remove or install the experiment box, the movable table is required to be moved two or more feet away from the reactor core. The movable table is positioned remotely relative to the reactor core but has positive mechanical stops that prevent moving the box closer than 5 centimeters from the outer ring. When at the position closest to the core, the large component irradiation box cannot encompass more than a 120° arc of the core.

The large component irradiation box must be purged of air prior to exposure to neutrons. Therefore, when installed, the large component irradiation box is pressurized with CO_2 to 0.5 psi above the water pressure. The CO_2 is supplied through aluminum and plastic tubing from a supply at the top of the reactor pool. A relief valve is attached to the top of the box. The box can be configured to accommodate electrical leads between the box and the top of the pool, if required, for any experiment. The box is weighted with lead to eliminate buoyancy.

The large component irradiation box is not currently installed in the reactor water tank. Because ARRR does not currently use and has no plans for future use of the large component irradiation box, Technical Specifications were simplified by prohibiting the use of this component. An amendment of the ARRR operating license would be required before installation and use of the large component irradiation box.

10.1.9 Pneumatic Transfer Facility:

The pneumatic transfer facility is designed to quickly transfer individual specimens into and out of the reactor core. The specimens are placed in a small polyethylene holder, "rabbit," which in turn is placed into the receiver. The rabbit travels through aluminum tubing to the terminus at reactor core centerline and then returns along the same path to the receiver. Directional CO_2 flow moves the rabbit between receiver and terminus. A solenoid valve directs air flow using a timer to regulate the exposure of the sample. A manual control, capable of overriding the automatic timer control, is also provided.

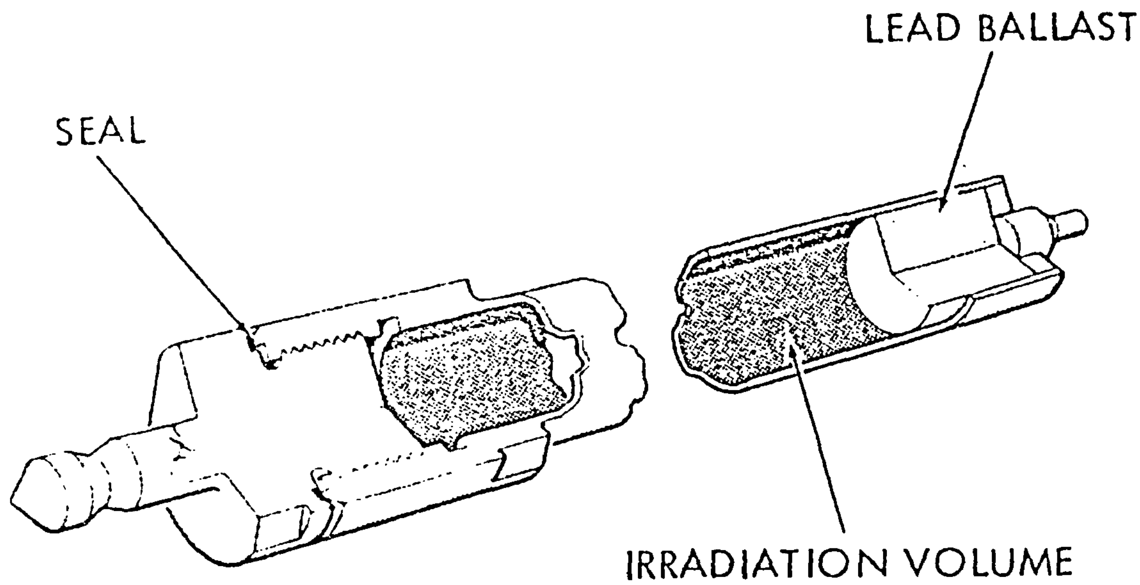
The pneumatic transfer facility may be located in any reactor core position. When installed, the facility is operated with dry CO_2 and exhausted through a filter ventilation system, which is monitored for radioactivity. The in-core portion of the transfer facility has a maximum void volume of [redacted] so that the affects on reactivity are similar to other experiments that are placed in the active core.

The pneumatic transfer facility is not currently installed in the reactor water tank. Because ARRR does not currently use and has no plans for future use of the pneumatic transfer facility, Technical Specifications were simplified by prohibiting the use of this component. An amendment of the ARRR operating license would be required before installation and use of the pneumatic transfer facility.

10.1.10 Beam Port

The ARRR was designed to facilitate future installation of a horizontal beam port. The ARRR reactor tank's concrete embedment includes one penetration consisting of one [redacted] outside diameter pipe about [redacted] long that butts up against the outside of the reactor tank on the center line of the core. The pipe sleeve was provided so that a horizontal beam port could be installed without having to break through the concrete around the tank. However, the beam port facility was never installed and the reactor tank wall is not cut open at this location. An amendment of the ARRR operating license would be required before installation of the beam port.

Figure 10-1
In-Core Irradiation Capsules



10.2 LIMITS ON EXPERIMENTS

Technical Specifications establish limits that ensure that operation and use of the experimental facilities remain within ARRR design and accident analysis assumptions. In addition to specific restrictions on the operation of each type of experimental facility, Technical Specifications establish specific requirements for evaluation and approval of experiments that include limits on explosive materials in experiments, limits on reactivity introduced by experiments, limits on special nuclear material (SNM) included in experiments, and restrictions on types of materials included in experiments.

10.2.1 Limits on Explosive Materials Experiments

The ARRR uses the neutron radiography facility to radiograph various types of explosive devices. The neutron radiography facility is the only experimental facility that is permitted to include explosive material. A combination of the design of the facility, Technical Specification limitations on the type, amount and length of exposure of the explosives being radiographed, and strict operating procedures protect the operating personnel and the public from the effects of inadvertent detonation of the explosive material being radiographed and the release of radioactive material.

As described earlier, the neutron radiography facility is located in a shielded enclosure above the reactor water tank. The shielded enclosure consists of [redacted] thick concrete block shielding stacked to a nominal height of [redacted] above the floor that surrounds the entire top of the reactor water tank. This shielding supports the [redacted] beams which cover the entire enclosure of the reactor water tank. This shielding provides operating personnel shielding from both radiation and blast effects in case of a detonation of any of the explosive devices during radiography. Additionally, potential for damage to the CRDs by the inadvertent detonation of the explosive material being radiographed is prevented by a can with a wall thickness of about [redacted] that surrounds the CRDs [redacted]

Technical Specifications provide the following restrictions on explosive materials:

(1) The maximum amount of explosive material contained in devices that may be placed in the radiography facilities at a time shall be limited to five pounds equivalent TNT.

(2) Explosive material in the radiation field at one time shall be limited to 1 pound equivalent TNT.

(3) Explosive material contained in long devices shall be limited to 0.5 pound equivalent TNT per foot.

(4) The explosive devices shall be subjected to a total exposure not to exceed 3×10^{11} neutrons/cm^2 and 3×10^3 roentgens of gammas.

(5) Explosive devices that have or provide a thrust in a definite direction upon ignition shall be positioned so as to be aimed away from the reactor and components.

This combination of the design of the radiography facility and location of explosives is consistent with the facility design which is to provide protection for the public under the maximum credible accident conditions associated with neutron radiography of explosive devices up to [redacted] and with a maximum of [redacted] in process or in storage. The [redacted] explosive limit is based on Federal Explosives Law and Regulations, ATF P 5400.7 (09/00), table 55.218 which requires a 60 foot minimum distance between the barricaded explosive magazine and the nearest public street which is the employee parking lot.

Personnel who handle explosives are normally exposed to the greatest hazard but with the precautions specified in these Technical Specifications and with adequate training, the probability of a detonation under these circumstances is highly remote as evidenced by almost 40 years of accident-free explosive handling at ARRR.

The limit on radiation exposure for explosives is based on a study (M.J. Urizar, et al., A Study of the Effects of Nuclear Radiation on Organic Explosives, TID-12491) that determined that thermal fluences of 10^{15} neutrons/cm^2 and gamma exposures of 5×10^6 roentgens on explosives have been experimentally evaluated and produce no detrimental effects.

Personnel handling the ordnance devices are provided with and use, as appropriate, non-sparking tools, grounded footwear and clothing that does not produce static electricity. All benches, loading areas and other equipment which can create static electricity are grounded. The operation of unshielded, high frequency generating apparatus within 50 feet of any explosive device is prohibited.

10.2.2 Limits on Reactivity Introduced By Experiments

Technical Specifications require that limits on excess reactivity and shutdown margin are not exceeded including the reactivity worth of the experiment when evaluated in the most reactive condition. This restriction provides the foundation on how the reactor is protected from power excursions and ensures that the reactor can always be safely shut down even during the conduct of experiments. Excess reactivity is that amount of reactivity that would exist if all reactivity control devices were moved to the maximum reactive condition from the point where the reactor is exactly critical ($k_{eff} = 1$) at ambient temperature with no transient poisons. Shutdown margin is the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition, with or without experiments in place, assuming all control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn.

10.2.3 Limits on SNM Included In Experiments

Technical Specifications require that SNM introduced into the reactor for experiments are limited to either: 5 grams of SNM in the form of solid samples; or, 3 grams of SNM in the form of liquid.

10.2.4 Restrictions on Types of Materials Included In Experiments

Technical Specifications require that experiments never include materials that could: contaminate the reactor pool causing corrosive action on the reactor components or experiments; cause excessive production of airborne radioactivity; or, produce an uncontained violent chemical reaction. The restriction on materials that could produce an uncontained violent chemical reaction does not prohibit the radiography of explosives specifically permitted by the Technical Specifications.

10.3 EXPERIMENT REVIEW

Technical Specification 6.5, Experiments, provides detailed requirements for the review and approval of experiments and changes to experiments.

10.4 REFERENCES

10.4.1 "Aerojet-General Nucleonics Industrial Reactor (AGNIR): Hazards Summary Report (AN-1193)," R. L. Newacheck, Project Engineer et al, September 1964.

10.4.2 "Aerojet-General Nucleonics Industrial Reactor (AGNIR): Reactor Physics Tests (AN-1527)," R. L. Tomlinson, August 1966.

11.0 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

The ARRR Technical Specifications require that ARRR maintain a radiation safety program that complies with the requirements of both 10 CFR 20 and the guidelines of American National Standard for Radiation Protection at Research Reactor Facilities, ANSI/ANS-15.11-1993. The radiation safety program includes a program for management of radioactive waste. The radiation safety program also includes the ARRR management commitment and programs to maintain exposure and release as low as reasonably achievable (ALARA) in accordance with the guidelines of ANSI/ANS-15.11-1993.

The President, Aerotest Operations, is ultimately responsible for ensuring that all operations are conducted safely and in full compliance with applicable regulations of the United States Nuclear Regulatory Commission and the State of California. Advising him are the Reactor Safeguards Committee and the Reactor Supervisor.

The ARRR Reactor Safeguards Committee (RSC) provides oversight of facility operations including reactor safeguards, radiological safety, and industrial safety. The membership consists of a Chairman, Radiological Safety Officer (RSO), Reactor Supervisor, and at least two other members who are qualified in the fields of reactor, radiological, industrial, and explosive safety. The responsibilities of the RSC include reviewing reactor operations, new procedures used in handling radioactive material, and items related to industrial safety. The committee is responsible to the President, Aerotest Operations, for ensuring that such operations are planned for execution in a safe manner and in complete accordance with the regulations of all governmental agencies. The RSC meets at least annually to review and audit ARRR operations and submits a written report to the President. The RSC chairman also verifies that the ARRR is operated within the limits of the license by performing at least one unannounced visit and audit annually.

The ARRR Radiological Safety Officer (RSO) is assigned the responsibility for implementing the radiation protection program and is directly responsible for enforcing all rules, regulations, and procedures relating to radiological safety. The RSO must review and approve all procedures and experiments involving radiological safety and is responsible for conducting routine radiation surveys. The RSO must approve all operations involving radiological safety and has the authority to stop any operation which, in his or her opinion, is unsafe. The Radiological Safety Officer reports directly to the General Manager, Aerotest Operations, and is a member of the facility Reactor Safeguards Committee. The qualifications of the RSO must include either: Certification as a Health Physicist by the Health Physics Society; or, a Bachelor's degree in Biological or Physical Science and a minimum of 2 years experience in personnel and environmental radiation monitoring programs at a nuclear facility.

All ARRR personnel are given 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 currently are given an examination on health physics practices and procedures during each requalification cycle. Each ARRR facility employee is responsible for properly implementing rules, regulations, and procedures written to assist the individual in performing the job safely. Each employee is also responsible for familiarizing himself with the potential safety problems associated with the job and implementing the ALARA program.

11.1 RADIATION PROTECTION

Radiation protection is based on an understanding of radiation sources, a radiation protection program, an ALARA program, radiation monitoring and surveying, radiation exposure control and dosimetry, and environmental monitoring.

11.1.1 Radiation Sources

Airborne Radiation Sources

The primary source of airborne radiation under normal operating conditions is the production of ^{16}N in the reactor water tank and neutron activation of air in the reactor pool tank and air filled experimental facilities.

Nitrogen-16 (^{16}N) is a gamma emitting isotope with a 7.1 second half-life that is produced by fast neutron irradiation of oxygen in the water in the reactor water tank via the reaction ^{16}O (n,p) ^{16}N. Personnel are shielded from the ^{16}N gamma by the water above the top of core. However, some ^{16}N is transported by either diffusion or convection current and reaches the reactor water tank surface before it decays. Once at the top of the pool, ^{16}N can exchange with atmospheric nitrogen, leave the water, and become airborne. Due to the short half-life, ^{16}N released to the atmosphere will not travel far from the reactor water tank before it decays. Therefore, ^{16}N is a radiological hazard only on the reactor bridge.

The amount of ^{16}N that reaches the surface of the reactor water tank before it decays depends on the transit time of the ^{16}N from the core region to the reactor water tank surface. As described in USAR Section 5.6.1, Nitrogen-16 Control, ARRR experience indicates that operating the primary cooling loop increases the transit time for ^{16}N to reach the water surface. Because ^{16}N has a 7.1 second half-life, the increase in transit time is sufficient to decreases the amount of induced radioactivity, particularly ^{16}N, released through the surface of the water of the reactor tank. The difference in radiation levels measured by the air monitor, with its intake just above the water surface, with and without the primary cooling loop in operation confirms the effectiveness of the primary cooling loop in reducing radiation levels from decay of ^{16}N above the reactor. To reduce radiation exposure to ALARA, ARRR operates the primary cooling loop continuously as soon as the reactor exceeds the point of adding heat (i.e., a few watts power) whether or not the reactor water temperature is above the automatic temperature control setting.

Argon-41 (^{41}Ar) is produced by neutron activation of the argon in air dissolved in the reactor water tank water or in the air in experiments such as glory hole or vertical tubes as discussed in USAR Chapter 10, Experiments. ARRR operating experience is consistent with the results of the analysis in Appendix E of Reference 11.4.1 that radiation exposure from activation of argon dissolved in the reactor water does not pose a significant risk to operating personnel. Control of argon activation during experiments, particularly during the use of a glory hole or the vertical tube, is addressed in USAR 10, Experiments.

The actual values for ^{41}Ar produced (i.e., the values used for the EPA COMPLY program) are based on geometry and observed values and are computed using air monitor counts, reactor operating hours, atmospheric dilution, and the number of irradiations performed. The actual value of 480 microcuries per hour or 0.13 microcuries per second, at 250 kW is much lower than the value of 0.987 microcuries per second predicted in the Hazards Summary (Reference 11.4.1).

Liquid Radioactive Sources

Small quantities of liquid radioactive waste are generated by regeneration of the demineralizer in the demineralizer loop and, infrequently, from liquids irradiated as part of sample irradiation. The radiation level from such liquids is extremely low and does not produce radiation exposure hazards. Disposal of this material is addressed in section 11.2.

Waste liquids with exempt levels of radioactivity are normally released into the sewer system every few years when the primary waste storage tank is full as discussed in section 11.2.1(4). Typical releases, based on holdup tank sample analysis, are 5.68E-12 curies gross alpha and 2.46E-10 curies gross beta in 600 gallons of waste liquid. Sampling and analysis of a liquid release is performed by a certified laboratory and the release is controlled by a permit from the local sanitary district.

Solid Radioactive Sources

Solid radioactive waste at the ARRR facility consists of spent fuel, neutron activated experiments, wastes from experiment handling (e.g., gloves, holders, plastic sheeting, tape, etc.), check and calibration sources, and waste from the demineralizer system (e.g., filters, ion exchange resin, etc.).

One fuel element was identified as leaking shortly after the ARRR initial criticality and was immediately removed from the core and shipped offsite for disposal. As of 2005, [redacted] spent fuel elements have been removed from the reactor core or transported from the facility and [redacted] spent fuel elements are currently being stored in the [redacted] fuel storage pits, [redacted]. Additionally, [redacted] highly radioactive waste or highly contaminated component is stored in the [redacted] fuel storage pits. Disposal of any high level waste, including fuel elements, will be done in coordination with the Department of Energy. As stated in USAR Chapter 1, Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982 provides that the NRC may require, as a precondition to issuing or renewing an operating license for a research or test reactor, that the applicant must have entered into an agreement with the Department of Energy (DOE) for the disposal of high-level radioactive waste and spent nuclear fuel. Aerotest Operations, Inc. has a fully executed contract with DOE that provides that DOE is obligated to take the spent fuel and/or any high-level waste for storage or reprocessing when the ARRR facility is decommissioned.

Annual generation of solid radioactive waste is minimal and consists of waste from experiment handling, demineralizer system filters, swipe samples and filter paper. This waste is stored in drums for several years where it typically decays to exempt levels and is ultimately disposed as regular waste. Otherwise, solid radioactive waste is disposed by a licensed radioactive waste disposal company.

Radioactive Material Control

ARRR has a radioactive materials control program to maintain physical control of special nuclear material and radioactive materials used or stored at ARRR facility. The RSO is responsible for maintaining custody of these materials and is required to be cognizant of the location, form and condition of all nuclear material in the possession of Aerotest Operations at all times. A complete set of accountability records are maintained for this purpose. Periodic inventories of special nuclear material and radioactive materials are conducted. Sealed sources are inventoried quarterly and leak tested every six months. Sealed sources are disposed of (or controlled as unsealed radioactive material) if their leak rate is sufficient to produce 0.005 micro- curie per swipe of their exposed surface.

11.1.2 Radiation Protection Program

As stated above, the ARRR facility operates under a radiation safety program that complies with the requirements of both 10 CFR 20 and the guidelines of American National Standard for Radiation Protection at Research Reactor Facilities, ANSI/ANS-15.11-1993. The President, Aerotest Operations, is ultimately responsible for the ARRR radiation protection program. The ARRR RSO is assigned the responsibility for implementing the radiation protection program and is directly responsible for enforcing all rules, regulations, and procedures relating to radiological safety.

Special Work Permits are required for all maintenance and other support personnel performing jobs in radiation areas to which they are not regularly assigned. The SWP must include instructions by the Radiological Safety Officer concerning the job to be performed as well as instructions concerning mandatory precautions. Personnel routinely assigned to these areas need not be covered by a SWP since they are under direct supervision and therefore familiar with the radiation safety requirements of the area.

11.1.3 ALARA Program

Technical Specifications require that Aerotest Operations establish and implement programs to maintain exposure and release as low as reasonably achievable (ALARA) in accordance with the guidelines of ANSI/ANS-15.11-1993. In order to meet the management goals for ALARA, specific administrative guides have been established.

Aerotest Operations policy is that weekly cumulative whole body exposures are limited to <100 mrem unless higher exposures are specifically approved by the Radiological Safety Officer. Employee quarterly exposures > 1.0 rem and visitor exposures of > 30 mrem must be investigated and the type and circumstances of exposure documented.

Management actively uses plant design features and personnel policy and practices to minimize both individual and collective exposure. Nearly all exposure of Aerotest personnel is the result of routine operations. Neutron radiography technicians receive the largest individual exposures when changing the aluminum film cassettes and setting up the radiography shots.

Design features intended to minimize exposure include the [redacted] thick high density concrete that surrounds the reactor pool and the active radiography area which includes [redacted] beams on top which act as a neutron shield. Additionally, when the radiography tray is in either shooting position, several pieces of shielding are fitted together to shield personnel from the N-Ray tunnel. Personnel exposure is further reduced by locating a decay box for the aluminum film cassettes within arms reach of the N-Ray tunnel. The decay box is used to allow activated aluminum film cassettes to decay for at least two half-lives before the film cassettes are changed. When film cassettes are transported to the dark room, the technician must use a wheeled cart that keeps the activated cassettes away from their bodies.

Management emphasizes with the staff that each individual must be cognizant of and apply the principles of time, distance, and shielding to reduce their exposure. Personnel are encouraged to minimize their time near and maximize their distance from the activated aluminum facility. Management has adopted practices such as placing chairs at the opposite end of the room to further encourage technicians to retreat to places with the lowest levels of radiation between radiography shots.

Individual radiation exposure is minimized by rotating the assignments of neutron radiography technicians. As of 2005, there are nine people trained to perform this task, so teams do radiography about one quarter of the time. Management trending also identified that the stature of neutron radiography technicians affects individual exposure because a taller employee bent at the waist has more distance between the tray and his torso. However, willingness to perform overtime work is the most significant contributor to differences in individual exposures. Employee doses for those performing radiography averaged 2.07 rem per year for the last five years. Employee doses for persons not routinely performing neutron radiography are almost always under 1 rem per year, with most under 0.5 rem per year.

Non-routine radiation exposure includes maintenance on the N-Ray imaging system (e.g., an aperture or a motor). Non-routine radiation exposures are always pre-planned and emphasize component substitution using spare parts. This allows maintenance and repairs to be deferred until the component has decayed and been decontaminated and allows the work to be performed in areas with low background radiation. The RSO prepares the work area (e.g., surveys, posting and contamination control measures) and briefs workers prior to and supervises the work. Radiation monitoring is performed during the work. The RSO typically performs or directly supervises the cleanup and storage of waste.

Personnel dosimetry (i.e., film badges) is monitored monthly. The RSO reviews dose reports monthly and prepares trend reports. Dose reports are posted and employees are encouraged to monitor their dose. Dose reports are also provided to the supervisor of the radiographers. Reducing dose is a primary concern for all personnel.

Although individual exposures have trended between the end of 2001 and the end of 2004 because of staff reductions, the effectiveness of the ARRR ALARA program has been demonstrated because the collective exposure has not increased commensurate with the dramatic increase in the number of radiographs made in the last several years.

11.1.4 Radiation Monitoring and Surveying

The Criticality Alarm and Area Radiation Monitor, described in USAR Chapter 7, is a fixed gamma monitor employing Geiger tube detectors [redacted]. This monitor serves as both an area radiation monitor and a criticality alarm. The area radiation monitor actuates a siren within the reactor building on high radiation level that warns personnel to evacuate the reactor room in the event of high radiation levels. [redacted]

The building gaseous effluent monitor, described in USAR Chapter 7, ensures building gaseous effluents are continuously monitored by sampling air from the area above the reactor water tank. The air intake is either just above the reactor pool, which is the normal position, or in the ceiling of the reactor room. These locations ensure gaseous fission product release will be detected.

The building particulate sampler continuously withdrawals an air sample from the reactor room and collects the particulate on filter paper. The reactor room particulate sample is counted monthly.

Gas samples are required by the Technical Specifications to be taken near the pool periodically and ^{41}Ar presence is monitored whenever the vertical tube is inserted or the glory hole facility is operated without a shield plug.

Radiation sensitive badges are located at strategic locations within the reactor building to obtain a representative sample for radiation analysis. These radiation badges are read quarterly.

Radiation detector packets containing a series of threshold detectors are placed at several locations within the reactor building for post-accident radiation analysis. Radiation detector packets containing threshold detectors are verified to be present annually.

Radiation surveys are conducted by either an operator or health physics technician quarterly.

Surface contamination swipe samples are taken at least once per quarter in areas in which radioactive material is routinely handled. Quarterly spot checks are also made in areas in which radioactive material is not handled such as offices, hallways, etc.

Reactor water tank radioactivity monitor, described in USAR Chapter 7, is a radiation monitor on the suction side of the primary coolant cleanup (demineralizer) loop which provides continuous indication of reactor water radioactivity and activates an alarm if a preset limit is exceeded. This monitor provides an early indication of fuel element failure.

Reactor water tank radioactivity is sampled and analyzed annually.

11.1.5 Radiation Exposure Control and Dosimetry

All Aerotest Operations, Inc., employees who work with, or in the vicinity of, radioactive material or other sources of radiation are provided with thermoluminescent dosimeters or film badges containing beta/gamma and neutron sensitive material. These dosimeters are changed at least once per month. More frequent badge changes are made for personnel when their known or estimated exposure exceeds 300 mrem/month. Pocket dosimeters and/or finger films are worn for all operations expected to result in total integrated exposures of > 100 mrem whole body and/or > 300 mrem to the extremities.

Visitors to Aerotest Operations, Inc. are issued pocket dosimeters if their possible exposure will exceed the limits of 10 CFR 20.1502. The daily dose accumulated is recorded in the visitors' log.

11.1.6 Contamination Control

The principal method of contamination control is the ARRR building ventilation strategy described in USAR Chapter 9, Auxiliary Systems. The ventilation system is designed to maintain pressure differentials between selected areas to achieve two objectives: confinement of airborne radioactive material to the reactor high bay area; and, exclusion of airborne radioactive material from inhabited areas, especially the control room. During any radiological event that could spread contamination or airborne radiation within the building or release it to the environment, the reactor high bay ventilation system, chemical laboratory hood blower, and rest room vent fans are shut off to reduce the potential for spread of contamination and airborne radiation

ARRR policy is that all radioactive material must be handled in well ventilated areas. Unsealed powdered radioactive material must be handled in glove boxes; radioactive material that has a potential for removable contamination must be handled in a hood with 100 feet per minute face velocity, or processed on equipment located in a controlled area and serviced by a close-capture exhaust ventilation device. The exhaust of these devices, other than that used for wet chemistry (radioactive material in liquid form), must be passed through an absolute filter before discharge to the atmosphere. All such ventilation equipment must be checked for proper operation at least quarterly when in use or prior to use.

ARRR policy is that respirators are not required for routine operations because adequate ventilation is provided for all routine tasks. Filter type or supplied air respirators are available for use when performing non-routine operations during which significant air contamination could occur.

ARRR policy is that detectable removable contamination is not allowed outside of an established contamination zone. Sealed sources are disposed of if their leak rate is sufficient to produce 0.005 micro curie per swipe of their exposed surface. For routine cleaning, the laboratory has cleaning equipment which is dedicated to use in the potentially contaminated areas, and custodial personnel use this equipment in order to prevent the possibility of spreading unidentified contamination. Floor sweepings from the laboratory area are surveyed for radioactivity before disposal.

All personnel engaged in operations with radioactive material, which is in a form that is spreadable, and personnel who are working with radioactively contaminated equipment are required to wear Aerotest Operations, Inc., provided coveralls, shoe covers, and gloves. Personnel in the vicinity of such operations may wear smocks and shoe covers in lieu of complete protective clothing. Once an item is taken into a contamination control area, it must be considered contaminated until proven otherwise. Personnel working with radioactive material with a glove box need only wear smocks and gloves. Work with fully sealed sources does not require protective clothing.

Washing hands is mandatory after working with radioactive material. Smoking, eating and drinking are prohibited in areas where protective clothing is required.

11.1.7 Environmental Monitoring

The environmental monitoring for the ARRR facility is similar to the environmental monitoring program for most TRIGA reactors. Radioactive gas and airborne particulate are the only radioactive material potentially released to the environment as a result of the routine operation of the ARRR. Monitoring is performed at the ultimate source (i.e., the areas just above the reactor water tank) by the building gaseous effluent monitor, described above and in USAR Chapter 7, and the building particulate sampler described above. Both of these instruments are required to be operable by the Technical Specifications. These monitors provide a very conservative estimate of radioactive material potentially released to the environment because the samples are collected at the point of release in the reactor high bay area where the ventilation system is designed to minimize the potential release to the environment.

In addition to the monitoring for radioactive gas and airborne particulate, radiation sensitive badges (i.e., TLDs and calcium sulfate) are located in selected areas of the Aerotest Operations, Inc. facility for routine monitoring of facility radiation levels. Currently, 19 locations are monitored.

11.2 RADIOACTIVE WASTE MANAGEMENT

11.2.1 Types of Radioactive Waste

The ARRR generates very little radioactive waste. Most of the induced radioactivity is short half life material that is stored onsite until the radioactivity decays. Four categories of radioactive waste are produced during normal operation.

(1) High level waste in the form of irradiated fuel elements.

Fuel elements no longer used in the reactor will be stored until a number of them are available for the disposal in accordance with the ARRR agreement with the DOE described earlier. ARRR has [redacted] storage locations for used fuel elements or other high level waste. They can be stored in the rack on the tank wall in the pool or they can be stored in one of the dry storage pits. ARRR currently has [redacted] fuel elements with mechanical defects stored in the pool storage rack. These elements show no sign of fission product leakage, so it is not necessary to seal them and move them to the dry storage pits. [redacted] fuel elements are currently stored in the storage pits.

(2) Low level solid waste in the form of irradiated components and devices like neutron radiography source mechanisms.

These typically measure 10's of millirems per hour at the time of removal and decay to a few millirems per hour after a few months. Also included in this category is the spent resin from the demineralizer in the primary coolant cleanup loop. This type of material is placed in plastic bags to prevent spread of contamination and stored in a lead shielded area until sufficient material is collected to warrant a radioactive waste shipment to the disposal site.

(3) Low level solid waste which may or may not be contaminated, such as swipe samples, cleaning tissues and material exposed to reactor water.

This type of material is placed in plastic bags to prevent spread of contamination and stored in a lead shielded area until sufficient material is collected to warrant a radioactive waste shipment to the disposal site or until it has decayed and is determined to be exempt from disposal requirements.

(4) Low level liquid waste from the reactor pool or decontamination of components.

The ARRR license does not require a system for collecting potentially contaminated water around the reactor. However, because the reactor building originally housed a hot cell, a radioactive liquid waste system was installed. The hot cell was removed from the building in 1969 but the liquid waste system was left in place.

Essentially, all the low level liquid waste is from the reactor pool. The pool water contains very low levels of long half-life radioactivity so it is necessary only to store the water until the short half-life material decays out.

[redacted]

When the primary tank is nearly full, samples are taken for laboratory analysis of the level and type of radioactivity. If the laboratory tests show exempt levels of radioactivity, as they normally do, then the waste is released to the sanitary sewer system after permission is received from the Central Contra Costa Sanitary District. If radioactive levels higher than the allowable for release to the sewer are encountered, the waste will be treated to separate the radioactive material or removed to a waste disposal burial site. Because of the purity of the pool water, it is unlikely that any significant radioactivity will be found in the liquid waste tank.

[redacted]

11.2.2 Radioactive Waste Disposal

All radioactive waste is collected in special marked containers obtained from a radioactive waste disposal contractor. When filled, these containers are checked, tagged and sealed, and placed in storage for ultimate disposal by a licensed radioactive waste disposal company.

(1) Solid Waste

Contaminated dry waste is placed in labeled steel drums provided for this purpose. These drums are checked for surface contamination and tagged to indicate their approximate contents.

(2) Liquid Waste

Contaminated liquid waste is stored for ultimate disposal by a licensed waste disposal company unless activity levels are verified to be low enough for discharge to the sewer system. Records are maintained of all potentially contaminated waste discharged to the sanitary sewer system. These records may be based either on holdup tank sample analysis or nuclear material accountability records. Disposal of readily soluble or dispersible radioactive material into the sanitary sewer system is permissible provided that the concentration, when diluted by the average daily quantity of industrial and domestic waste discharged at the plant, does not exceed the limits indicated in 20.2003 of the Federal and Section 30287 of the State regulations. Up to 1 curie per year of radioactive waste may be disposed of in this manner.

(3) Gaseous Waste

Gaseous radioactive waste is filtered where practical and always sampled prior to, or monitored during, discharge to the atmosphere.

11.2.3 Transportation of Radioactive Materials

All transportation of radioactive material must be approved by, or under the direct supervision of the Radiological Safety Officer. Regulations governing off-site shipment of radioactive material are contained in 49 CFR 170 through 179

11.3 RECORDS

The Radiological Safety records are maintained by the Radiological Safety Officer and include:

(1) Personnel exposure; i.e., film badge and dosimeter, bioassay results, etc.

(2) Radiation surveys; i.e., air sample results, radiation intensities, contamination levels, liquid and solid waste analysis.

(3) Environmental monitoring data.

(4) Incidents: i.e., unusual events that result, or might have resulted in radiation exposure in excess of recommended limits.

(5) Waste disposal.

(6) Radioactive material shipment.

(7) Reports to the Nuclear Regulatory Commission in accordance with 10 CFR 20.401.

(8) Reports, as required, to the State Bureau of Radiological Health in accordance with Title 17, Article 6.

11.4 REFERENCES

11.4.1 "Aerojet-General Nucleonics Industrial Reactor (AGNIR): Hazards Summary Report (AN-1193)," R. L. Newacheck, Project Engineer et al, September 1964.

11.4.2 ARRR Radiological Safety Procedures.

11.4.3 American National Standards Institute/American Nuclear Society, ANSI/ANS 15.11, "Radiation Protection at Research Reactor Facilities," ANS, LaGrange Park, Illinois, 1993.

12.0 CONDUCT OF OPERATIONS

Conduct of Operations refers to the management and administrative aspects of facility operation that are important to safety and includes requirements for the following aspects of facility operation:

1. Organizational structure including assignment of responsibility, staffing levels, and selection and training of personnel for both operating the reactor and radiation safety.
2. Oversight review and audit of facility operations by independent and qualified personnel.
3. Review and approval of activities important to safety, including experiments, through the use of written and approved procedures.
4. Actions when technical requirements are not met.
5. Reporting and record keeping.
6. Emergency Planning.
7. Security Planning.
8. Quality Assurance.
9. Operator Training and Requalification.

ARRR Technical Specifications provide detailed requirements for the conduct of operations in Section 6, Administrative Controls. The Administrative Controls established in the ARRR Technical Specifications are consistent with the guidance given in ANSI/ANS-15.1-1990, "American National Standard for the Development of Technical Specifications for Research Reactors" (Reference 12.14.1).

Requirements for emergency planning, security planning, quality assurance, operator training and requalification for the ARRR facility were developed based on the applicable regulations and government standards identified below.

12.1 ORGANIZATION

Technical Specification 6.1, Organization, establishes the requirements for the ARRR management structure and assigns responsibility for safe operation of the reactor facility including an organization chart establishing the President, Aerotest Operations, Inc. as having ultimate responsibility for safe operation of the plant.

Technical Specification 6.1 establishes minimum staffing when in the reactor operating or reactor shutdown condition including minimum requirements for reactor operators and senior reactor operators. This section also establishes the requirement that selection, training, and requalification of reactor operators must meet or exceed the requirements of American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4-1988 (Reference 12.14.2).

12.2 REVIEW AND AUDIT ACTIVITIES

Technical Specification 6.2, Review and Audit Function, establishes a requirement for oversight review and audit of facility operations by independent and qualified personnel. Technical Specification 6.2 also establishes a requirement for a Reactor Safeguards Committee (RSC) of at least five members, of whom no more than three are members of the operating organization. The RSC members are required to represent a broad spectrum of expertise in reactor technology.

The RSC is required to meet at least annually and more frequently as circumstances warrant, consistent with effective monitoring of facility activities. The RSC is responsible to determine that proposed changes in equipment, systems, test, experiments, or procedures do not involve an unreviewed safety question. The RSC must approve new and major revisions to procedures having safety significance, proposed changes in reactor facility equipment, or proposed changes to systems having safety significance.

The RSC Chairman is also responsible for proper implementation of the audit function which must include selective (but comprehensive) examination of operating records, logs, and other documents.

12.3 RADIATION SAFETY

Technical Specification 6.3, Radiation Safety, requires that ARRR maintain a radiation safety program that complies with the requirements of 10 CFR 20 and the guidelines of American National Standard for Radiation Protection at Research Reactor Facilities, ANSI/ANS-15.11-1993 (Reference 12.14.3). This program is the responsibility of a Radiological Safety Officer who reports directly to the General Manager, Aerotest Operations, Inc.

12.4 PROCEDURES

Technical specification 6.4, Procedures, requires that approved written procedures be used for each of the following activities:

(1) Startup, operation, and shutdown of the reactor.

(2) Loading, unloading, and movement of fuel within the reactor.

(3) Maintenance of major components of systems that could have an effect on reactor safety.

(4) Surveillance checks, calibrations, and inspections required by the Technical Specifications or those that may have an effect on reactor safety.

(5) Personnel radiation protection, consistent with applicable regulations or guidelines.

(6) Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity.

(7) Implementation of required plans such as emergency or security plans.

Procedures governing these activities must be reviewed by the RSC and approved by the Reactor Supervisor or designated alternates and such reviews and approvals must be documented. Technical Specification 6.3 also establishes requirements for temporary procedure changes and deviations from procedures.

12.5 EXPERIMENTS

Technical Specification 6.5, Experiments, requires that all new experiments or class of experiments or changes to experiments must be reviewed by the RSC and approved in writing by the reactor supervisor prior to initiation.

12.6 REQUIRED ACTIONS

Technical Specification 3.0 specifies the actions be taken when a Limiting Condition for Operation (LCO) or its associated Surveillance Requirement (SR) is not met. Technical Specification 6.6.1 specifies the actions be taken when a Safety Limit is exceeded. Technical Specification 6.6.2 specifies the actions be taken if an event that is required to be reported to the NRC occurs. Technical Specification 6.7.3 identifies events that must be reported to the NRC and the requirements for making these reports.

12.7 REPORTS

Technical Specification 6.7 identifies requirements for routine operating reports and for reporting changes to the facility or facility organization.

12.8 RECORDS

Technical Specification 6.8 identifies requirements for retaining records at the ARRR facility. Records include logs, data sheets, or other forms or reports.

12.9 EMERGENCY PLANNING

The ARRR emergency plan was developed to meet the requirements of 10 CFR 50.54 (q) and (r) which requires that a licensee authorized to possess and/or operate a research reactor shall follow and maintain in effect an emergency plan that meets the requirements of Appendix E to 10 CFR Part 50. The ARRR emergency plan is consistent with the guidance provided in Regulatory Guide 2.6, and ANSI/ANS 15.16, "Standard for Emergency Planning for Research Reactors," (Reference 12.14.4).

12.10 SECURITY PLANNING

ARRR Operating Licensee, Docket No. 50-228, License R-28, Section B(2), allows the ARRR facility to receive, possess, and use up to 5.0 kilograms of contained uranium 235 in connection with operation of the reactor pursuant to the Atomic Energy Act of 1954 and 10 CFR Part 70, Domestic Licensing of Special Nuclear Material. This special nuclear material is in the form of reactor fuel of contained uranium-235 enriched to less than 20 percent in the isotope uranium-235. Therefore, the ARRR possession limits fall within the definition of Special Nuclear Material of Low Strategic Significance as defined in 10 CFR 73.2 (i.e., less than 10,000 grams but more than 1,000 grams of uranium-235 contained in uranium enriched to 10 percent or more but less than 20 percent in the U-235 isotope.

The regulations in 10 CFR 73.67 (c)(1) require facilities to submit a security plan that meets the requirements of 10 CFR 73.67(f) if they possess special nuclear material of low strategic significance. In accordance with 10 CFR 73.67(f), ARRR has submitted a physical security plan that requires ARRR to meet the following: (1) store or use material only within a controlled access area; (2) monitor with an intrusion alarm or other device or procedures the controlled access areas to detect unauthorized penetrations or activities; (3) assure that a watchman or offsite response force will respond to all unauthorized penetrations or activities; and, (4) establish and maintain response procedures for dealing with threats of thefts or thefts of this material. The security plan describes the measures provided to protect special nuclear material, including details of the protective equipment and police agencies, and is thus withheld from public disclosure. The President, Aerotest Operations, Inc. is responsible for administering the security program and assuring that it is updated as required.

The NRC issued a Confirmatory Action letter, dated March 26, 2004, that required implementation of compensatory measures to address security issues identified as a result of the events of September 11, 2001. Subsequently, the NRC conducted an on-site inspection of the ARRR facility from August 16 through 19, 2004, which included a review of the implementation of these compensatory measures and the security program for the ARRR. This inspection found that the ARRR security facilities, equipment, and procedures satisfied Physical Security Plan requirements.

12.11 QUALITY ASSURANCE

The Reactor Supervisor has responsibility for quality assurance activities, and thus has the authority to identify problems, to initiate corrective actions, and to insure that corrective actions are performed. He exercises QA oversight by assuring that operating and maintenance procedures include specific requirements to assure that modification, maintenance, and calibration of safety- related systems are performed in a manner that maintains the quality and reliability of equipment. Further, experiment reviews use written requirements to assure that installation and operation of the experiment does not degrade the performance of safety equipment. Modification of safety-related equipment is planned and reviewed using formal written checklist-type procedures that assure that equipment continues to meet the original specifications.

The ARRR maintains a quality assurance program that meets the requirements of ANSI/ANS 15.8, "Quality Assurance Program Requirements for Research Reactors," (Reference 12.14.5). However, most of the reactor equipment in use in the facility does not have formal QA documentation because the ARRR was built before the QA requirements were in effect. This equipment is covered under the provisions of Section 4 of ANSI/ANS-15.8.

12.12 OPERATOR TRAINING AND REQUALIFICATION

The ARRR maintains an operator training and requalification program that meets the requirements of 10 CFR Part 55.59(c) and ANSI/ANS 15.4, "Selection and Training of Personnel for Research Reactors," (Reference 12.14.2).

12.13 ENVIRONMENTAL REPORTS

The ARRR facility was constructed in 1965, prior to National Environmental Protection Act of 1969. Therefore, no environmental assessment was performed prior to construction. On January 23, 1974 the AEC staff concluded in a memorandum addressed to D. Skovholt and signed by D. R. Miller, "that there will be no significant environmental impact associated with the licensing of research reactors or critical facilities designed to operate at power levels of 2 Mwt or lower and that no environmental impact statements are required to be written for the issuance of construction permits or operating licenses for such facilities."

As stated in USAR Chapter 1, The Facility, Aerotest Operations has a fully executed contract with DOE that provides that DOE retain title to the fuel and that DOE is obligated to take the spent fuel and/or high-level waste for storage or reprocessing when the ARRR facility is decommissioned. As described throughout this USAR, the ARRR generates very little radioactive waste. Neither routine operation nor the maximum credible accident for the ARRR result in any significant release of radioactive material from the site. Most of the induced radioactivity is short half life material that is stored onsite until the radioactivity decays.

No changes in land and water use are contemplated and emissions of radioactive materials or other effluents will not change as a result of extending the license expiration date for the ARRR facility for an additional 20 years.

12.14 REFERENCES

12.14.1 American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, "The Development of Technical Specifications for Research Reactors," ANS, LaGrange Park, Illinois, 1990.

12.14.2 American National Standards Institute/American Nuclear Society, ANSI/ANS 15.4, "Selection and Training of Personnel for Research Reactors," ANS, LaGrange Park, Illinois, 1988.

12.14.3 American National Standards Institute/American Nuclear Society, ANSI/ANS 15.11, "Radiation Protection at Research Reactor Facilities," ANS, LaGrange Park, Illinois, 1993.

12.14.4 American National Standards Institute/American Nuclear Society, ANSI/ANS 15.16, "Emergency Planning for Research Reactor Facilities," ANS, LaGrange Park, Illinois, 1982.

12.14.5 American National Standards Institute/American Nuclear Society, ANSI/ANS 15.8, "Quality Assurance Program Requirements for Research Reactors," ANS, LaGrange Park, Illinois, 1995.

13.0 ACCIDENT ANALYSES

The ARRR accident analyses address transient and accident scenarios associated with the TRIGA research reactor and the potential hazards resulting from the storage and radiography of explosive devices at the ARRR facility. The ARRR accident analyses establish each of the following:

1. Analyses of transient and accident scenarios for the ARRR facility indicate the ARRR facility is very conservatively bounded by generic and ARRR historical accident analyses that concluded that none of the credible accidents postulated pose a significant risk of failure of the cladding of any fuel element or the uncontrolled release of fission products. Additionally, the ARRR facility is very conservatively bounded by accident analyses that concluded that potential exposure to personnel outside the facility from the maximum hypothetical accident (i.e., a fuel handling accident that ruptures the cladding of a TRIGA fuel element removed from the core and not submerged in water) would be small and have little of no health significance.
2. Analyses of the potential hazards resulting from the storage and radiography of explosive devices at the ARRR facility concluded that the combination of the ARRR facility design and Technical Specification limits on the amount and type of explosive materials located in the ARRR facility: (1) does not create the possibility of a new or different kind of accident from any accident previously evaluated; and, 2) does not result in a significant increase in the probability or consequences of any transient or accident previously evaluated.
3. Based on the evaluations of the historic accident analyses for the ARRR facility (Reference 13.5.2) and generic analyses for TRIGA reactors (Reference 13.5.1), there is no credible event at the ARRR facility that could result in exposure to the general public greater than that allowed by 10 CFR 20, App. B, Table II, for unrestricted areas. This conclusion is consistent with the results of various analyses that are described in NUREG/CR-2387, "Credible Accident Analyses For TRIGA And TRIGA-Fueled Reactors (Reference 13.5.1). Any significant radiological consequences are limited to radiological exposure or contamination of Aerotest personnel or visitors within the Aerotest property boundary. Therefore, public access to the facility is restricted and minimized.

13.1 APPLICATION OF HISTORIC AND GENERIC ACCIDENT ANALYSES TO THE ARRR

The ARRR accident analyses are based on a combination of the generic TRIGA accident analyses described in NUREG/CR-2387, "Credible Accident Analyses For TRIGA And TRIGA-Fueled Reactors (Reference 13.5.1) and the historic ARRR specific accident analyses described in "Aerojet-General Nucleonics Industrial Reactor (AGNIR): Hazards Summary Report (AN-1193)" (Reference 13.5.2), originally published in 1964. The bases for the conclusion that the ARRR facility is very conservatively bounded by the conclusions of both the generic and the ARRR historic accident analyses are described below.

13.1.1 TRIGA Fuel Design

TRIGA accident analyses are based primarily on the design features of the TRIGA fuel element. These analyses are applicable to the ARRR because the ARRR uses standard TRIGA fuel elements that are identical to the fuel elements used in more than 50 TRIGA and TRIGA-fueled reactors that have been in operation since 1958. TRIGA fuel elements fall into three basic types: aluminum clad; stainless steel clad; and Fuel Lifetime Improvement Program (FLIP). The general design of the TRIGA fuel elements is the same irrespective of type. Additionally, because all TRIGA reactors use fuel elements with very similar design and construction, TRIGA reactors have limited variability in core arrangement and lattice spacing regardless of the rated thermal power of the reactor. As indicated in Reference 13.5.1, there is no impact on accident analyses between different TRIGA reactors as a result of the small variations in core arrangement and lattice spacing or the small variations in fuel element end-plug design, dimensions, or the inclusion of burnable poisons. Finally, the ARRR fuel elements are similar to other TRIGA fuel elements in that they have excellent fission product retention capability which minimizes potential for fission product release following any clad rupture.

13.1.2 TRIGA Fuel Type (Aluminum Clad versus Stainless Steel Clad)

Accident analyses that rely on limiting peak fuel temperature to prevent cladding damage from a phase change of the zirconium hydride in the fuel elements are applicable to the ARRR because the ARRR safety limit for fuel temperature and Technical Specification limit for excess reactivity are based on the more limiting aluminum clad fuel.

At initial criticality, the ARRR core included [redacted] aluminum clad TRIGA fuel elements. The design was intended to allow for more fuel elements to be added, as necessary, to compensate for fuel burn up. Because aluminum clad TRIGA fuel elements are no longer manufactured, most of the fuel elements added to the ARRR are stainless steel clad. At the present time (as of 2005), the ARRR reactor contains [redacted] active fuel elements consisting of [redacted] aluminum clad elements and [redacted] stainless steel clad elements. The ARRR does not use FLIP fuel.

Aluminum clad and stainless steel clad fuel elements both use uranium fuel that is enriched to approximately 20%. Other than the clad material, the only significant difference between aluminum clad fuel and stainless clad fuel is the hydride composition of the zirconium hydride moderator that is mixed with the fuel. The older aluminum clad fuel typically has zirconium with a 1.1% hydride ratio and the newer stainless clad fuel typically has zirconium with a 1.6% to 1.7% hydride ratio. For aluminum clad fuel with the lower zirconium hydride ratio, a peak centerline temperature as low as 535°C has the potential to cause clad damage due to a phase change of the zirconium hydride. For stainless steel clad fuel with a the higher zirconium hydride ratio, zirconium hydride phase change does not occur until fuel temperature approaches 2000°C; however, this fuel also has the potential for clad damage due to hydrogen gas pressure in the fuel element if fuel temperature reaches approximately 1100°C (Reference 13.5.1). The Technical Specification safety limit for TRIGA fuel is based on limiting maximum fuel temperature to prevent clad damage either from a phase change of the zirconium hydride or gas pressure associated with the evolution of hydrogen in the fuel element. Therefore, ARRR Technical Specifications establish 500°C as the safety limit for fuel temperature based on the more limiting aluminum clad fuel.

Accident analyses that rely on limiting peak fuel temperature to prevent cladding damage are applicable to the ARRR if peak steady state and transient power levels at the ARRR will limit fuel temperature to less than the 500°C safety limit. ARRR does not have the equipment required to pulse reactor power above the rated thermal power of 250 kW and is prohibited from pulsing power by Technical Specifications. Fuel temperature during steady state operation with the fuel submerged in the reactor water tank is significantly lower than the safety limit. Therefore, the maximum fuel temperature is bounded by the largest possible power excursion which could be caused by a step insertion of all available excess reactivity. By establishing Technical Specification limits on the ARRR core's maximum excess reactivity, the step insertion of all of the core's excess reactivity will be mitigated by the prompt, negative coefficient and the resulting power excursion will not produce sufficient energy to heat the fuel to the point where cladding failure or fuel melting will occur. As indicated in References 13.5.1 and 13.5.2, multiple TRIGA reactors have been pulsed with reactivity insertions equal to the maximum excess reactivity permitted in the ARRR core without exceeding 500°C and without clad damage. Additionally, the newer and, thus, more reactive stainless steel clad fuel elements are likely to see higher power levels during a large reactivity addition than the more depleted aluminum clad elements. The significantly higher fuel temperature safety limits for the stainless steel fuel elements provide the ARRR additional margin than that assumed in the analyses of the limiting reactivity addition event. Note that General Atomic has used a mixed core of stainless steel and aluminum clad fuel since 1960 when they were first authorized to use a limited number of stainless steel clad together with aluminum clad elements as long as fuel temperature in the mixed aluminum and stainless steel core did not exceed 550°C (versus the more conservative 500°C limit specified in the ARRR Technical Specifications).

13.1.3 Number of TRIGA Fuel Elements in the Reactor

Accident analyses described in References 13.5.1 and 13.5.2 that address fission product release and decay heat removal typically assume that a TRIGA reactor core contains between [redacted] fuel elements. At initial criticality, the ARRR core consisted of [redacted] aluminum clad TRIGA fuel elements but currently (as of 2004) consists of [redacted] fuel elements. It is anticipated that additional fuel elements will be added in the future to compensate for fuel depletion.

Accident analyses that assume the number of fuel elements in the core is less than the number of fuel elements in the ARRR core conservatively bound the consequences of similar events at the ARRR facility because using more fuel elements to produce the same amount of power will result in a lower average fission product inventory and lower average decay heat load in any individual fuel element. Newer and, thus, more reactive stainless steel clad fuel elements are likely to see higher burnup during steady state operation; however, it is expected that power distribution following insertion of a new fuel element remains within the TRIGA reactor norm and accident analysis assumption that power density of the most reactive fuel element is no more than twice the core average power density. Because the newer fuel elements with higher power density are stainless steel fuel elements with significantly higher fuel temperature safety limits, the limiting fuel element in the ARRR core will maintain substantially more margin to temperatures likely to cause clad damage than the margin assumed in the accident analyses.

13.1.4 Operating Assumptions

Generic accident analyses described in Reference 13.5.1 that address fission product release and decay heat removal assume that the reactor has operated continuously at power levels greater than 1 megawatt thermal for periods of 1 year prior to the initiation of any event. Additionally, generic accident analyses described in Reference 13.5.1 assume that the reactors are 'pulsed' to power levels many times the rated thermal power during performance of experiments.

The ARRR has a rated thermal power level of 250 kW; however, the ARRR has been operating at 180 kW or below since 1992 and at 150 kW or below since 2000. These lower steady state power levels provide the exposure times that became necessary due to the conversion to faster Kodak x-ray film for neutron radiography. Additionally, the ARRR is used only for research and neutron radiography and is shut down except during normal working hours. The ARRR is typically critical and at power for approximately 31 hours per week. Therefore, assumptions about decay heat levels and fission product inventory in the accident analyses described in both Reference 13.5.1 and Reference 13.5.2 are very conservative when applied to the ARRR facility as it is operated today.

The ARRR does not have the equipment required to pulse reactor power above the rated thermal power of 250 kW and is prohibited from pulsing power by Technical Specifications. Pulsing the reactor has the potential to make a fuel element more susceptible to cladding damage because pulsing may produce chunks or breaks in the fuel moderator. Additionally, the additional breaks in the fuel moderator caused by pulsing will cause an increase in fission product release during any subsequent clad failure. Therefore, accident analyses assumptions in Reference 13.5.1 regarding both the probability of clad failure and the fission product release caused by such a failure are very conservative when applied to the ARRR facility.

13.2 ARRR ACCIDENT EVENTS AND SCENARIOS

The ARRR Hazards Summary Report (Reference 13.5.2) evaluates various abnormal events that fall into three categories including insertion of excess reactivity, loss of cooling and fuel cladding damage.

13.2.1 Insertion of Excess Reactivity

Startup Rod Withdrawal Transient

The startup transient assumes that the reactor is subcritical and each control rod is withdrawn sequentially as permitted by Technical Specification requirements for Reactor Sequence and Interlock at the maximum rate permitted by Technical Specifications of 12 inches per minute for the safety and shim rods and 20 inches per minute for the regulating rod. Assuming that the control rod is active only over the active [redacted] of the fuel element and that the peak worth of the control rod will be about 1.6 times the average worth, the maximum ramp insertion rate will be 0.058% Δk/k per second. Based on the Technical Specification limit for maximum excess reactivity, the sequential withdrawal of all three control rods would result in a maximum reactivity change of less than 2.19% Δk/k ($3.00) over the more than three minutes required to withdraw all three control rods. This transient would produce a peak power significantly lower than the power peak produced if the entire available excess reactivity were inserted faster which is addressed below. Additionally, as described in USAR Chapter 7, Instrumentation and Control Systems, redundant and diverse reactor protection system channels will automatically terminate the reactivity addition when the reactor period becomes too short to allow the reactor operator to control the reactor power level.

Uncontrolled Rod Withdrawal

Simultaneous withdrawal of all three control rods is prevented by the Technical Specification requirements for reactor sequence and interlocks described in USAR Chapter 7, Instrumentation and Control Systems. Therefore, simultaneous withdrawal of all three control rods is highly unlikely because this transient could be caused only by sabotage of the reactor interlock system. However, based on the Technical Specification limit for maximum excess reactivity, the simultaneous withdrawal of all three control rods would result in a maximum reactivity change of less than 2.19% Δk/k ($3.00) over the more than one minute required to withdraw the control rods with a maximum transient insertion rate of 0.13% Δk/k per second. This transient would produce a much lower peak power than would result if the entire available excess reactivity were inserted faster which is addressed below. Additionally, as described in USAR Chapter 7, Instrumentation and Control Systems, redundant and diverse reactor protection system channels will automatically terminate the reactivity addition when the reactor period becomes too short to allow the reactor operator to control the reactor power level.

Step Insertion of Excess Reactivity

The stepwise insertion of all of the reactor's excess reactivity is a worst case scenario that assumes simultaneous ejection of all three control rods. Additionally, the control rod ejection must be initiated from the subcritical or just critical condition because some of the core's reactivity would be needed to increase reactor power above the critical condition.

Based on the Technical Specification limit for maximum excess reactivity, the ejection of all three control rods would result in a maximum step reactivity change of less than 2.19% $\Delta k/k$ ($3.00). Based on analyses described in References 13.5.4, 13.5.5 and 13.5.6, the ejection of all three control rods would result in a peak power of 1000 MW with a reactor period of 4.0 milliseconds for a total energy release of 18 MW seconds. The maximum fuel temperature associated with this pulse would be less than 500° C. Similarly fueled reactors have been routinely pulsed at reactivity insertions in excess of this amount with no measurable damage to the fuel. These results are consistent with and conservatively bounded by the results described in Reference 13.5.1 for similar events. Therefore, the power excursion associated with the ejection of all three control rods will not cause any significant damage to the reactor fuel. The combination of the very low probability of a stepwise insertion of all of the excess reactivity in the core and the minimal consequences of this event indicate that this event does not represent a significant threat to the public health and safety.

Fuel Loading Accident

Reference 13.5.2 postulated continuous loading of additional fuel elements to a critical or near critical core at 5 minute intervals. Although adding fuel elements to a critical reactor is highly improbable, the analysis concluded that under no conditions would the addition of a single fuel element result in a power increase of more than 150 kW allowing the insertion to be recognized and terminated before core limits are exceeded.

Poison Experiment Removal Accident

Technical Specifications requirements for shutdown margin and excess reactivity establish limits that are applicable with or without experiments in place. Additionally, Technical Specifications limit the total reactivity worth of any experiment to a value significantly lower than the Technical Specification limit for the core's total excess reactivity. Therefore, the stepwise insertion of the total reactivity worth of any experiment is bounded by the analysis for the step insertion of the core's entire excess reactivity.

13.2.2 Loss of Coolant

An instantaneous loss of all reactor core cooling due to a reactor tank rupture is considered very unlikely because the reactor tank is supported on the bottom and sides by reinforced concrete and the tank is open to the atmosphere and cannot be pressurized. Design features described in USAR Chapter 5, Reactor

Coolant Systems, minimize the potential for the occurrence of any significant loss of inventory from the reactor water tank and the redundant and diverse alarms and indications would alert the operator to a loss of coolant inventory prior to the core being uncovered.

Loss of coolant accidents are bounded by the instantaneous loss of all cooling water. Reference 13.5.1 summarizes the results of a loss of coolant analysis for the Reed College TRIGA reactor, a typical TRIGA reactor with aluminum clad elements. The results indicated that the maximum fuel temperature would be less than 150°C after infinite operation at 250 kW which was terminated by the instantaneous loss of water. At this temperature the equilibrium pressure from fission gases, entrapped air and dissociated hydrogen was reported to produce a stress of only 660 psi which is well below the yield stress of >5000 psi for the aluminum cladding at 150°C. These results very conservatively bound the results expected at the ARRR for the reasons detailed which include the significantly reduced power history at the ARRR, the larger number of fuel elements in the ARRR and the use of stainless steel clad for the fuel elements expected to have the highest power density at the time of the event. Reference 13.5.1 also indicates that for TRIGA fueled reactors even continuous operation at the steady state power levels significantly greater than the ARRR rated thermal power will not provide enough decay heat to produce fuel melting after a loss of coolant accident.

Calculations described in Reference 13.5.2 indicate that radiation levels above the reactor water tank after a complete loss of coolant would allow sufficient time for personnel to view the interior of the tank with a mirror and to make emergency repairs. Additionally, if an individual did not expose himself to the core directly, he could work for approximately 90 minutes at the top of the reactor tank after one day without being exposed to radiation in excess of approximately 1.25 rem.

13.2.3 Loss of Coolant Flow

As described in USAR Chapter 5, Reactor Coolant Systems, the ARRR is cooled primarily by natural circulation in the reactor water tank. Loss of coolant flow in the secondary cooling loop would have no detrimental effect on the core.

13.2.4 Mishandling or Malfunction of Fuel

Mishandling or malfunction of fuel while in the reactor water tank when the reactor is not critical could result in damage to the clad and release of fission products into the coolant. This would require isolation or removal of the affected fuel element. The reactor coolant would be decontaminated by using the demineralizer system and by radioactive decay. This event does not represent a significant threat to the public health and safety.

13.2.5 Experiment Malfunction

Malfunction of an experiment or an experimental apparatus could result in the release of radioactive materials either into the reactor water tank or into the reactor high bay area. However, Technical Specification limits on experiment types, reactivity values of experiments, and experimental materials (including explosives that are discussed elsewhere), limit the effect an experimental failure could have on the facility. Reactivity limits on experiments result in any conceivable failure having less of an effect than the stepwise insertion of all core excess reactivity. Limits on experimental materials including explosives and requirements for experimental facilities limit potential releases of volatile radioactive materials to less than those postulated for other events described in this chapter.

13.2.6 Loss of Normal Electrical

As described in USAR Chapter 8, Electrical Power Systems, there are no accidents or transients that depend on the availability of electrical power to protect the public heath and safety. There are no safety-related electrical power supplies because none are needed for preventing or mitigating an accident or for maintaining the facility in safe shutdown condition, even for extended periods of time.

13.2.7 External Events

As described in USAR Chapter 1, The Facility, USAR Chapter 2, Site Characteristics, and USAR Chapter 3, Design of Structures, Systems and Components, the ARRR is a Mark I TRIGA type reactor with the reactor water tank located below the ground level. The effects of any external events including earthquake or damage to the building by other causes are bounded by the events described in this chapter.

13.2.8 Mishandling or Malfunction of Equipment

As described in USAR Chapter 5, Reactor Coolant Systems, Chapter 7, Instrumentation and Control Systems, and Chapter 8, Electrical Power Systems, the ARRR reactor is designed to be fail-safe in that no system or component is needed for preventing or mitigating an accident or for maintaining the facility in safe shutdown condition, even for extended periods of time. As described in USAR Chapter 10, Experimental Facilities and Explosives, procedures and Technical Specification requirements for experiment review include consideration of chemical and explosive hazards to the reactor. Technical Specifications allow only very small amounts of special nuclear material (SNM) to be introduced into the reactor for experiments (5 grams of SNM in the form of solid samples or 3 grams of SNM in the form of a doubly encapsulated liquid). Potential reactivity effects are limited by Technical Specification limits on excess reactivity. Additionally, these small quantities of SNM ensure that any release of gaseous and volatile fission products from this material is bounded by the release associated with the maximum credible accident addressed later is this chapter.

Therefore, experiment malfunction will not result in consequences more severe than those listed in other parts of this chapter.

13.3 STORAGE AND RADIOGRAPHY OF EXPLOSIVE DEVICES AT THE ARRR FACILITY

The ARRR facility is used for research and for the neutron radiography of a wide range of military and industrial devices defined a "Low Explosives" in 27 CFR 555.202(b). The neutron radiography facility is located above the top of the reactor water tank and is integrated into the shielded enclosure directly above the vertical beam tube. The neutron radiography facility is described in USAR Chapter 10, Experiments.

Protecting operating personnel and the public from the effects of inadvertent detonation of explosive material and the release of radioactive material is accomplished using all of the following:

1) Procedural controls that minimize the potential for detonation of explosives in storage, in transport and during radiography;
2) Extensive training and experience of personnel that handle explosives also minimize the potential for detonation of explosives;
3) Technical Specification limits on the types and amounts of explosives allowed onsite, in the vicinity of the reactor and in the radiography facility that limit the consequences of an unplanned detonation;
4) The design of the neutron radiography facility which includes substantial shielding protects personnel from both radiation and blast effects and vent paths that minimize peak pressures resulting from detonation; and
5) Protective shielding surrounding the CRDs that minimizes the potential for blast effects causing rod ejection and preserves the CRD reactor shutdown capability following any detonation.

As described below, the combination of the ARRR facility design and Technical Specification limits on the amount and type of explosive materials located in various parts of the ARRR facility adequately protects the public health and safety under the maximum credible accident conditions associated with storage and neutron radiography of explosive devices.

13.3.1 Damage from Detonation of Stored Explosives

Procedural controls and the training and experience of personnel handling low explosives minimize the potential for an unplanned detonation of explosive devices stored at the ARRR facility. However, if an unplanned detonation did occur, the effects of the detonation are limited based on the Technical Specifications that limit the type and amount of explosive material that could be involved.

The ARRR neutron radiography facility is used for inspection of military and industrial explosive devices. Technical Specifications require that explosive devices at ARRR are limited to ≤1000 grains (2.28 ounces) equivalent TNT.

Additionally, Technical Specifications limit the maximum amount of explosives onsite to ≤ 50 pounds equivalent TNT. All explosive devices are received from the source in Department of Transportation (DOT) approved shipping containers in which they were transported either by air and/or by truck over public highways. The explosive devices are left in DOT approved shipping containers when not being prepared for radiography, being radiographed, or in storage in a designated explosive magazine.

13.3.2 Damage from Detonation when Handling Explosives in the Neutron Radiography Facility

The consequences of an explosive device of the type radiographed at the ARRR detonating while being handled in the neutron radiography facility was described in Reference 13.5.6. Technical Specifications limit the amount of explosive material that may be placed in the radiography facility at any time to ≤5 pounds equivalent TNT and the amount of explosive material that may be placed in the radiation field at one time to ≤1 pound equivalent TNT. Any single explosive device must contain ≤1000 grains (2.28 ounces) of explosive. As described above, tests have demonstrated that detonation of one of these devices is not likely to result in detonation of adjacent devices.

A detonation of either the 1000 grains (2.28 ounces) of explosive in one device or the total quantity of explosive permitted in the facility at one time could possibly result in significant shrapnel at the point of detonation. This type of explosion could cause injury to personnel handling the explosives and damage any equipment in the immediate area. The overpressure in the reactor high bay area would be great enough to fracture any unprotected windows joining the high bay area. To protect personnel from flying glass, the control room window is covered with a minimum of 1/4 inch Lucite, with additional reinforcing where needed. In addition, the window is laminated safety glass. No damage to the reactor is possible, as it is protected by a 20 inch thickness of concrete from the explosive loading and handling area. To minimize the potential for detonation, personnel handling the devices wear conductive footwear. Safety glasses are worn at all times when handling explosives.

13.3.3 Damage from Detonation during Neutron Radiography

The consequences of an explosive device of the type radiographed at the ARRR detonating during radiography were also described in reference 13.5.6. The maximum energy release is as follows: 70,000 calories for 1000 grains (2.28 ounces) of explosive (Technical Specification limit for a single device) and 490,000 calories for 1 pound of explosive (Technical Specification limit for amount that can be radiographed at one time). The energy released by detonation is equally divided between the shock wave and the internal blast pressure. The kinetic velocity of the fragments impinging on the inner wall of the neutron radiography facility is rapidly decelerated by the walls of the facility. Massive shielding surrounding the neutron radiography facility greatly attenuates the force of the explosion.

The CRDs are protected by a thick walled steel tube that surrounds the CRDs and a 10 inch steel I-beam structure described in USAR Chapter 10, Experiments and Explosives. Large concrete blocks 20 inches thick surrounding the area above the reactor water tank protect personnel in the immediate area and divert the shock wave upwards away from personnel. The blast is also vented out the open end or ends of the tube in which the radiography is performed. In addition, a frangible wooden section extending over the pool would be blown upward and free of the radiography facility, providing additional venting to the explosion in an upward direction inside the area protected by the concrete blocks.

The attenuated shock wave from an explosion would break the safety glass in the control room area, but fragments would he contained by a reinforced Lucite structure enclosing the window for the quantity of explosives under consideration. Damage to equipment above the reactor is expected to be minimal. Additionally, as described in USAR Chapter 5, Reactor Coolant Systems, Chapter 7, Instrumentation and Control Systems, and Chapter 8, Electrical Power Systems, the ARRR reactor is designed to be fail-safe in that no system or component is needed for preventing or mitigating an accident or for maintaining the facility in safe shutdown condition, even for extended periods of time. Therefore, this type of event does not create the possibility of a new or different kind of accident from any accident previously evaluated; and, does not result in a significant increase in the probability or consequences of any transient or accident previously evaluated.

13.3.4 Damage from Detonation to Control Rod Drive Mechanisms

Despite the low probability for inadvertent detonation of an explosive device being radiographed, the minimal damage cause by the detonation of a single device containing less than 1000 grains (2.28 ounces) of explosive and a blast shield around the CRDs that will minimize damage to the CRDs if detonation does occur, the consequences of blast damage to the CRDs from an explosion was evaluated. Although highly improbable, the bounding case for blast damage to the CRDs is that all three control rods are simultaneously ejected from the core. The evaluation of a stepwise insertion of all of the core's excess reactivity described earlier in this chapter will very conservatively bound the consequences of this event because ejection of all three control rods while at power during radiography will result in the introduction of significantly less than the Technical Specification limit on excess reactivity. As stated earlier, the power excursion associated with the ejection of all three control rods will not cause any significant damage to the reactor fuel. Additionally, as described in USAR Chapter 5, Reactor Coolant Systems, Chapter 7, Instrumentation and Control Systems, and Chapter 8, Electrical Power Systems, the ARRR reactor is designed to be fail-safe in that no system or component is needed for preventing or mitigating an accident or for maintaining the facility in safe shutdown condition, even for extended periods of time. Therefore, this type of event does not create the possibility of a new or different kind of accident from any accident previously evaluated; and, does not result in a significant increase in the probability or consequences of any transient or accident previously evaluated.

If the rod ejection resulted in physical damage to all three control rods or all three CRDs that prevented the control rods from being reinserted, reactor shutdown could still be performed by inserting supplemental poison. Alternately, fuel elements could be removed from the reactor core while the reactor was in operation without exposing personnel to excessive radiation.

Another potential consequence of an explosion in the neutron radiography facility is damage that would result in compacting the core. Although compacting the core is extremely unlikely, experimental data on the physics of ARRR have shown that reducing the hydrogen-to-uranium ratio in the core (as would be the case in a core compaction) has a strong negative effect that would drive the reactor sharply subcritical. This is consistent with the analysis described in Reference 13.5.1.

13.4 MAXIMUM HYPOTHETICAL ACCIDENT

Reference 13.5.2 describes the maximum credible accident for the ARRR is stepwise insertion of all the available excess reactivity coincident with a defect in the cladding of a fuel element that occurs either prior to or simultaneously with the reactivity addition. The analysis of this event determined that stepwise insertion of all the available excess reactivity coincident with a cladding failure, even the failure of the cladding of several fuel elements, would not constitute an undue hazard to the operating crew or the general public. Should an accident occur, it is possible that small amounts of radioactive noble gases would be dispersed from the reactor pool into the air of the reactor building, and these would decay into particulate matter. This event was determined to result in no significant risk to the public health and safety.

Subsequent to the analysis described in Reference 13.5.2, Reference 13.5.1 identified the maximum hypothetical accident for a TRIGA reactor as the fission product release directly to the atmosphere following a fuel handling accident that causes clad rupture and severely damages the fuel element. Reference 13.5.1 states that very conservative assumptions in the analysis resulted in calculated dose equivalents to the maximum exposed individual at <1 mrem to the total body from noble gases and <1.2 rem to the thyroid from radioiodines. Reference 13.5.1 states:

> The calculated dose equivalents are extremely conservative and thus represent an extreme upper limit. If such an accident occurred, exposure levels would more realistically be one to several orders of magnitude lower.
>
> Hence, even under the worst of circumstances, the potential exposure to personnel outside the facility from any credible fuel—handling accident would be small and of little or no health significance. Whole body and thyroid lifetime dose equivalents are

well within those put forth by regulatory requirements or by international bodies concerned with radiation protection (ICRP 1977, 1978; NCRP 1971, 1975, 1976).

However, the results from Reference 13.5.1 described above are for a hypothetical TRIGA fueled reactor and, based on the approach described at the beginning of this chapter, overstate the consequences of a similar event at the ARRR by at least an additional order of magnitude. Specifically, the hypothetical TRIGA reactor evaluated in Reference 13.5.1 was assumed to operate continuously for one year at one MW thermal (365 MWd). The ARRR rated thermal power is 250 kW and typically operates at ≤ 150 kW. Additionally, the ARRR is operated only during normal working hours and is typically critical approximately 1600 hours per year. The hypothetical TRIGA reactor evaluated in Reference 13.5.1 was assumed to contain only 50 fuel elements and the damaged fuel element was assumed to contain 4% of the total activity in the core. Assuming a similar power distribution as that used in Reference 13.5.1, a single fuel element from the ARRR core which, as of 2004, uses [redacted] fuel elements would contain less than 3% of the activity in the core. Therefore, it can be assumed that a fuel handling accident at the ARRR would have no significant impact on the public heath and safety.

13.5 REFERENCES

13.5.1 NUREG/CR 2387, "Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors," S.C. Hawley and R.L. Kathren, US Nuclear Regulatory Commission, April 1982.

13.5.2 "Aerojet-General Nucleonics Industrial Reactor (AGNIR): Hazards Summary Report (AN-1193)," R. L. Newacheck, Project Engineer et al, September 1964.

13.5.3 "Aerojet-General Nucleonics Industrial Reactor (AGNIR): Reactor Physics Tests (AN-1527)," R. L. Tomlinson, August 1966.

13.5.4 Letter, H. W. Davis to Donald J. Skovlolt, April 26, 1968.

13.5.5 Letter, H. W. Davis to Donald J. Skovholt, July11, 1968.

13.5.6 Letter, Richard L. Newacheck to Donald J. Skovholt, May 8, 1970.

13.5.7 W. Dobbs, E. Cohen and S. Weissman, "Blast Pressures and Impulse Loads for Use in the Design and Analysis of Explosive Storage and Manufacturing Facilities," Annals of the New York Academy of Sciences, Vol. 152, New York, 1968.

13.5.8 H. Hannagon, Explosive Technology, Fairfield, California, "I.C.C. Packaging of X-Cord," June 1969, Internal Memo, R & D-14.

13.5.9 J. Sperrazza and W. Kokinokis, "Ballistic Limits of Tissue and Clothing," Annals of the New York Academy of Sciences, Vol. 152, Art. I, New York, 1968.

14.0 TECHNICAL SPECIFICATIONS

In accordance with the requirements of 10 CFR 50.36, Technical Specifications governing the operation of the ARRR facility were developed based on the analyses and evaluations described in this Updated Safety Analysis Report (USAR). These Technical Specifications are promulgated as Appendix A to the Aerotest Radiography and Research Reactor Operating Licensee, Docket No. 50-228, License R-28.

The ARRR Technical Specifications include the following categories:

Safety Limits: Safety limits are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor must be shut down.

Limiting safety system settings (LSSS): LSSS for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is reached. If, during operation, it is determined that the automatic safety system does not function as required, the appropriate action must be taken, which may include shutting down the reactor.

Limiting conditions for operation (LCOs): LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the operator must shut down the reactor or follow any remedial action permitted by the technical specifications until the condition is met.

Surveillance requirements (SRs): SRs are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

Design features: Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered by LSSS, LCOs or SRs.

Administrative controls: Administrative controls are the provisions relating to organization and management, procedures, record keeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

The ARRR Technical Specifications were developed based on the analyses and evaluations described in this USAR. Normal operation of the reactor within the limits of the ARRR Technical Specifications will not result in offsite radiation exposure in excess of 10 CFR Part 20 guidelines. Operation within the provisions of the Technical Specifications will limit both the likelihood and consequences of malfunctions. The content, requirements, and format of the ARRR Technical Specifications are consistent with the guidelines in Reference 14.1.1.

The ARRR Technical Specification LSSS, LCOs, and SRs include a summary statement of the bases or reasons for the requirements. As specified in 10 CFR 50.36, these bases are not part of the Technical Specifications. The bases are included for information purposes only, and are not part of the Technical Specifications in that they do not constitute requirements or limitations which must be met in order to meet the Technical Specifications.

Changes to the Technical Specifications must be made in accordance with 10 CFR 50.36. Changes to the bases may be made without prior NRC approval in accordance with the requirements of 10 CFR 50.59 provided the changes do not require either of the following: a change in the Technical Specification; or, a change to the USAR that requires NRC approval in accordance with 10 CFR 50.59. Changes to the bases of the Technical Specifications shall be made under appropriate administrative controls and reviews.

14.1 REFERENCE

14.1.1 American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, "The Development of Technical Specifications for Research Reactors," ANS, La Grange Park, Illinois, 1990.

15.0 FINANCIAL QUALIFICATIONS

Information demonstrating Aerotest Operations, Inc. financial qualifications to operate the ARRR is being treated as proprietary in accordance with 10 CFR 9.17(a)(4) as "confidential commercial information."

3

AEROTEST OPERATIONS, INC.
Aerotest Radiography and Research Reactor (ARRR)

APPENDIX A:

TECHNICAL SPECIFICATIONS

Amendment 9 [Proposed R0]

DOCKET No. 50-228
License No. R-98

Aerotest Operations, Inc.
3455 Fostoria Way
San Ramon, CA 94583

TABLE OF CONTENTS

1.0 DEFINITIONS

Channel	A channel is the combination of sensor, line, amplifier, and output devices which are connected for the purpose of measuring the value of a parameter.
Channel Calibration	A channel calibration is an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include a channel test.
Channel Check	A channel check is a qualitative verification of acceptable performance by observation of channel behavior, or by comparison of the channel with other independent channels or systems measuring the same variable.
Channel Test	A channel test is the introduction of a signal into the channel for verification that it is operable.
Control Rod	A control rod is a device fabricated from neutron absorbing material that is used to establish neutron flux changes and to compensate for routine reactivity losses. A control rod can be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged.
Excess Reactivity	Excess reactivity is that amount of reactivity that would exist if all reactivity control devices were moved to the maximum reactive condition from the point where the reactor is in the reference core condition exactly critical (k_{eff} =1).
Exclusion Area	Exclusion area means that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities, including exclusion or removal of personnel and property from the area.
Experiment	Any operation, hardware, or target (excluding devices such as detectors, foils, etc.), that is designed to investigate non-routine reactor characteristics or that is intended for irradiation within the pool, on or in a beamport or irradiation facility, and is not rigidly secured to a core or shield structure so as to be a part of their design.
Experimental Facilities	Experimental facilities mean the glory hole, vertical tubes, central thimble, beam tubes, thermal column, inpool irradiation facilities, and the two triangular exposure locations.

Explosive Material	Explosive materials are any chemical compound, mixture, or device, the primary or common purpose of which is to function by explosion; or, any device containing a detonating charge that is used for initiating detonation in an explosive, or which can be caused to deflagrate when confined.
Glory Hole	A dry glory hole facility is an aluminum tube of 1.5 in. outside diameter which will fit into any fuel element hole and extends from above the top wooden reactor shield to the lower grid plate. The tube is not filled with water and is used to lower material to be irradiated through the tube into the core region.
Moveable Experiment	A moveable experiment is one where it is intended that all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating
Operable	Operable means a component or system is capable of performing its required function.
Operating	Operating means a component or system is performing its required function.
Rated Thermal Power (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 250 kW.
Reactivity (Dollars ($))	Reactivity may be expressed in units of dollars and cents where reactivity in Dollars ($) equals reactivity ($\Delta k/k$) divided by β, the fraction of delayed neutrons, which is equal to 0.0073. Therefore, a reactivity of $1.00 equals 0.73% $\Delta k/k$ and reactivity of 1.00% $\Delta k/k$ equals $1.37.
Reactivity Worth of An Experiment	The reactivity worth of an experiment is the value of the reactivity change that results from the experiment being inserted into or removed from its intended position.
Reactor Operating	Reactor operating is any condition with fuel in the reactor when the reactor is not in the reactor secured condition or the reactor shutdown condition.
Reactor Operator	A reactor operator is an individual who is licensed to manipulate the controls of a reactor.
Reactor Protection System	Reactor protection systems are those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

Reactor Secured	Reactor secured is the condition with fuel in the reactor and any fixed experiments in place, when: 1. There is insufficient moderator available in the reactor to attain criticality; or; 2. There is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection; or; 3. All of the following conditions exist: a. The console key switch is in the off position and the key is removed from the console and under the control of a licensed operator or stored in a locked storage area; and, b. Sufficient control rods are inserted so as to assure the reactor is subcritical by a margin greater than 0.73% $\Delta k/k$ ($1.00) when in reference core condition; and, c. No work is in progress involving refueling operations or maintenance of control rod mechanisms.
Reactor Shutdown	Reactor shutdown is the condition when the reactor is subcritical by at least 0.73% $\Delta k/k$ ($1.00) when in the reference core condition with the reactivity worth of all installed experiments included and the reactor is not in the reactor secured condition.
Reference Core Condition	The condition of the core when it is critical, at ambient temperature, and the reactivity worth of xenon is negligible (<0.219% $\Delta k/k$ ($0.30)), (i.e. cold, clean, and critical).
Shutdown Margin (SDM)	Shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition and with the most reactive rod in its most reactive position, and that the reactor will remain subcritical without further operator action.

2.0 SAFETY LIMIT (SL)

2.1 MAXIMUM FUEL ELEMENT TEMPERATURE

The temperature in any fuel element in the ARRR TRIGA reactor shall not exceed 500 °C under any condition of operation.

Technical Specification 6.6.1 lists the actions that shall be taken following a safety limit violation.

3.0 LIMITING CONDITIONS FOR OPERATION

3.0 GENERAL REQUIREMENTS

3.0.1 Abnormal Operation:

Upon occurrence of abnormal operation of the reactor, including its controls, safety systems and auxiliary systems, action shall be taken immediately to secure the safety of the facility and determine the cause of the abnormal behavior.

3.0.2 Defects in Fuel Elements, Control Rods, or Control Circuitry:

The reactor shall not be operated wherever there are significant defects in fuel elements, control rods, or control circuitry.

3.0.3 Actions for LCO Not Met:

LCOs must be met whenever the reactor is in the condition specified in the associated applicability statement. If an LCO is not met, the following actions shall be taken:

(1) The failure to meet an LCO shall be reported to the Reactor Supervisor or designated alternate immediately.

(2) If the LCO includes actions required when the LCO is not met, the LCO shall be considered met if the actions are completed within the specified completion time.

(3) If the LCO does not include actions required when the LCO is not met, action shall be initiated to place the reactor and facility in a condition where the LCO is no longer applicable or the reactor is in the reactor secured condition.

(4) If the LCO is applicable at all times, the Reactor Supervisor or designated alternate shall identify any additional compensatory actions required to place the reactor and facility in a safe condition. Actions to meet the LCO shall continue until the LCO is met.

(5) If the reactor is placed in the reactor shutdown condition, the reactor shall not be placed in the reactor operating condition until authorized by the Reactor Supervisor or designated alternate.

(6) The Reactor Supervisor or designated alternate shall make a determination if the event is reportable in accordance with Technical Specification 6.7.3 and, if necessary, initiate requirements of Technical Specification 6.6.2 for a reportable event.

3.1 REACTOR CORE PARAMETERS

3.1.1 Rated Thermal Power (RTP):

Reactor thermal power shall not exceed 250 kW as measured by the calibrated power channels.

Applicability: At all times.

3.1.2 Excess Reactivity:

The maximum excess reactivity with the reactor in reference core condition, with and without experiments in place, shall be 2.19% $\Delta k/k$ ($3.00).

Applicability: At all times.

3.1.3 Shutdown Margin (SDM):

SDM, with and without experiments in place, shall be ≥ 0.365% $\Delta k/k$ (≥ $0.50).

Applicability: At all times.

3.1.4 Reactivity Coefficients:

Reactivity coefficients shall be maintained such that the reactivity decrement at full power is ≥ 0.584% $\Delta k/k$ (≥ $0.80) when measured with respect to source range power level.

Applicability: Whenever the reactor is in the reactor operating condition or reactor shutdown condition.

3.1.5 Core Configuration:

Core configuration shall be maintained within the following limits:

(1) The reactor core lattice shall contain ≤ 90 cylindrical TRIGA type fuel elements.

(2) The reactor core lattice shall be loaded with ≤ 3.30 kg of U-235.

Applicability: At all times.

3.2 REACTOR CONTROL AND SAFETY SYSTEMS

3.2.1 Control Rods:

Three control rods (1 safety rod, 1 shim rod and 1 regulating rod) shall be operable.

Applicability: Whenever the reactor is in the reactor operating condition.

3.2.2 Scram Time:

The total time for complete insertion of the control rods shall be ≤ 600 milliseconds following receipt of a scram signal by the safety system channels listed in Table 3.2-1.

Applicability: Whenever the reactor is in the reactor operating condition.

3.2.3 Reactivity Insertion Rate:

The maximum rate of reactivity addition by the control rods shall be +0.080% $\Delta k/k$/second (+$0.11/second).

Applicability: Whenever the reactor is in the reactor operating condition.

3.2.4 Reactivity Insertion Monitoring:

Subcritical multiplication levels shall be plotted using input from a minimum of three instrumentation channels. Nuclear instrumentation channels listed in Table 3.2-1 may be used to satisfy requirements for one or more channels.

Applicability: Prior to criticality during experiments with the potential to affect core reactivity.

3.2.5 Reactor Sequence and Interlocks:

(1) Interlocks shall prevent withdrawal of the safety rod unless all of the following conditions exist:

(a) The master switch is in the ON position.

(b) The safety system has been reset except for those components bypassed by the Channel 1 disconnect in accordance with Table 3.2-1 footnotes (a) and (b).

(c) All four nuclear instrument channels are in the operate mode.

(d) The startup channel count rate is > 120 counts per minute unless Channel 1 is bypassed in accordance with Table 3.2-1 footnote (a).

(2) Interlocks shall prevent withdraw of the shim rod and the regulating rod unless the safety rod is withdrawn to its upper limit.

(3) Interlocks shall prevent simultaneous withdraw of the shim rod and the regulating rod.

(4) Any one interlock in LCO 3.2.5 may be removed from service for maintenance when the reactor is in the reactor shutdown condition.

Applicability: Whenever the reactor is in the reactor operating condition or reactor shutdown condition.

3.2.6 Reactor Protection System Instrumentation:

(1) Each Reactor trip system, alarm, annunciator, and associated monitoring instrument channel listed in Table 3.2-1 shall be operable.

(2) The minimum number of operable channels, limiting safety system settings, and alarm and annunciator set points shall be within the limits specified in Table 3.2-1.

Applicability: Whenever the reactor is in the reactor operating condition.

3.2.7 Reactor Monitoring System Instrumentation:

(1) Each Reactor monitoring system, alarm, annunciator, and associated monitoring instrument channel listed in Table 3.2-2 shall be operable.

(2) The minimum number of operable channels and alarm and annunciator set points shall be within the limits specified in Table 3.2-2.

Applicability: Whenever the reactor is in the reactor operating condition.

Table 3.2-1
Reactor Protection System Instrumentation

Channel	Minimum No. of Required Channels (c)(d)	Required Function(s)	Limiting Safety System Setting	Annunciator and Alarm Set Point
Neutron Flux, Channel 1	1(a)	Reactor Trip: Short Period Low Source Level	 ≥ 3 seconds ≥ 120 cpm	 ≥ 3 seconds ≥ 120 cpm
Neutron Flux, Channel 2	1	Reactor Trip: Short Period Loss of Inst Power Low Detector Voltage	 ≥ 3 seconds X ≥ 500 volts	 ≥ 3 seconds X ≥ 500 volts
Neutron Flux, Channel 3	1(b)	Reactor Trip: High Neutron Flux Low Neutron Flux Low Detector Voltage	 ≤ 98% of full scale and ≤120% RTP ≥ 5% of full scale ≥ 500 volts	 ≤ 98% of full scale and ≤120% RTP ≥ 5% of full scale ≥ 500 volts
Neutron Flux, Channel 4	1(b)	Reactor Trip: High Neutron Flux Low Neutron Flux Low Detector Voltage	 ≤ 98% of full scale and ≤120% RTP ≥ 5% of full scale ≥ 500 volts	 ≤ 98% of full scale and ≤120% RTP ≥ 5% of full scale ≥ 500 volts
Reactor Tank Water Level Low	1	Reactor Trip	≥ 16 feet above top of core	≥ 16 feet above top of core
Seismic Disturbance	1	Reactor Trip	IV on Modified Mercalli Scale max	IV on Modified Mercalli Scale max
Reactor Tank Water Temp High	1	Reactor Trip	≤ 130 °F	≤ 130 °F
Manual Scram Bar	1	Reactor Trip	Bar depressed	N/A
Master Key Switch	1	Reactor Trip	"OFF" position	N/A

(a) Channel 1 scrams are bypassed when Channel 2 exceeds a fixed setting of approximately 1×10^{-10} amps. When bypassed, Channel 1 detector is shorted and high voltage is removed.
(b) Low level scrams are bypassed on Channels 3 and 4 when Channel 2 is below a fixed setting of approximately 1×10^{-10} amps.
(c) When any one required channel is inoperable, the reactor shall be placed in the reactor secured condition within 8 hours.
(d) When more than one required channel is inoperable, the reactor shall be placed in the reactor secured condition within 1 hour.

Table 3.2-2
Reactor Monitoring System Instrumentation

Channel	Minimum No. of Required Channels	Required Function(s)	Annunciator and Alarm Set Point
Bridge Crane Location	1[a]	Annunciator/Alarm	When located off storage position
Primary Coolant Conductivity	1[b]	Control Room Indication	Indication only
Reactor Demineralizer Water Flow	1[c]	Control Room Annunciator/Alarm	≥ 4 gpm
Reactor Water Radioactivity	1[a]	Control Room Indication/ Annunciator/Alarm	≤ 20 mr/hr

(a) When a Bridge Crane Location or Reactor Water Radioactivity channel is inoperable, an alternate method of monitoring the parameter shall be established within 24 hours. The alternate monitoring method may be substituted for the required channel for a maximum of 30 days from the time the channel is inoperable. Otherwise, the reactor shall be placed in the reactor secured condition within following 4 hours.

(b) When a Primary Coolant Conductivity channel is inoperable, primary coolant conductivity shall be verified using an alternate method within 7 days and every 7 days thereafter. The alternate method may be substituted for the required channel for a maximum of 30 days from the time the channel is inoperable. Otherwise, the reactor shall be placed in the reactor secured condition within following 4 hours.

(c) When a Reactor Demineralizer Water Flow channel is inoperable, reactor demineralizer water flow shall be verified using an alternate method immediately and every 24 hours thereafter. The alternate method may be substituted for the required channel for a maximum of 30 days from the time the channel is inoperable. Otherwise, the reactor shall be placed in the reactor secured condition within following 4 hours.

3.3 COOLANT SYSTEMS

3.3.1 Reactor Tank Water Level:

The depth of water above the top of the active core shall be ≥ 16 feet.

Applicability: Whenever the reactor is in the reactor operating condition or reactor shutdown condition.

3.3.2 Primary Coolant Water Temperature:

The primary coolant bulk water temperature shall be ≥ 60 °F and ≤ 130 °F

Applicability: Whenever the reactor is in the reactor operating condition or reactor shutdown condition.

3.3.3 Primary Coolant Water Quality:

Primary Coolant Water Quality shall be maintained within the following limits:

(1) pH shall be ≤ 7.5; and,

(2) Conductivity shall be ≤ 5 μmho/cm.

Applicability: At all times.

If LCO 3.3.3 is not met, action shall be initiated within 24 hours to restore primary coolant water pH and conductivity to within the specified limits.

3.4 VENTILATION SYSTEMS

3.4.1 Control Room:

The control room shall be maintained at a positive pressure with respect to the reactor room.

Applicability: Whenever the reactor is in the reactor operating condition *or reactor shutdown condition and during movement of* irradiated fuel.

3.4.2 Reactor Building Circulation Fans:

All air conditioning systems and building circulating fans in or normally open to the high bay area shall have the capability to be shut off from a single control in the control room.

Applicability: Whenever the reactor is in the reactor operating condition or reactor shutdown condition and during movement of irradiated fuel.

3.5 CRITICALITY ALARM, RADIATION AND RADIOACTIVE EFFLUENT MONITORING

3.5.1 Criticality Alarm and Area Radiation Monitoring:

(1) A fixed gamma monitor employing Geiger tube detectors shall be located on the wall connecting the control room and the reactor room within the limits specified in Table 3.5-1 for the Criticality Alarm and Area Radiation Monitor.

(a) This monitor shall annunciate through an automatic monitoring system; and,

(b) This monitor shall actuate a siren within the reactor building on high radiation level.

(2) An appropriate number of radiation sensitive badges shall be placed at strategic locations within the reactor building to obtain a valid representative sample for radiation analysis.

Applicability: At all times.

3.5.2 Gaseous Effluent Monitoring:

A gas sample shall be continuously withdrawn from the roof vent above the reactor, or from the vicinity of the reactor pool, and pumped through a radioactive gas detector chamber within the limits specified in Table 3.5-1 for the Building Gaseous Effluent Monitor.

Applicability: Whenever the reactor is in the reactor operating condition or reactor shutdown condition and during movement of irradiated fuel.

3.5.3 Particulate Effluent Monitoring:

A particulate sample shall be continuously withdrawn from the reactor room and collected on filter paper.

Applicability: Whenever the reactor is in the reactor operating condition or reactor shutdown condition and during movement of irradiated fuel.

3.5.4 Post Accident Radiation Monitoring:

Two radiation detector packets containing a series of threshold detectors shall be placed at strategic locations within the reactor building for post-accident radiation analysis.

Applicability: At all times.

3.5.5 Portable Radiation Monitoring Instruments:

The following portable radiation monitoring instruments shall be operable and onsite as follows:

(1) A portable survey instrument for measuring beta-gamma dose rates in the range of 0.01 mr/hr to 50 r/hr.

(2) A portable instrument for measuring fast and thermal neutron dose rates from 0.1 mr/hr to 1.0 r/hr.

Applicability: At all times.

3.5.6 Radioactive Effluent Limits:

Normal releases of radioactive effluents from reactor operation shall not exceed 10 CFR 20 limits.

Applicability: At all times.

Table 3.5-1
Criticality Alarm, and Area and Effluent Radiation Monitoring Instrumentation

Channel	Minimum No. of Required Channels	Required Function(s)	Annunciator and Alarm Set Point
Criticality Alarm and Area Radiation Monitor	1(a)(b)	Annunciator/ Alarm for LCO 3.5.1	≤ 10 mr/hr
Building Gaseous Effluent Monitor	1(a)(b)	Annunciator/ Alarm for LCO 3.5.2	≤ 2 mr/hr

(a) When the Criticality Alarm and Area Radiation Monitor or Building Gaseous Effluent Monitor is inoperable, alternate methods of monitoring the parameter shall be established within 8 hours. The alternate monitoring method may be substituted for the required channel for maximum of 7 days from the time the channel is inoperable. Otherwise, the reactor shall be placed in the reactor secured condition within following 4 hours.

(b) When the Criticality Alarm and Area Radiation Monitor or Building Gaseous Effluent Monitor is inoperable during movement of a fuel element, movement of the fuel element shall be stopped immediately. This action shall not preclude completion of movement of a fuel element to a safe position.

3.6 EXPERIMENTS

3.6.1 Evaluation and Approval of Experiments:

Experiments shall be evaluated and approved in accordance with the *requirements of Technical Specification* 6.5.1.

Applicability: Prior to insertion of the experiment into the reactor.

3.6.2 Reactivity Limits during Experiments:

Evaluations of experiments performed in accordance with the requirements of Technical Specification 6.5.1 shall demonstrate that the following reactivity limits will be maintained during the experiment when in its most reactive condition:

(1) The reactivity worth of any single independent experiment shall be ≤ 1.46% Δk/k (≤ $2.00). If such experiments are connected or otherwise related so that their combined reactivity could be added to the core simultaneously, their combined reactivity shall be ≤ 1.46% Δk/k ($2.00).

(2) The reactivity worth of any single independent experiment not rigidly fixed in place shall be ≤ 0.73% Δk/k (≤ $1.00). If such experiments are connected or otherwise related so that their combined reactivity could be added to the core simultaneously, their combined reactivity worth shall be ≤ 0.73% Δk/k (≤ $1.00).

(3) Experiments having moving parts shall be designed to have reactivity insertion rates < 0.073% Δk/k /second (< $0.10/second) except that moving parts worth < 0.0365% Δk/k /second (< $0.05/second) may be oscillated or removed at higher frequencies.

(4) Limits in LCO 3.1.2, Excess Reactivity, and LCO 3.1.3, Shutdown Margin, shall not be exceeded including the reactivity worth of the experiment when evaluated in the most reactive condition.

Applicability: Prior to initiation of the experiment.

3.6.3 Special Nuclear Material (SNM) included in Experiments:

SNM introduced into the reactor for experiments shall be limited as follows:

(1) The amount of special nuclear material contained in an experiment shall be limited to either of the following:

(a) 5 grams of SNM in the form of solid samples; or,

(b) 3 grams of SNM in the form of liquid.

(2) Liquid special nuclear materials shall be doubly encapsulated.

Applicability: At all times.

3.6.4 Materials Used during Experiments:

No experiment shall be performed involving materials which could:

(1) Contaminate the reactor pool causing corrosive action on the reactor components or experiments;

(2) Cause excessive production of airborne radioactivity; or,

(3) Produce an uncontained violent chemical reaction.

Applicability: At all times.

3.6.5 Explosive Materials in Experiments:

Solid explosive materials may be brought into the facility for the purpose of being radiographed in the neutron radiography facilities located above the pool provided that the following conditions are met:

(1) The maximum amount of explosive material contained in devices that may be placed in the radiography facilities at a time shall be limited to five pounds equivalent TNT.

(2) Explosive material in the radiation field at one time shall be limited to 1 pound equivalent TNT.

(3) Explosive material contained in long devices shall be limited to 0.5 pound equivalent TNT per foot.

(4) The explosive devices shall be subjected to a total exposure not to exceed 3×10^{11} neutrons/cm^2 and 3×10^3 roentgens of gammas.

(5) Explosive devices that have or provide a thrust in a definite direction upon ignition shall be positioned so as to be aimed away from the reactor and components.

Applicability: At all times.

3.6.6 Configuration and Potential Failure Mechanisms:

(1) No experiment shall be installed in the reactor in such a manner that it could shadow the nuclear instrumentation detectors.

(2) No experiment shall be installed in the reactor in such a manner that a failure could interfere with the insertion of a reactor control rod.

(3) Experiments shall not be performed involving equipment whose failure could result in fuel element damage.

Applicability: At all times.

3.6.7 Glory Hole Facility:

One dry glory hole facility may be located in any reactor core position with the following restrictions:

(1) The glory hole shall accept capsules with a maximum diameter of 1.35 inches.

(2) The glory hole shall be purged with CO_2 to prevent the formation of excessive amounts of Argon-41 as follows:

(a) A glory hole facility operated with a shield plug shall be purged prior to each insertion of the shield plug.

(b) A glory hole facility operated without a shield plug shall be purged with CO_2 continuously.

Applicability: Whenever a glory hole facility is located in any reactor core position and the reactor is in the reactor operating condition.

3.6.8 Vertical Tubes:

Vertical irradiation tubes with diameters up to 6 inches may be attached to the thermal column with the following restrictions:

(1) Each vertical tube shall be purged with CO_2 continuously.

(2) Gas samples shall be taken near the pool continuously and Argon-41 presence monitored when the vertical tube is inserted.

Applicability: Whenever a Vertical Tubes is attached to the thermal column and the reactor is in the reactor operating condition.

3.6.9 Other Irradiation Facilities:

The following Irradiation Facilities may be used within the limitations specified:

(1) The central 7 fuel elements of the reactor may be removed from the core and a central irradiation facility installed provided the cross-sectional area of the facility does not exceed 16 square inches.

(2) Two triangular exposure facilities are available which shall allow the insertion of circular experiments to a maximum of 2.35 inches diameter or triangular experiments to a maximum of 3.0 inches on a side.

(3) Irradiation capsules in the shape of dummy fuel elements shall have a maximum inner void volume of 34 cubic inches in the active fuel region.

Applicability: At all times.

3.6.10 Large Component Irradiation Box:

The large component irradiation box shall not be installed in the reactor tank.

Applicability: At all times.

3.6.11 Pneumatic Transfer Facility:

A pneumatic transfer facility shall not be installed in any reactor core position.

Applicability: At all times.

3.7 FUEL STORAGE AND TRANSFER

3.7.1 Fuel Handling Tools:

The fuel handling tools shall be secured with a locking mechanism under the cognizance of the Reactor Supervisor when not authorized for use.

Applicability: At all times except when the fuel handling tools are in use.

3.7.2 Fuel Storage in the Reactor Tank:

Fuel may be stored in fuel storage racks located in the reactor tank within the following limitation:

(1) Fuel in the reactor tank but not part of the reactor core lattice shall be stored in a geometric array where $k_{eff} \leq 0.9$ for all conditions of moderation and reflection using light water.

Applicability: At all times.

3.7.3 Fuel Storage in the Fuel Storage Pits:

Fuel may be stored in the fuel storage pits within the following limitations:

(1) Each fuel storage pit shall hold ≤19 fuel elements and ≤ 700 grams of U-235.

(2) Each fuel storage pit containing a fuel element shall be secured with a lock and chain except during fuel transfer operations involving the fuel element being stored in the associated pit.

Applicability: At all times fuel is stored in any fuel storage pit.

3.7.4 Fuel not in the Reactor and not in Storage:

Not more than one fuel element shall be allowed in the facility which is not in storage or in the core lattice.

Applicability: At all times.

3.7.5 Fuel in Shipping Containers:

(1) Fuel may be contained in an approved fuel shipping container within the limitations of LCO 3.7.4.

(2) When an approved fuel shipping container is used, then the licensed limitations on k_{eff} for the container shall apply.

Applicability: *Whenever a fuel shipping container is in use.*

3.7.6 Fuel Transfer in the Reactor Tank:

All fuel transfers in the reactor tank shall be conducted by a minimum staff of three members, and shall include a licensed Senior Operator and a licensed Operator. The staff members shall monitor the operation using appropriate radiation monitoring instrumentation.

Applicability: *During fuel transfers in the reactor tank.*

3.7.7 Fuel Transfer Outside the Reactor Tank:

Fuel transfers outside the reactor tank but within the facility shall be supervised by a licensed Operator.

Applicability: *During transfers of fuel outside the reactor tank.*

3.8 REACTOR FACILITY

3.8.1 Reactor Building Alarm:

The reactor building alarm shall be operable and monitored continuously.

Applicability: At all times.

3.8.2 Explosive Material at the Reactor Facility:

Explosive materials may be brought into the facility for the purpose of being radiographed provided that the following conditions are met:

(1) Explosive material shall be stored in designated areas within the reactor facility.

(2) Individual explosive devices shall be limited to 1000 grains equivalent TNT.

(3) Only solid or encased explosive materials may be brought into the facility.

(4) Any explosive device containing loose explosive powders shall be completely encased.

(5) The maximum quantity of explosive material that may be possessed at one time shall be limited to 50 pounds equivalent TNT.

(6) Unshielded high frequency generating equipment shall not be operated within 50 feet of any explosive device.

(7) Personnel handling the explosive devices shall be trained and familiar with the devices being radiographed.

(8) Personnel handling the explosive devices shall use special equipment such as non-sparking tools and shoes, protective clothing, safety shields, and grounded benches as required for the explosives being handled.

(9) Only the explosive devices that do not exceed a maximum of ten pounds equivalent TNT and that will be radiographed within 8 hours, including preparation time, may be removed from the storage areas at one time. This restriction does not apply to explosives packaged for shipment.

(10) A daily accountability log shall be maintained to show the amount of explosive material in the reactor facility and shall contain a description of the explosive material, and the location within the facility (e.g., storage, radiography facility, or shipping dock).

Applicability: At all times.

4.0 SURVEILLANCE REQUIREMENTS

4.0 GENERAL REQUIREMENTS

4.0.1 Surveillance Requirements:

SRs shall be met whenever the associated LCO is required to be met.

4.0.2 Surveillance Requirement Frequency:

SRs shall be performed at the frequency specified for the SR as follows:

(a) Five-year (interval not to exceed six years)
(b) Biennial (interval not to exceed two and one-half years)
(c) Annual (interval not to exceed fifteen months)
(d) Semiannual (interval not to exceed seven and one-half months)
(e) Quarterly (interval not to exceed four months)
(f) Monthly (interval not to exceed six weeks)
(g) Weekly (interval not to exceed ten days)
(h) Daily (must be done during the calendar day).

4.1 REACTOR CORE PARAMETERS

4.1.1 Rated Thermal Power (RTP):

(1) A reactor thermal power calibration shall be performed annually. Neutron Flux Channel 3 and Channel 4 shall reflect the results of this thermal power calibration.

4.1.2 Excess Reactivity:

(1) Excess reactivity shall be verified to be within the limits of LCO 3.1.2 annually and following any significant change to the core or any control rod.

4.1.3 Shutdown Margin (SDM):

(1) SDM shall be verified to be within the limits of LCO 3.1.3 annually and following any significant change to the core or any control rod.

4.1.4 Reactivity Coefficients:

(1) The reactivity decrement at full power shall be verified to be within the limits of LCO 3.1.4 annually and following any significant change to the core or any control rod.

4.1.5 Core Configuration:

(1) A core inventory shall verify that the reactor core lattice contains ≤ 90 TRIGA type *fuel elements prior to reactor startup following the addition of* any fuel elements to the reactor core.

(2) An administrative review shall verify that the reactor core lattice contains ≤ 3.30 kg of U-235 prior to reactor startup following the addition of any fuel elements to the reactor core.

(3) *A minimum of 20% of the fuel elements in the reactor core lattice shall be* examined by visual inspection annually.

(4) Each fuel element in the core shall have been examined by visual inspection within the previous five years.

4.2 REACTOR CONTROL AND SAFETY SYSTEMS

4.2.1 Operable Control Rods:

Free movement of each control rod for insertion is verified by the performance of SR 4.2.2 (1).

4.2.2 Scram Time:

(1) The control rod scram times shall be measured and verified to be within the limits of LCO 3.2.2 semiannually and following any significant change to the core or any significant work on the control rods or the control rod drive system.

4.2.3 Reactivity Insertion Rate:

(1) The reactivity worth of each control rod shall be measured annually and following significant changes to the core or any control rod.

(2) The withdrawal speed of each control rod shall be measured semiannually and the maximum rate of reactivity addition verified to be within the limits of LCO 3.2.3.

4.2.4 Reactivity Insertion Monitoring:

(1) Subcritical multiplication levels shall be plotted in accordance with the requirements of LCO 3.2.4 during experiments from start of control rod withdrawal until reactor criticality.

4.2.5 Reactor Sequence and Interlocks:

(1) The function of each of the following interlocks shall be verified annually:

(a) Safety rod withdrawal is prevented by each of the following:

(i) The master switch is not in the ON position; or,

(ii) The safety system has not been reset; or,

(iii) Any one of the four nuclear instrument channels not in the OPERATE mode; or

(iv) The neutron flux channel 1 count rate is ≤120 counts per minute unless bypassed when channel 2 exceeds a fixed setting of approximately 1×10^{-10} amps.

(b) Withdrawal of the shim rod and withdrawal of the regulating rod is prevented until the safety rod is withdrawn to its upper limit.

(c) Simultaneous withdrawal of the shim rod and the regulating rod is blocked.

4.2.6 Reactor Protection System Instrumentation:

(1) Channel checks of each of the applicable channels listed in Table 3.2-1 shall be performed prior to exiting the reactor secured condition on the first reactor startup each day.

(2) Channel tests of each of the applicable channels listed in Table 3.2-1 shall be performed semiannually.

(3) Channel calibrations of each of the applicable channels listed in Table 3.2-1 shall be performed prior to initial use.

4.2.7 Reactor Monitoring System Instrumentation:

(1) Channel checks of each of the applicable channels listed in Table 3.2-2 shall be performed prior to exiting the reactor secured condition on the first reactor startup each day.

(2) Channel tests of each of the applicable channels listed in Table 3.2-2 shall be performed semiannually except for the reactor water radioactivity channel which shall be tested prior to exiting the reactor secured condition on the first reactor startup each day.

(3) Channel calibrations of each of the applicable channels listed in Table 3.2-2 shall be performed prior to initial use except for the reactor water radioactivity channel which shall be calibrated biennially.

4.3 COOLANT SYSTEMS

4.3.1 Reactor Tank Water Level:

Periodic verification of reactor water tank level is addressed in SR 4.2.6.1, SR 4.2.6.2, and SR 4.2.6.3.

4.3.2 Primary Coolant Water Temperature:

Periodic verification of primary coolant water temperature is addressed in SR 4.2.6.1, SR 4.2.6.2, and SR 4.2.6.3.

4.3.3 Primary Coolant Water Quality:

(1) The pH and conductivity of the primary coolant shall be measured verified to be within the limits of LCO 3.3.3 monthly.

(2) The radioactivity level of the primary coolant shall be analyzed annually.

4.4 VENTILATION SYSTEMS

4.4.1 Control Room:

(1) The control room shall be verified to be at a positive air pressure with respect to the reactor room quarterly.

4.4.2 Reactor Building Circulation Fans:

(1) Proper operation of the emergency shutoff from the control room of all air conditioning systems and building circulating fans in or normally open to the high bay area shall be verified quarterly.

4.5 CRITICALITY ALARM, RADIATION AND RADIOACTIVE EFFLUENT MONITORING

4.5.1 Criticality Alarm and Area Radiation Monitoring:

(1) A channel test of the channel listed in Table 3.5-1 shall be performed quarterly. This test shall include the reactor building siren and receipt of the alarm by the automatic alarm monitoring company.

(2) A channel calibration of the channel listed in Table 3.5-1 shall be performed biennially.

(3) Radiation sensitive badges placed at strategic locations within the reactor building shall be analyzed quarterly.

4.5.2 Gaseous Effluent Monitoring:

(1) A channel test of the channel listed in Table 3.5-1 shall be performed prior to exiting the reactor secured condition on the first reactor startup each day.

(2) A channel calibration of the channel listed in Table 3.5-1 shall be performed biennially.

4.5.3 Particulate Effluent Monitoring:

(1) The reactor room particulate sampler shall be verified to be operating daily each day the reactor is operated.

(2) The reactor room particulate sample shall be counted monthly.

4.5.4 Post Accident Radiation Monitoring:

(1) The two radiation detector packets containing threshold detectors shall be verified to be in place annually.

4.5.5 Portable Radiation Monitoring Instruments:

(1) Channel calibrations of each portable radiation monitoring instrument required by LCO 3.5.5 shall be performed quarterly.

4.5.6 Radioactive Effluent Limits:

(1) Ar-41 production shall be determined annually.

(2) Administrative verification that normal releases of radioactive effluents from reactor operation did not exceed 10 CFR 20 limits shall be performed annually.

4.6 EXPERIMENTS

4.6.1 Evaluation and Approval of Experiments:

(1) The reactor operator shall verify that experiments have been evaluated and approved in accordance with the requirements of Technical Specification 6.5.1 prior to insertion of the experiment into the reactor.

4.6.2 Reactivity Limits during Experiments:

No requirements. SR 4.6(1) will ensure that the requirements of LCO 3.6.2 are met.

4.6.3 Special Nuclear Material (SNM) included in Experiments:

(1) The reactor operator shall verify that experiments that introduce SNM into the reactor meet the requirements of LCO 3.6.3 prior to insertion of the experiment into the reactor.

4.6.4 Materials Used during Experiments:

No requirements. SR 4.6(1) will ensure that the requirements of LCO 3.6.2 are met.

4.6.5 Explosive Materials in Experiments:

(1) Administrative controls shall verify, as applicable, that the requirements of LCO 3.6.5 are being implemented.

4.6.6 Configuration and Potential Failure Mechanisms:

No requirements. SR 4.6(1) will ensure that the requirements of LCO 3.6.2 are met.

4.6.7 Glory Hole Facility:

(1) Continuous gas samples shall be taken near the glory hole opening and Argon-41 presence shall be monitored when the glory hole facility is operated without a shield plug.

4.6.8 Vertical Tubes:

(1) Verification that each vertical tube is being purged continuously with CO_2 shall be performed daily.

(2) Gas samples shall be taken near the pool continuously and Argon 41 presence shall be monitored when the vertical tube is inserted.

4.6.9 Other Irradiation Facilities:

No requirements. SR 4.6(1) will ensure that the requirements of LCO 3.6.2 are met.

4.7 FUEL STORAGE AND TRANSFER

4.7.1 Fuel Handling Tool:

(1) Verification that the fuel handling tools are secured with a locking mechanism under the cognizance of the Reactor Supervisor when not authorized for use shall be performed weekly.

4.7.2 Fuel Storage in the Reactor Tank:

(1) Verification that any planned addition or movement of fuel in the reactor tank that is not part of the reactor core lattice will not result in a geometric array where keff is > 0.9 for all conditions of moderation and reflection using light water shall be made prior to any addition or movement of fuel in the reactor tank.

4.7.3 Fuel Storage in the Fuel Storage Pits:

(1) Verification that any planned addition or movement of fuel stored in each fuel storage pit will not result in > 19 fuel elements or > 700 grams of U-235 in each fuel storage pit shall be made prior to any addition or movement of fuel in the fuel storage pits.

(2) Verification that each fuel storage pit containing a fuel element is secured with a lock and chain except during fuel transfer operations involving the fuel element being stored in the associated pit shall be performed quarterly.

4.7.4 Fuel not in the Reactor and not in Storage:

(1) Verification that any movement or receipt of fuel at the facility will not result in more than one fuel element in the facility which is not in the reactor core lattice or in storage shall be made prior to any movement or receipt of fuel at the facility.

4.7.5 Fuel in Shipping Containers:

(1) Verification that an outgoing fuel shipping container is approved shall be made prior to use.

4.7.6 Fuel Transfer in the Reactor Tank:

No requirements.

4.7.7 Fuel Transfer Outside the Reactor Tank:

No requirements.

4.8 REACTOR FACILITY

4.8.1 Reactor Building Alarm:

(1) The Reactor Building Alarm system shall be tested monthly.

4.8.2 Explosive Material at the Reactor Facility:

(1) Administrative controls that implement the requirements of LCO 3.8.2 shall be reviewed annually.

5.0 DESIGN FEATURES

5.1 SITE AND FACILITY DESCRIPTION

5.1.1 A steel, locked perimeter fence shall surround the ARRR facility forming an exclusion area. The reactor and associated equipment are located within an exclusion area. The restricted area, as defined in 10 CFR 20, shall consist of the entire exclusion area.

5.1.2 The minimum distance from the center of the reactor pool to the boundary of the exclusion area fencing shall be 50 feet.

5.1.3 The principal activities carried on within the exclusion area shall be those associated with the operation of the reactor.

5.1.4 The reactor shall be housed in a steel building.

5.1.5 Reactor building ventilation shall be achieved by gravity ventilators located on the roof of the building.

5.1.6 An alarm system shall be installed to detect unauthorized entry into the reactor building.

5.2 REACTOR COOLANT SYSTEM

5.2.1 The reactor shall be cooled by the pool water which circulates by natural convection.

5.2.2 The pool water shall be cooled by a pumped cooling system consisting of a primary and secondary loop.

5.2.3 The water purity shall be maintained by a mixed bed demineralizer.

5.3 REACTOR CORE AND FUEL

5.3.1 *Reactor Core:*

(1) The reactor core shall consist of standard TRIGA fuel elements, graphite reflector elements, 3 control rods and guide tubes, a neutron source, and irradiation facilities.

(2) Fuel elements may be added to compensate for fuel burn up within the limits in Technical Specification 3.1.5.

(3) The overall reflector elements' dimensions shall be the same as the fuel elements.

(4) Core and fuel design shall ensure that reactivity coefficients are maintained as follows:

(a) The bath temperature coefficient and the prompt fuel temperature coefficient are negative at all operating temperatures.

(b) The coolant void coefficient is negative across the active core.

(c) Maximum in-core operating void is 10% of the coolant core volume as defined by a cylinder bounded by the grid plates.

5.3.2 Reactor Fuel:

The TRIGA fuel elements used in the ARRR shall have the following nominal characteristics at fabrication:

Fuel alloy:	uranium-zirconium hydride
Enrichment:	≤ 20 wt % U-235
Cladding:*	0.030 inch thick (aluminum), or 0.020 inch thick (stainless steel)
Fuel matrix:	8 wt. % (aluminum), or ≤ 12 wt. % (stainless steel)
Hydrogen-to-zirconium atom ratio in the ZrH_x:	1.0 to 1.1 % (aluminum), or 1.6 to 1.7 % (stainless steel)
Fuel loading:	36 to 38 grams (aluminum), or ≤ 55 grams (stainless steel)

5.3.3 Control Rods:

(1) The reactor shall have three control rods: safety, shim, and regulating.

(2) Each control rod shall contain boron carbide as a neutron poison, sealed in an aluminum or stainless steel tube.

5.4 FISSIONABLE MATERIAL STORAGE

5.4.1 The fuel storage pits located in the floor of the reactor room shall each accommodate a maximum of 19 fuel elements and a maximum of 700 grams of U-235 in storage racks. The fuel storage pits may be dry or flooded with water.

5.4.2 Fuel may be stored in fuel storage racks located in the reactor tank. Fuel in the reactor tank but not part of the reactor core shall be stored in a geometric array where keff is ≤ 0.9 for all conditions of moderation and reflection using light water.

5.4.3 Fuel may be contained in an approved fuel shipping container within the limitations in LCO 3.7.4, Fuel not in the Reactor and not in Storage, and LCO 3.7.5, Fuel in Shipping Containers.

5.4.4 A fuel handling tool shall be used for transferring fuel elements of low radioactivity between the storage pits and the reactor. A shielded fuel transfer cask shall be used for the transfer of highly radioactive fuel elements between the storage pits and the reactor.

5.5 EXPERIMENTAL FACILITIES

5.5.1 Neutron Radiography Facility:

(1) The beam tube shall consist of a two-section tapered tube having a rectangular cross section.

(2) The upper and lower sections of the beam tube shall be equipped with a fill and drain line.

(3) All components contacting the pool water shall be fabricated from aluminum or stainless steel.

(4) The beam catcher shield shall consist of a movable radiation shield.

5.5.2 Thermal Column:

(1) The thermal column shall be composed of a three-foot cube of graphite encased in aluminum containing five rows of 1.5 inch diameter irradiation holes placed 6 inches apart with seven holes per row.

(2) The thermal column shall be positioned remotely on steel locating pins immediately adjacent to the reactor core.

(3) Slotted beams shall be provided to allow experiments to be attached directly to the thermal column.

(4) Vertical irradiation tubes, having diameters up to 6 inches, may be attached to the thermal column.

5.5.3 Other Irradiation Facilities:

(1) The central 7 fuel elements of the reactor may be removed from the core within the limits specified in LCO 3.6.9(1).

(2) Two triangular exposure facilities are available which shall allow the insertion of circular experiments to a maximum of 2.35 inches diameter or triangular experiments to a maximum of 3.0 inches on a side.

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION

6.1.1 Structure:

The management and operation of the reactor facility shall be in accordance with the organizational structure indicated in Figure 6.1.

6.1.2 Responsibility:

(1) Responsibility for the safe operation of the reactor facility shall be with the chain of command established in Figure 6.1.

(2) Individuals at all management levels shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to all requirements of the operating license and Technical Specifications.

(3) Responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon appropriate qualifications.

6.1.3 Staffing:

(1) The minimum staffing when in the reactor operating or reactor shutdown condition shall be:

- (a) A licensed Reactor Operator in the control room.
- (b) A second designated person present at the facility able to carry out prescribed written instructions. Unexpected absence for as long as two hours to accommodate a personal emergency may be acceptable provided immediate action is taken to obtain a replacement.
- (c) A designated Senior Reactor Operator shall be readily available on call. Readily available on call means an individual who:
  - (i) Has been specifically designated and the designation known to the operator on duty.
  - (ii) Keeps the operator on duty informed of where he may be rapidly contacted and the phone number.
  - (iii) Is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g., 30 minutes or within a 15-mile radius).

(2) Events requiring the presence of a Senior Reactor Operator at the facility include:

   (a) Initial startup and approach to power.

   (b) All fuel or control rod relocations within the reactor core region.

   (c) Relocation of any in-core experiment with reactivity worth $\geq 0.73\%\ \Delta k/k$ ($\geq \$1.00$).

(3) A list of reactor facility personnel by name and telephone number shall be readily available in the control room for use by the operator. The list shall include:

   (a) Management personnel;

   (b) Radiation safety personnel; and,

   (c) Other operations personnel.

6.1.4 Selection and Training of Personnel:

The selection, training, and requalification of reactor operators shall meet or exceed the requirements of American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4-1988, Sections 4-6.

6.2 REVIEW AND AUDIT FUNCTION

6.2.1 Reactor Safeguards Committee (RSC) Composition and Qualifications:

(1) The RSC shall be composed of not less than five members, of whom no more than three are members of the operating organization.

(2) RSC members and alternates shall be appointed by and report to the President, Aerotest Operations, Inc.

(3) RSC members shall collectively represent a broad spectrum of expertise in reactor, radiological, industrial, and explosive safety.

(4) Individuals on the RSC may be either from within or outside the operating organization.

(5) RSC members from outside the operating organization shall include a chairman and at least two individuals who are qualified in appropriate technologies listed in Technical Specification 6.2.1(3).

(6) RSC members from within the operating organization shall include:

(a) A Reactor Supervisor, who:

(i) Shall have responsibility of the reactor facility;

(ii) Shall be responsible to the General Manager, Aerotest Operations, Inc. in all matters pertaining to reactor operations and to the Technical Specifications;

(iii) Shall have a Bachelor's degree in Engineering or Physical Science;

(iv) Shall have a minimum of 4 years experience in the operation of a nuclear facility during which competence in supervision and reactor operations shall have been demonstrated; and,

(v) Shall hold a Senior Reactor Operator's license for the facility.

(b) A Radiological Safety Officer, who:

(i) Shall review and approve all procedures and experiments involving radiological safety;

(ii) Shall enforce rules, regulations and procedures relating to radiological safety;

(iii) Shall conduct routine radiation surveys;

(iv) Shall be responsible to the General Manager, Aerotest Operations, Inc.;

(v) Shall have a Bachelor's degree in Biological or Physical Science;

(vi) Shall have a minimum of 2 years experience in personnel and environmental radiation monitoring programs at a nuclear facility;

(vii) May be certified as a Health Physicist by the Health Physics Society in lieu of the education and experience requirements given above.

(7) Qualified and approved alternates may serve in the absence of regular members.

6.2.2 RSC Charter and Rules:

(1) The RSC shall meet on the call of the chairman and shall meet at least annually and more frequently as circumstances warrant, consistent with effective monitoring of facility activities.

(2) Official RSC action requires a quorum of not less one-half of the membership where the operating staff does not constitute a majority.

(3) The dissemination, review, and approval of RSC meeting minutes shall be conducted in a timely manner.

6.2.3 RSC Review Function:

(1) The following items shall be reviewed by the RSC:

(a) Determinations that proposed changes in equipment, systems, test, experiments, or procedures do not involve an unreviewed safety question.

(b) All new procedures and major revisions thereto having safety significance, proposed changes in reactor facility equipment, or systems having safety significance.

(c) All new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity.

(d) Proposed changes in Technical Specifications, license, or charter.

(e) Violations of Technical Specifications, license, or charter.

(f) Violations of internal procedures or instructions having safety significance.

(g) Operating abnormalities having safety significance.

(h) Reportable occurrences listed in 6.7.2.

(i) Audit reports.

(2) A written report or minutes of the findings and recommendations of the RSC shall be submitted to the President, Aerotest Operations, Inc. and to the RSC members in a timely manner after the review has been completed.

6.2.4 Audit Function:

(1) The RSC Chairman shall be responsible for proper implementation of the audit function.

(2) The audit function shall include selective (but comprehensive) examination of operating records, logs, and other documents.

(3) Discussions with cognizant personnel and observation of operations should be used also as appropriate.

(4) In no case shall the individual immediately responsible for the area perform an audit in that area.

(5) The following items shall be audited:

(a) Facility operations for conformance to the Technical Specifications and applicable license or charter conditions, at least once per calendar year (interval between audits not to exceed 15 months);

(b) The requalification program for the operating staff, at least once every other calendar year (interval between audits not to exceed 30 months);

(c) The results of actions taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operations that affect reactor safety, at least once per calendar year (interval between audits not to exceed 15 months);

(d) The reactor facility emergency plan and implementing procedures at least once every other calendar year (interval between audits not to exceed 30 months).

(6) Deficiencies uncovered that affect reactor safety shall immediately be reported to the President, Aerotest Operations, Inc.

(7) A written report of the findings of the audit shall be submitted to the President, Aerotest Operations and to the RSC members within three months after the audit has been completed.

6.3 RADIATION SAFETY

6.3.1 Radiation Safety Program:

The radiation safety program shall comply with the requirements of 10 CFR 20 and the guidelines of American National Standard for Radiation Protection at Research Reactor Facilities, ANSI/ANS-15.11-1993.

6.3.2 Radiological Safety Officer:

(1) The Radiological Safety Officer shall be assigned the responsibility for implementing the radiation protection program.

(2) The Radiological Safety Officer shall report to the General Manager, Aerotest Operations, Inc.

6.4 PROCEDURES

6.4.1 Written Procedures:

(1) Written procedures shall be prepared, reviewed, and approved for each of the following prior to initiating these activities:

(a) Startup, operation, and shutdown of the reactor.

(b) Loading, unloading, and movement of fuel within the reactor.

(c) Maintenance of major components of systems that could have an effect on reactor safety.

(d) Surveillance checks, calibrations, and inspections required by the Technical Specifications or those that may have an effect on reactor safety.

(e) Personnel radiation protection, consistent with applicable regulations or guidelines. These procedures shall include management commitment and programs to maintain exposure and release as low as reasonably achievable (ALARA) in accordance with the guidelines of ANSI/ANS-15.11-1993.

(f) Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity.

(g) Implementation of required plans such as emergency or security plans.

6.4.2 Procedure Approval:

Procedures shall be reviewed by the RSC and approved by the Reactor Supervisor or designated alternates and such reviews and approvals shall be documented in a timely manner.

6.4.3 Procedure Changes:

(1) Temporary procedures which do not change the intent of previously approved procedures may be utilized on approval by a Senior Reactor Operator and one other qualified individual. Such procedures shall be subsequently reviewed by the RSC.

(2) Substantive changes to the previous procedures shall be made effective only after documented review by the RSC and approval by the Reactor Supervisor or designated alternates.

6.4.4 Deviation from Procedures:

Temporary deviations from the procedures may be made by a Senior Reactor Operator in order to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported to the Reactor Supervisor or designated alternates.

6.5 EXPERIMENTS

6.5.1 Experiment Review and Approval:

(1) All new experiments or class of experiments shall be reviewed by the RSC and approved in writing by the Reactor Supervisor or designated alternates prior to initiation.

The documentation of experiments shall include the following:

(a) The purpose of the experiment;

(b) A description of the experiment; and

(c) An analysis of the possible hazards associated with the performance of the experiment.

(2) The description, review, and approval shall be documented as shown in the Aerotest Experiment Type Review (AETR) form. Supporting material may be attached to that form as required.

(3) The safety review for each experiment type shall include but not be limited to those items delineated on the AETR form and any other items which could credibly reduce reactor safety or subject personnel to unacceptable hazards.

(4) The safety review for each experiment type shall evaluate the experiment in the most reactive condition.

6.5.2 Experiment Performance:

(1) It shall be the responsibility of the Reactor Supervisor to delegate responsibilities as required to ensure that experiments fall within approved experiment types and restrictions.

(2) It shall be the responsibility of all persons to immediately notify appropriate personnel if an experiment does not fall within established limits.

(3) The console operator shall review all information and shall exercise final control over that particular experiment with respect to reactor operation.

(4) The Reactor Supervisor shall be notified immediately if a deviation occurs or a problem arises.

6.5.3 Changes to Experiments:

(1) Substantive changes to previously approved experiments shall be made only after review by the RSC and approved in writing by the Reactor Supervisor or designated alternates.

(2) Minor changes that do not significantly alter the experiment may be approved by the Reactor Supervisor or designated alternates.

6.6 REQUIRED ACTIONS

6.6.1 Action to be Taken in Case of Safety Limit (SL) Violation:

(1) The reactor shall be placed in the reactor secured condition, and reactor operations shall not be resumed until authorized by the Nuclear Regulatory Commission (NRC).

(2) The SL violation shall be promptly reported to the Reactor Supervisor or designated alternates.

(3) The SL violation shall be reported to the NRC.

(4) A SL violation report shall be prepared. The report shall describe the following:

(a) Applicable circumstances leading to the violation including, when known, the cause and contributing factors;

(b) Effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public; and,

(c) Corrective action to be taken to prevent recurrence.

(5) The report shall be reviewed by the RSC and any follow-up report shall be submitted to the NRC when authorization is sought to resume operation of the reactor.

6.6.2 Action to be Taken for Reportable Event described 6.7.3:

(1) Any event that may be reportable in accordance with Technical Specification 6.7.3 shall be reported to the Reactor Supervisor or designated alternate immediately; and,

(2) Action shall be initiated immediately to place the reactor and facility in a condition where the reportable condition no longer exists or the reactor is in the reactor secured condition. Actions shall continue until the reportable condition is corrected.

(3) If the reactor is placed in the reactor shutdown condition, the reactor shall not be placed in the reactor operating condition until authorized by the Reactor Supervisor or designated alternate.

(4) Reactor Supervisor or designated alternate shall make a determination if the event is reportable in accordance with Technical Specification 6.7.3.

(5) Event shall be reviewed by the RSC at their next scheduled meeting.

6.7 REPORTS

6.7.1 Operating Reports:

(1) Routine operating reports covering the operation of the facility during the previous calendar year shall be submitted to the NRC and shall include the following:

(a) The Annual Summary of Changes, Tests, and Experiments as required by 10 CFR 50.59.

(b) The Annual Occupational Exposure Reports as required by 10 CFR 20.2206.

(c) The Annual Material Status Reports as required by NUREG/BR-0007

(d) The Nuclear Material Transaction Reports as required by NUREG/BR-0006.

6.7.2 Changes to the Facility or Organization:

(1) A written report to the NRC shall be made within 30 days of any of the following:

(a) Permanent changes in the facility organization involving the President, Aerotest Operations, Inc. or the General Manager.

(b) Significant changes in the transient or accident analysis as described in the Safety Analysis Report.

6.7.3 Reportable Events:

(1) There shall be a report no later than the following working day by telephone and confirmed in a written report by FAX or similar conveyance to the NRC that describes the circumstances of the event within 14 days of any of the following:

(a) Violation of a Safety Limit;

(b) Release of radioactivity from the site above allowed limits;

(c) Any of the following:

(i) Operation with actual safety-system settings for required systems less conservative than the Limiting Safety System Settings specified in the Technical Specifications.

(ii) Operation in violation of LCO established in the Technical Specifications unless prompt remedial action is taken.

(iii) A reactor protection system component malfunction which renders or could render the reactor protection system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdowns. (Note: Where components or systems are provided in addition to those required by the Technical Specifications, the failure of the extra components or systems is not considered reportable provided that the minimum number of components or systems specified or required by the Technical Specifications perform their intended reactor safety function.)

(iv) An unanticipated or uncontrolled change in reactivity $\geq 0.73\%\ \Delta k/k$ ($\geq$ \$1.00). Reactor trips are excluded.

(v) Abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or containment boundary (excluding minor leaks) where applicable, which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both.

(vi) An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.

6.8 RECORDS

6.8.1 Records Format:

Records may be in the form of logs, data sheets, or other suitable forms. The required information may be contained in single or multiple records, or a combination thereof.

6.8.2 Records to be Retained for Five Years:

(1) The following records shall be retained for a period of at least five years or for the life of the component involved if less than five years:

(a) Normal reactor facility operation (but not including supporting documents such as checklists, log sheets, etc., which shall be maintained for a period of at least one year);

(b) Principal maintenance operations;

(c) Reportable occurrences;

(d) Surveillance activities required by the Technical Specifications;

(e) Reactor facility radiation and contamination surveys where required by applicable regulations;

(f) Experiments performed with the reactor;

(g) Fuel inventories, receipts, and shipments;

(h) Approved changes in operating procedures;

(i) Records of meeting and audit reports of the RSC.

6.8.3 Records to be Retained for at Least One Certification Cycle:

Records of retraining and requalification of licensed operators shall be maintained at all times the individual is employed or until the license is renewed.

6.8.4 Records Retained for Life of the Facility:

(1) The following records shall be retained for the life of the reactor facility:

(a) Gaseous and liquid radioactive effluents released to the environs;

(b) Radiation exposure for all personnel monitored;

(c) Drawings of the reactor facility.

(2) Applicable annual reports, if they contain all of the required information, may be used as records in this section.

6.8.5 Facility Specific Records:

(1) In addition to those records required under the facility license and applicable regulations, the following records shall be kept for the life of the reactor facility:

(a) Reactor operating records, including power levels;

(b) Records of in-pile irradiations;

(c) Records showing radioactivity released or discharged into the air or water beyond the effective control of the licensee as measured at the point of such release or discharge; and,

(d) Records of emergency reactor scrams, including the reasons for emergency shutdowns.

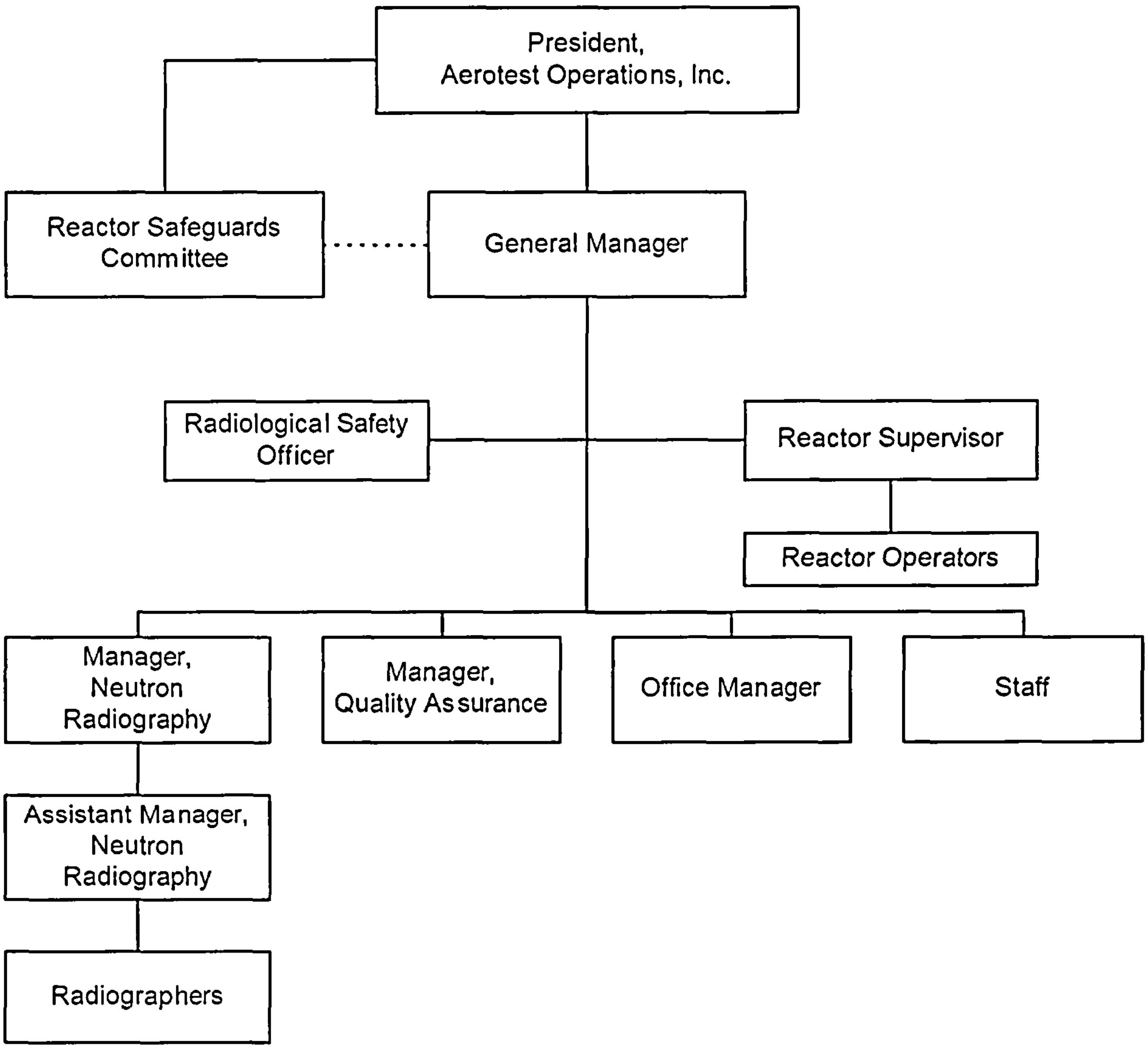
(2) The following records shall be kept for 7 years when explosive materials are to be irradiated or radiographed:

(a) The type and quantity of explosive material;

(b) The date, time of day, and length of exposure;

(c) The total neutron and gamma exposure levels.

Figure 6.1
Aerotest Operations, Inc.
Organization Chart



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AEROTEST OPERATIONS, INC.

Aerotest Radiography and Research Reactor (ARRR)

TECHNICAL SPECIFICATION BASES

Revision 0 [Proposed R0]

DOCKET No. 50-228
License No. R-98

Aerotest Operations, Inc.
3455 Fostoria Way
San Ramon, CA 94583

Table of Contents

2.0 SAFETY LIMITS (SLs)

Safety limits for nuclear reactors are defined in 10 CFR 50.36 as limits for important process variables that are necessary to reasonably protect the integrity of physical barriers that guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor must be shut down and the Nuclear Regulatory Commission (NRC) *must be notified. Operation of the reactor must not be resumed until authorized* by the NRC.

The SLs for the ARRR are designed to protect the integrity of the cladding of the TRIGA fuel elements. The fuel element cladding is the primary barrier for the prevention of the uncontrolled release of radioactivity because the ARRR design does not include either containment or a reactor pressure vessel.

SL 2.1 MAXIMUM FUEL ELEMENT TEMPERATURE

SL 2.1 for the ARRR specifies that the temperature in any fuel element in the ARRR shall not exceed 500 °C under any condition of operation. Because the ARRR uses both aluminum clad and stainless steel clad fuel elements, this SL is based on the older aluminum clad fuel which is more limiting. The older aluminum clad fuel typically has zirconium with a 1.1% hydride ratio and the newer stainless clad fuel typically has zirconium with a 1.6% to 1.7% hydride ratio. For aluminum clad fuel with the lower zirconium hydride ratio, a peak centerline temperature as low as 535°C has the potential to cause clad damage due to a phase change of the zirconium hydride. For stainless steel clad fuel with a the higher zirconium hydride ratio, zirconium hydride phase change does not occur until fuel temperature approaches 2000°C; however, this fuel also has the potential for clad damage due to hydrogen gas pressure in the fuel element if fuel temperature reaches approximately 1100°C. Therefore, ARRR uses 500°C as the safety limit for fuel temperature based on the more limiting aluminum clad fuel.

Fuel temperature during steady state operation with the fuel submerged in the reactor water tank is significantly lower than the safety limit. Therefore, the maximum fuel temperature is bounded by the largest possible power excursion which could be caused by a step insertion of all available excess reactivity or by a loss of cooling.

As described in the ARRR Updated Safety Analyses Report (USAR), Chapter 13, by limiting on the ARRR core's maximum excess reactivity, the step insertion of all of the core's excess reactivity will be mitigated by the prompt, negative coefficient. The resulting power excursion will not produce sufficient energy to heat the fuel to the point where cladding failure or fuel melting will occur.

As described in the ARRR USAR, Chapters 5 and 13, by establishing Technical Specification limits on the ARRR core's maximum rated thermal power level, decay heat is limited so that even an instantaneous loss of all cooling water will not cause the fuel temperature to increase to the point where cladding failure or fuel melting will occur.

Technical Specification 6.6.1 lists the actions that shall be taken following a safety limit violation.

3.0 LIMITING CONDITIONS FOR OPERATION (LCOS) GENERAL REQUIREMENTS

Limiting conditions for operation are defined in 10 CFR 50.36 as the lowest functional capability or performance levels of equipment required for safe operation of the facility. LCO 3.0.1 through LCO 3.0.3 establish the general requirements applicable to all Technical Specifications and apply at all times, unless otherwise stated.

LCO 3.0.1 ABNORMAL OPERATION

LCO 3.0.1 requires that prudent compensatory action, which may include reactor shutdown, be initiated immediately to ensure the safety of the facility in response to any abnormal operation or behavior of the reactor, including its controls, safety systems, and auxiliary systems. The abnormal operation or behavior must be investigated in a timely manner commensurate with the potential safety significance. Timely and appropriate notifications about the abnormal operation must be made within the Aerotest Operations organization and appropriate approvals must be obtained for the corrective action and continued operation of the reactor.

LCO 3.0.2 DEFECTS IN FUEL ELEMENTS, CONTROL RODS, OR CONTROL CIRCUITRY

LCO 3.0.2 specifies that the reactor shall not be operated wherever there are significant defects in fuel elements, control rods, or control circuitry. This LCO recognizes that situations or conditions may exist that are not specifically addressed by Technical Specifications but that could place the ARRR outside of design requirements or analysis assumptions. The reactor shall not be operated when any defect in fuel elements, control rods, or control circuitry could place the ARRR outside of design requirements or analysis assumptions. This LCO requires timely and appropriate notifications within the Aerotest Operations organization about defects that could place the ARRR outside of design requirements or analysis assumptions.

LCO 3.0.3 ACTIONS FOR LCO NOT MET

LCOs must be met whenever the reactor is in the condition specified in the associated applicability statement. If an LCO is not met, the ARRR may be operating outside of design requirements or analysis assumptions specified in the Updated Safety Analysis Report (USAR). The Technical Specifications identify the compensatory actions required upon discovery of a failure to meet an LCO.

Some LCOs identify a specific action required to be performed when that LCO is not met. The action specified is typically a requirement to use an alternate method to perform the function required by the LCO not being met and usually includes a limit on how long the alternate method may be used. Alternately, the action specified may be a requirement that the LCO be met within a specified completion time which is intended to provide an opportunity to correct the reason for the LCO not being met before shutting down the reactor is required.

If an LCO includes actions required when the LCO is not met, the LCO is considered met if the actions are completed within the completion time. If the action specified is a requirement that the LCO be met within a specified completion time and the LCO is not met within that completion time, then the LCO is treated as if no actions were specified. If the LCO is met or is no longer applicable prior to expiration of the completion time, completion of the required action is not required unless otherwise stated in the LCO.

If an LCO does not include actions required when the LCO is not met, action must be initiated to place the reactor and facility in a condition where the LCO is no longer applicable or the reactor is in the reactor secured condition. Except when the reactor operator determines that immediate reactor shutdown is warranted, this action is satisfied if a normal reactor shutdown is initiated within one hour after the discovery of a failure to meet the LCO and the reactor shutdown is completed using normal operating procedures. The one hour delay for the initiation of a normal reactor shutdown is allowed to permit the appropriate investigation and consultation prior to initiation of a shutdown. If the LCO is met or is no longer applicable prior to completion of the reactor shutdown, completion of the required action is not required unless otherwise stated in the LCO.

If an LCO is applicable at all times, placing the reactor in the reactor secured condition may not ensure that the ARRR is operating within design requirements or analysis assumptions specified in the USAR. Therefore, the Reactor Supervisor or designated alternate must identify any additional compensatory actions required to place the reactor and facility in a safe condition until the LCO is met. Actions to meet the LCO must continue until the LCO is met.

4.0 SURVEILLANCE REQUIREMENTS GENERAL REQUIREMENTS

Surveillance Requirements (SR) are defined in 10 CFR 50.36 as requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

SR 4.0.1 SURVEILLANCE REQUIREMENTS

SRs must be met whenever the associated LCO is required to be met. Unless otherwise specified, an SR must be completed successfully before the reactor can be placed in a condition where the associated LCO is applicable. If an SR is not met or is not performed at the required frequency, it must be assumed that the associated LCO is not met.

SR 4.0.2 SURVEILLANCE REQUIREMENT FREQUENCY

The required frequency for performing an SR is typically expressed as daily, weekly, monthly, etc. SRs should be performed at the specified frequency (i.e., weekly, monthly, etc.). SR 4.0.2 provides an allowance that may be used for extending the interval between performances of SRs to facilitate flexible scheduling. The allowance in SR 4.0.2 is not intended to be used to routinely increase the interval between required performances of an SR.

3.1 REACTOR CORE PARAMETERS

LCO 3.1.1 RATED THERMAL POWER (RTP)

LCO 3.1.1 specifies that the total reactor core heat transfer rate to the reactor coolant must not exceed 250 kW as measured by the calibrated power channels.

The 250 kW limit for RTP ensures that the ARRR is operated within the assumptions used to determine that TRIGA fuel elements have the ability to tolerate credible events without damage. As discussed in USAR Chapter 13, loss of coolant accidents are bounded by the instantaneous loss of all cooling water. USAR Chapter 13, Accident Analyses, describes the results of a loss of coolant analysis for the Reed College TRIGA reactor, a typical TRIGA reactor with aluminum clad elements. The results indicated that the maximum fuel temperature would be less than 150°C after infinite operation at 250 kW which was terminated by the instantaneous loss of water. These results very conservatively bound the results expected at the ARRR because of the significantly reduced power history at the ARRR, the larger number of fuel elements in the ARRR, and the use of stainless steel clad for the fuel elements which are expected to have the highest power density at the time of the event. Therefore, the 250 kW limit for thermal power provides a very high degree of assurance that SL 2.1 will not be exceeded following a loss of coolant accident. Because a sudden and complete loss of reactor coolant will not result in fuel damage or release of fission products at the ARRR, there is no requirement either for an emergency core cooling system or for containment or confinement of the reactor.

SR 4.1.1 requires comparing the results of a calorimetric heat balance calculation to the indication from neutron flux channels 3 and 4 and adjusting channel output accordingly. Operating experience indicates that an annual thermal power calibration of the neutron flux channels is sufficient to maintain required accuracy. In addition, the control room operator periodically monitors and compares the output from the redundant channels. This monitoring ensures that LCO 3.1.1 is being met and comparing the output of the redundant channels ensures that any significant deviation between channels is promptly identified and investigated.

LCO 3.1.2 EXCESS REACTIVITY

LCO 3.1.2 specifies that the excess reactivity with the reactor in reference core condition, with and without experiments in place, must be ≤ 2.19% Δk/k ($3.00). Excess reactivity is defined in Section 1.0 of the Technical Specifications.

The limit for the amount of excess reactivity available in the reactor ensures that the ARRR is operated within the assumptions used to determine that TRIGA fuel elements have the ability to tolerate credible events without damage. As discussed in USAR Chapter 13, Accident Analyses, the stepwise insertion of all of the reactor's excess reactivity is a worst case scenario that assumes simultaneous ejection of all three control rods. Additionally, the control rod ejection must be initiated from the subcritical or just critical condition because some of the core's reactivity would be needed to increase reactor power above the critical condition. The limit for maximum excess reactivity ensures that the ejection of all three control rods would result in a maximum step reactivity change of less than 2.19% Δk/k ($3.00). Based on analyses described in USAR Chapter 13, the ejection of all three control rods would result in a peak power of 1000 MW with a reactor period of 4.0 milliseconds for a total energy release of 18 MW seconds. The maximum fuel temperature associated with this pulse would be less than 500° C and would not violate SL 2.1.

SR 4.1.2 requires that excess reactivity be verified to be within the limits of LCO 3.1.2 annually and following any significant change to the core or any control rod. The requirement to verify excess reactivity following any significant change to the core or any control rod ensures that any changes to core configuration that could result in excess reactivity being introduced into the core is identified promptly. The verification of excess reactivity following a change to the core configuration should be made as soon as possible following the first reactor startup following the change in core configuration. Otherwise, annual verification of excess reactivity is sufficient because core reactivity changes slowly in the conservative direction due to fuel depletion absent any changes to the core configuration.

LCO 3.1.3 SHUTDOWN MARGIN (SDM)

LCO 3.1.3 specifies that the SDM, with and without experiments in place, must be ≥ 0.365% Δk/k (≥ $0.50). SDM is defined in Section 1.0 of the Technical Specifications.

The limit for the minimum SDM available in the reactor ensures that the ARRR reactor can be made subcritical starting from any permissible operating condition by inserting the control rods assuming the most reactive control rod remains fully withdrawn and that the reactor will remain subcritical without further operator action. Maintaining SDM above the required minimum ensures that the reactor will be shut down and remain shut down following a reactor scram.

SR 4.1.3 requires that SDM be verified to be within the limits of LCO 3.1.3 annually and following any significant change to the core or any control rod. The requirement to verify SDM following any significant change to the core or any control rod ensures that any changes to core configuration that could reduce SDM are identified promptly. The verification of SDM following a change to the core configuration should be made as soon as possible following the first reactor startup following the change in core configuration. Otherwise, annual verification of SDM is sufficient because core reactivity changes slowly in the conservative direction due to fuel depletion absent any changes to the core configuration.

LCO 3.1.4 REACTIVITY COEFFICIENTS

LCO 3.1.4 specifies that core reactivity coefficients must be maintained such that the reactivity decrement at full power is ≥ 0.584% Δk/k (≥ $0.80) when measured with respect to source range power level. The reactivity decrement is the amount of reactivity that must be added to increase reactor power from the source range to RTP.

The limit for the minimum reactivity decrement ensures that negative reactivity coefficient resulting from the zirconium-hydride moderator (U-ZrH) is within the limits assumed in the accident analyses for terminating increases in reactor power due to the inadvertent insertion of positive reactivity. The ARRR uses TRIGA fuel-moderator elements which consist of a solid homogenous alloy of uranium fuel and zirconium-hydride moderator (U-ZrH). These unique fuel elements provide the TRIGA reactor with a large prompt negative temperature coefficient. This means that any increase in power heats both the fuel and the moderating material in the fuel simultaneously, and immediately the fuel becomes less effective (decreases in reactivity). This causes the reactor to return automatically to normal operating levels within milliseconds. This characteristic is basic to the TRIGA reactor core, and permits safe, steady state operation and pulsing to high power levels without the use of mechanical or electronic control devices to return the reactor to a safe power level.

SR 4.1.4 requires that the reactivity decrement be verified to be within the limits of LCO 3.1.4 annually and following any significant change to the core or any control rod. The requirement to verify the reactivity decrement following any significant change to the core or any control rod ensures that any changes to core configuration that could reduce the reactivity decrement is identified promptly. The verification of the reactivity decrement following a change to the core configuration should be made as soon as possible following the first reactor startup following the change in core configuration. Otherwise, annual verification of the reactivity decrement is sufficient because the fuel moderator ration does not change significantly in a one year period and core reactivity changes slowly in the conservative direction due to fuel depletion absent any changes to the core configuration.

LCO 3.1.5 CORE CONFIGURATION (INCLUDES INVENTORY AND FUEL INSPECTION)

LCO 3.1.5 specifies that the core may include no more than 90 cylindrical TRIGA type fuel elements and no more than 3.30 kg of U-235 at any time.

The ARRR design was intended to allow for more fuel elements to be added to the core, as necessary, to compensate for fuel burn up. There are [redacted] grid positions in the core lattice. At initial criticality, the ARRR core included [redacted]3 aluminum clad TRIGA fuel elements. As of 2004, the ARRR contains [redacted] active fuel elements consisting of [redacted] aluminum clad elements and [redacted] stainless steel clad elements. [redacted] core lattice positions are occupied by graphite dummy elements, 3 by control rods, one by a neutron startup source, and one by a removable glory hole facility. The limit on the number of fuel elements and the amount of U-235 that may be contained in the core is an administrative limit intended to ensure periodic NRC review and approval as fuel elements are added to the core.

SR 4.1.5(1) requires a core inventory verify that the reactor core lattice contains no more than 90 TRIGA type fuel elements prior to reactor startup following the addition of any fuel elements to the reactor core.

SR 4.1.5(2) requires an administrative review to verify that the reactor core lattice contains ≤ 3.30 kg of U-235 prior to reactor startup following the addition of any fuel elements to the reactor core.

SR 4.1.5(3) requires that a minimum of 20% of the fuel elements in the reactor core lattice shall be examined by visual inspection annually.

SR 4.1.5(4) requires that each fuel element in the core have been examined by visual inspection within the previous five years. Typically, an administrative review performed in conjunction with the annual performance of SR 4.1.5(3) ensures this requirement is met.

3.2 REACTOR CONTROL AND SAFETY SYSTEMS

LCO 3.2.1 CONTROL RODS

LCO 3.2.1 requires that all three control rods (1 safety rod, 1 shim rod and 1 regulating rod) are operable whenever the reactor is in the reactor operating condition. A control rod is considered operable if it is capable of being fully inserted both manually and within the limits of LCO 3.2.2 following an RPS scram signal.

The requirement that all three control rods are capable of being fully inserted in combination with the requirement for a minimum SDM specified in LCO 3.1.3 ensures that the ARRR reactor can be made subcritical starting from any permissible operating condition by inserting the control rods assuming an unanticipated failure results in the most reactive control rod remaining fully withdrawn. Therefore, requiring that all three control rods are operable ensures that the reactor will be shut down and remain shut down following a reactor scram.

No specific SR is required to verify LCO 3.2.1 is met because free movement of each control rod for insertion is verified by the performance of SR 4.2.2 (1) and whenever the reactor is shutdown.

LCO 3.2.2 SCRAM TIME

LCO 3.2.2 requires that the total time for complete insertion of the control rods is ≤ 600 milliseconds following receipt of a scram signal by the safety system channels listed in Table 3.2-1. This LCO requires that control rods fully insert within the original equipment design specification time limits following any reactor scram signal to confirm that control rods are not binding and that the reactor protection system scram bus is functioning properly. USAR Chapter 13, Accident Analyses, does not make any assumptions regarding a minimum negative reactive insertion rate following a reactor scram for the mitigation of any event.

SR 4.2.2(1) requires that control rod scram times are measured and within the limits of LCO 3.2.2 semiannually and following any significant change to the core or any significant work on the control rods or the control rod drive system. This SR may be satisfied using any of the Functions listed in Table 3.2-1 to initiate the scram. Performance of this SR also verifies that SR 4.2.1 is met.

In order to minimize impact on normal reactor operations, the semiannual performance of this SR on one or more control rods may be initiated from the reactor critical condition at any power level by initiating a reactor scram using any function listed in Table 3.2-1. The performance of this SR following significant change to the core or any significant work on the control rods or the control rod drive system should be performed on one rod at a time prior to the initial criticality following the change that required performance of the SR.

LCO 3.2.3 REACTIVITY INSERTION RATE

LCO 3.2.3 requires that the rate of reactivity addition by the control rods shall be ≤ 0.080% Δk/k/second (≤ $0.11/second) whenever the reactor is in the reactor operating condition. This LCO limits the maximum rate of reactivity addition during a normal reactor startup to a rate that allows the operator to easily control reactor power. USAR Chapter 13, Accident Analyses, does not make any assumptions regarding a maximum reactive insertion rate from the control rods as an initial condition for any accident or transient (e.g., startup rod withdrawal accident). In fact, the accident analysis indicates that no fuel damage is expected to occur following a stepwise insertion of all of the reactor's excess reactivity resulting from a simultaneous ejection of all three control rods even if the rod ejection was initiated with the reactor critical in the source range.

SR 4.2.3(1) requires that the reactivity worth of each control rod is measured annually and following significant changes to the core or any control rod. The annual frequency for verification of control rod reactivity worth is consistent with the expected slow change in reactivity worth over time.

SR 4.2.3(2) requires that the withdrawal speed of each control rod is measured semiannually and the maximum rate of reactivity addition verified to be within the limits of LCO 3.2.3. When SR 4.2.3(1) is performed following significant changes to the core, the withdrawal speed of each control rod does not need to be measured but the maximum rate of reactivity addition should be verified to be within the limits of LCO 3.2.3 based on the most recent performance of SR 4.2.3(1).

LCO 3.2.4 REACTIVITY INSERTION MONITORING

LCO 3.2.4 requires that subcritical multiplication levels be plotted using input from a minimum of three instrumentation channels prior to criticality during experiments with the potential to affect core reactivity. Nuclear instrumentation channels listed in Table 3.2-1 may be used to satisfy requirements for one or more channels. This LCO requires careful monitoring of reactivity additions during the approach to criticality when experiments are in place that could have an unexpected effect on reactivity.

SR 4.2.4 requires that subcritical multiplication levels be plotted in accordance with the requirements of LCO 3.2.4 from start of control rod withdrawal until reactor criticality.

LCO 3.2.5 **REACTOR SEQUENCE AND INTERLOCKS**

LCO 3.2.5 requires the following whenever the reactor is not in the reactor secured condition:

a) Instrumentation that allow the operator to monitor changes in reactivity and reactor power level during a reactor startup is operating and on-scale;

b) Instrumentation in Table 3.2-1 that initiates a reactor scram if specified limits are exceeded is operating;

c) Control rod withdrawal and sequence interlocks that limit the rate of reactivity addition from control rod withdrawal are operating; and

d) The interlock is operating that prevents the shim rod and regulating rod from being withdrawn until the safety rod is fully withdrawn and thus capable of inserting sufficient reactivity for reactor shutdown following a reactor scram.

These interlocks minimize the potential for an inadvertent reactivity addition and ensure that the equipment used to detect and respond to an inadvertent addition is operating before a reactor startup is initiated. LCO 3.2.5(4) allows one of the interlocks required by LCO 3.2.5 to be inoperable when the reactor is in the reactor shutdown condition to allow for maintenance or testing of the interlocks. Allowing one of the interlocks to be inoperable when in the reactor shutdown condition is acceptable because the reactor is already subcritical by at least 0.73% Δk/k ($1.00) and is not at significant risk from an inadvertent addition of reactivity.

SR 4.2.5 requires annual verification of the operability of each of the interlocks required by LCO 3.2.5. Annual verification is acceptable because the interlocks are highly reliable. Additionally, the interlocks are backups to the primary methods for preventing inadvertent reactivity additions which are operating procedures and operator training.

LCO 3.2.6 **REACTOR PROTECTION SYSTEM (RPS) INSTRUMENTATION**

LCO 3.2.6 requires that each of the 9 reactor protection system channels listed in Table 3.2-1 be operable with the limiting safety system settings (LSSS), and annunciator and alarm set points within the limits specified in Table 3.2-1. To be considered operable, all of the scram functions and associated annunciators and alarms listed in Table 3.2-1 for each channel must be operable. USAR Section 7.2, Reactor Protection System, provides a detailed description of each of the RPS functions.

Table 3.2-1, footnote (a) provides clarification that channel 1 scrams are bypassed when channel 2 exceeds a fixed setting, approximately 1 x 10^{-10} amps. When bypassed, Channel 1 detector is shorted and high voltage is removed to protect the channel 1 detector from high neutron flux when the power level is above the source range. The channel is considered operable when it is bypassed as required by the design.

Table 3.2-1, footnote (b) provides clarification that the low level scrams are bypassed on channels 3 and 4 when channel 2 is below a fixed setting approximately 1 x 10^{-10} amps. This feature ensures that the low level scrams on channels 3 and 4 are bypassed until neutron flux levels are high enough to be measured on channel 2 (i.e. until neutron flux levels are above the Function 3.b and 4.b low flux scram set points).

Required actions and completion times for inoperable RPS channels are provided in Table 3.2-1, footnotes (c) and (d).

Table 3.2-1, footnote (c) establishes requirements when any one required channel is inoperable. In this condition, the inoperable channel must be restored to operable or the reactor must be placed in the secured condition within 8 hours of the discovery that the channel is inoperable. The allowable out of service time of 8 hours for any one RPS channel is acceptable because none of the RPS channels are assumed to function to prevent, detect, or mitigate the consequences of any of the accidents or transients discussed in USAR Chapter 13, Accident Analyses. Additionally, one or more alternate methods exist for the operator to monitor the parameter with the inoperable instrumentation and, if necessary, initiate a manual reactor scram.

a) Alternate indication for an inoperable neutron flux channel is provided by at least one other neutron flux channel as shown on USAR Figure 7.2-4, Normal Operating Range of the 4 Neutron Flux Detectors. With one inoperable channel, neutron flux can be monitored on two alternate channels when in the power range and one alternate channel when in the source or intermediate range. As shown on Technical Specification Table 3.2-1, redundant scram functions exist for each scram function lost as a result of the inoperable channel. Although a reactor startup would not be initiated with any neutron flux channel inoperable, the requirements of LCO 3.2.5 and installed interlocks will prevent a reactor startup with an inoperable source range nuclear flux channel.

   None of the neutron flux channel scrams are assumed to function to prevent, detect, or mitigate the consequences of any of the accidents or transients discussed in USAR Chapter 13, Accident Analyses. Additionally, the probability of an event that would require initiation of a reactor scram from a neutron flux channel during any 8 hour period is low. Therefore, operation for up to 8 hours with one inoperable neutron flux channel does not create a significant safety hazard.

b) Alternate indication for an inoperable reactor tank low water level channel is provided by the demineralizer water flow alarm required by Technical

Specification Table 3.2-2. Loss of primary coolant cleanup (demineralizer) loop flow, which will occur when reactor water tank level falls below the level of the suction skimmer at the water surface, provides a diverse alarm that would be an early indication of a low reactor tank water level. Investigation of the demineralizer water flow alarm would allow the operator to initiate a reactor scram in a timely manner.

As discussed in USAR Chapter 13, Accident Analyses, a reactor scram is not assumed to mitigate the consequences of any loss of cooling event. Additionally, design features described in USAR Chapter 5, Reactor Coolant, minimize the potential for a loss of water inventory from the reactor water tank. These design features make the probability of an event that would require initiation of a reactor scram from low reactor water tank level very low. Therefore, operation for up to 8 hours with an inoperable reactor tank low water level channel does not create a significant safety hazard.

c) Alternate indication for an inoperable seismic disturbance channel is the control room operator. As described in USAR Section 7.2.6, Seismic Disturbance, the purpose of this scram function is to ensure the reactor is shut down as soon as the initiation of a seismic event is detected. The Technical Specification LSSS for the seismic monitor is very conservatively set for a seismic event with intensity 'IV' which corresponds to an average peak acceleration of 0.015 g to 0.02 g. ARRR components are designed for lateral acceleration in excess of 1.0 g. Since a seismic event with intensity 'IV' is typically felt by people inside buildings, the control room operator could be expected to initiate a reactor scram at the first indication of a significant seismic event.

As stated in USAR Section 2.5.6, Maximum Earthquake Potential, the largest ground motion expected from a reasonably expectable maximum earthquake has a design response spectrum anchored at a high frequency asymptote of 0.50 g. ARRR components are designed for lateral acceleration in excess of 1.0 g. The ARRR design is sufficiently conservative to assure that the reactor can be safely shut down in the event of a major earthquake. Additionally, the probability of a seismic event that would require initiation of a reactor scram during any 8 hour period is low. Therefore, operation for up to 8 hours with an inoperable seismic disturbance channel does not create a significant safety hazard.

d) As described in USAR Section 7.2.7, High Reactor Tank Water Temperature, the TS limit of ≤ 130 °F for reactor water temperature is intended to minimize thermal degradation of the ion exchange resin in the demineralizer loop and is not required for reactor protection. Therefore, operation for up to 8 hours with an inoperable high reactor tank water temperature channel does not create a significant safety hazard.

e) The manual scram bar and master key switch provide alternate methods for initiating a manual reactor scram.

Table 3.2-1, footnote (d) establishes requirements when more than one required channel is inoperable. In this condition, all but one of the inoperable channels must be restored to operable or the reactor must be placed in the secured condition within 1 hour of the discovery that more than one channel is inoperable. The requirements of Table 3.2-1, footnote (c), for restoration within 8 hours from the time the channel is discovered inoperable continues to apply.

Although the reactor operator is expected to act conservatively in response to the inoperability of multiple RPS channels, allowing one hour to attempt restoration of multiple inoperable RPS channels is acceptable because none of the RPS channels are assumed to function to prevent, detect, or mitigate the consequences of any of the accidents or transients discussed in USAR Chapter 13, Accident Analyses.

SR 4.2.6(1) requires a channel check of each of the applicable channels listed in Table 3.2-1 prior to exiting the reactor secured condition on the first reactor startup each day. A channel check is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. A channel check for a single channel instrument is satisfied by verification that the sensor or the signal processing equipment has not drifted outside its Technical Specification limit. A channel check will detect gross channel failure and is intended to ensure the instrumentation continues to operate properly between each channel calibration.

SR 4.2.6(2) requires channel tests of each of the applicable channels listed in Table 3.2-1 be performed semiannually.

SR 4.2.6(3) requires channel calibrations of each of the applicable channels listed in Table 3.2-1 be performed prior to initial use.

LCO 3.2.7 REACTOR MONITORING SYSTEM (RMS) INSTRUMENTATION

LCO 3.2.7 requires that each of the 4 instruments listed in Table 3.2-2 be operable with the alarm and annunciator set points within the limits specified in Table 3.2-2. To be considered operable, all of the alarm functions and associated annunciators and alarms listed in Table 3.2-1 for each channel must be operable. USAR Section 7.3, Reactor Monitoring Systems, provides a detailed description of each of the RMS functions.

Required actions and completion times for inoperable RMS channels are provided in Table 3.2-2, footnotes (a), (b) and (c).

Table 3.2-2, footnote (a) establishes the requirement that an alternate method of monitoring the parameter shall be established within 24 hours from the discovery that a bridge crane location or reactor water radioactivity channel is inoperable. The alternate monitoring method may be substituted for the required channel for maximum of 30 days from the time the channel is discovered to be inoperable. Otherwise, the reactor shall be placed in the reactor secured condition within following 4 hours.

Visual verification that the bridge crane in its proper location within 24 hours and every 7 days thereafter is an adequate alternate for the bridge crane location alarm. As described in USAR Section 7.3.1, the bridge crane is normally stored approximately [redacted] and is maintained in the storage position when not in use. The bridge crane is designed for lateral accelerations in excess of [redacted] which, as described in USAR 2.5.6, Maximum Earthquake Potential, is larger than the acceleration expected from an earthquake at the ARRR. However, the bridge crane is kept in the storage location when not in use as a precaution against it falling into the reactor pool as a result of an earthquake. The alarm required by Table 3.2-2 ensures that the console operator is cognizant of the location of the bride crane.

Sampling and counting the reactor tank water within 24 hours and every day thereafter is an adequate alternate for the reactor water radioactivity channel. The reactor water radioactivity monitor is intended to provide an early indication of a fuel element cladding failure. Very significant fuel element cladding failures will be detected by the gaseous effluent monitor required by LCO 3.5.2.

Table 3.2-2, footnote (b) requires that primary coolant conductivity is verified using an alternate method within 7 days of the discovery that the conductivity channel is inoperable and every 7 days thereafter. An acceptable alternate method is taking a representative sample of the reactor coolant and measuring the conductivity. The alternate method may be substituted for the required channel for a maximum of 30 days from the time the channel is inoperable. Otherwise, the reactor shall be placed in the reactor secured condition within following 4 hours.

Allowing 7 days to establish an alternate method for verification that reactor water conductivity is within required limits is acceptable because the channel is not assumed to function to prevent or mitigate the consequences of any accident or transient. The primary function of the conductivity monitor is to monitor the effectiveness of the demineralizer. Reactor water pH and conductivity not within the specified limits does not represent an immediate safety concern, especially at the low temperatures at which the ARRR is operated. Allowing 30 days to restore the inoperable channel provides a reasonable period of time for repair of the inoperable channel.

Table 3.2-2, footnote (c) requires that reactor demineralizer water flow is verified using an alternate method immediately upon discovery that the demineralizer water flow channel is inoperable and every 24 hours thereafter. Demineralizer water flow is a prerequisite for the operability of the reactor water radioactivity channel. The alternate method may be substituted for the required channel for a maximum of 30 days from the time the channel is inoperable. Otherwise, the reactor shall be placed in the reactor secured condition within following 4 hours.

Allowing periodic verification of reactor demineralizer water flow as an alternate method for the demineralizer water flow channel for up to 30 days is acceptable because neither reactor demineralizer water flow channel or the associated reactor water radioactivity channel is assumed to function to prevent or mitigate the consequences of any accident or transient. Allowing 30 days restore the inoperable channel provides a reasonable period of time for repair of the inoperable channel.

SR 4.2.7(1) requires a channel check of each of the applicable channels listed in Table 3.2-2 prior to exiting the reactor secured condition on the first reactor startup each day. A channel check is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. A channel check for a single channel instrument is satisfied by verification that the sensor or the signal processing equipment has not drifted outside its Technical Specification limit. A channel check will detect gross channel failure and is intended to ensure the instrumentation continues to operate properly between each channel calibration.

SR 4.2.7(2) requires channel tests of each of the applicable channels listed in Table 3.2-2 be performed semiannually except for the reactor water radioactivity channel which shall be tested prior to exiting the reactor secured condition on the first reactor startup each day.

SR 4.2.7(3) requires channel calibrations of each of the applicable channels listed in Table 3.2-2 be performed prior to initial use except for the reactor water radioactivity channel which shall be calibrated biennially.

3.3 COOLANT SYSTEMS

LCO 3.3.1 REACTOR TANK WATER LEVEL

LCO 3.3.1 requires that reactor tank water level be maintained ≥ 16 feet above the top of the active core whenever the reactor is in the reactor operating condition or reactor shutdown condition. This limit ensures adequate shielding for personnel working above the reactor during reactor operation. Additionally, as explained in USAR Chapter 7, Instrumentation and Control Systems, an automatic reactor scram on "Low Pool Water Level" is initiated when the reactor tank level is below normal but still ≥ 16 feet above the top of the core.

Performance of the channel checks, channel tests, and channel calibrations of RPS functions that monitor reactor tank water level in SR 4.2.6.1, SR 4.2.6.2, and SR 4.2.6.3 are sufficient to verify that LCO 3.3.1 is being met.

LCO 3.3.2 PRIMARY COOLANT WATER TEMPERATURE

LCO 3.3.2 requires that reactor tank water temperature be maintained ≥ 60 °F and ≤ 130 °F whenever the reactor is in the reactor operating condition or reactor shutdown condition. Maintaining the reactor tank water temperature ≥ 60 °F satisfies an assumption that ensures the bath temperature coefficient of reactivity remains negative. Maintaining the reactor tank water temperature ≤ 130 °F prevents damage to ion exchanger resin in the demineralizer system. A temperature switch located in the reactor tank initiates a scram prior to temperature reaching 130 °F.

Performance of the channel checks, channel tests, and channel calibrations of RPS functions that monitor reactor tank water temperature in SR 4.2.6.1, SR 4.2.6.2, and SR 4.2.6.3 are sufficient to verify that LCO 3.3.2 is being met.

LCO 3.3.3 PRIMARY COOLANT WATER QUALITY

LCO 3.3.3 requires that primary coolant water be maintained with the pH ≤ 7.5 and the conductivity ≤ 5 μmho/cm at all times. These limits are designed to minimize corrosion and the presence of activated corrosion products in the reactor coolant. Elevated conductivity levels in the reactor water indicate the presence of corrosion products and promote more corrosion. Experience with water quality control at many reactor facilities, including the ARRR, has shown that maintaining water conductivity and pH within limits provides acceptable corrosion control. Maintaining low levels of dissolved electrolytes also reduces the amount of induced radioactivity which in turn decreases the exposure of personnel to radiation. The reactor water conductivity monitor is located near the top of the pool, under the bridge, and provides continuous indication at the reactor control cabinet.

If LCO 3.3.3 is not met, action shall be initiated within 24 hours to restore primary coolant water pH and conductivity to within the specified limits. Conductivity and pH not within the specified limits does not represent an immediate safety concern, especially at the low temperatures at which the ARRR is operated. However, the adverse effects of accelerated corrosion and the resulting accumulation of activation products are time dependent. Therefore, actions to identify the cause and prevent recurrence must be initiated within 24 hours and completed within a reasonable period of time. Corrective action typically involves identification and elimination of the source of contamination of the reactor coolant and/or replacement of ion exchange resin.

SR 4.3.3(1) requires that reactor coolant pH and conductivity be measured and verified to be within the limits of LCO 3.3.3 monthly.

SR 4.3.3(2) requires that the radioactivity level of the primary coolant be analyzed annually.

3.4 VENTILATION SYSTEMS

LCO 3.4.1 CONTROL ROOM

LCO 3.4.1 requires that the control room be maintained at a positive pressure with respect to the reactor room whenever the reactor is in the reactor operating condition or reactor shutdown condition and during movement of irradiated fuel. As described in USAR Section 6.1, Containment and Confinement, and USAR Section 9.1, Heating, Ventilation, And Air Conditioning Systems, exclusion of airborne radioactive material from inhabited areas, especially the control room, is accomplished by maintaining these areas at a positive pressure relative to the reactor bay. The air flows throughout the facility have been designed so that the control room, the lunch room, the office complex, the rest rooms, and the machine shop are at positive pressure with respect to the reactor high bay area. The positive pressure is maintained by an oversized ventilation system with a high fraction of make-up air from the outside. LCO 3.4.1 requires that the control room is at a positive pressure with respect to the reactor high bay area whenever the reactor is not in a secured condition. However, exclusion of airborne radioactive material from the control room following an accident is not assumed in the accident analysis because control room habitability is not assumed.

SR 4.4.1 requires quarterly verification that the control room is at a positive air pressure with respect to the reactor room.

LCO 3.4.2 REACTOR BUILDING CIRCULATION FANS

LCO 3.4.2 requires that all air conditioning systems and building circulating fans in or normally open to the high bay area have the capability to be shut off from a single control in the control room whenever the reactor is in the reactor operating condition or reactor shutdown condition and during movement of irradiated fuel.

As described in USAR Section 6.1, Containment and Confinement, and USAR Section 9.1, Heating, Ventilation, And Air Conditioning Systems, the reactor high bay ventilation system, setup room ventilation system, chemical laboratory hood blower, and rest room vent fans are shut off to reduce the potential for spread of contamination and airborne radiation during any radiological event. All of the ventilation equipment that should be shut off during a radiological event is controlled by an air handling relay on a single circuit. A clearly marked emergency button located near the control room outside door locks out the power to this relay and prevents any of the equipment from operating.

SR 4.4.2 requires quarterly verification of proper operation of the emergency shutoff from the control room of all air conditioning systems and building circulating fans that are in or normally open to the high bay area.

3.5 CRITICALITY ALARM, RADIATION AND RADIOACTIVE EFFLUENT MONITORING

LCO 3.5.1 CRITICALITY ALARM AND AREA RADIATION MONITORING

LCO 3.5.1(1) and Table 3.5-1 require the operability of a fixed gamma monitor employing Geiger tube detectors located on the wall connecting the control room and the reactor room. This monitor serves as both an area radiation monitor and a criticality alarm and must be operable at all times. If the alarm occurs during normal working hours, the monitoring company contacts the ARRR facility for verification. Upon verification of the alarm, the monitoring company contacts the San Ramon Valley Fire Protection District. If the alarm occurs during non-working hours, the monitoring company contacts the on-call representative of Aerotest Operations, Inc. and the San Ramon Valley Fire Protection District. This feature is described in USAR Section 7.4.1, Criticality Alarm and Area Radiation Monitor, and USAR Section 8.3, Battery Backup Electrical Power Systems.

Table 3.5-1 specifies that the criticality alarm and area radiation monitoring must actuate at ≤ 10 mrem/hour which would be indicative of a significant radiological event at the ARRR facility. The area radiation monitor actuates a siren within the reactor building on high radiation level that warns personnel to evacuate the reactor room in the event of high radiation levels. In addition to the local alarm, this system annunciates at a commercial monitoring company. Both the local alarm and annunciation at the commercial monitoring company must be operable.

Table 3.5-1, footnote (a) requires that an alternate method of monitoring the parameter shall be established within 8 hours if the criticality alarm and area radiation monitor becomes inoperable. An alternate method may consist of a portable radiation monitoring instrument with an alarm that is monitored by the reactor console operator. The alternate monitoring method may be used for up to 7 days from the time the required monitored is discovered to be inoperable. If the alarm and area radiation monitor is not restored to operable within 7 days, the reactor should be placed in the reactor secured condition within the following 4 hours.

a) When the reactor is not in the secured condition, use of an alternate monitoring method for performing this function for up to 7 days is acceptable because of the following: the control room is continuously manned by a reactor operator when the reactor is not in the secured condition; control room personnel will be alerted to the radiological condition by the alternate monitor; and, control room personnel can alert facility personnel in a manner equivalent to the alarm.

b) When the reactor is in the secured condition and no fuel movement is in progress, use of an alternate monitoring method for performing this function for up to 7 days is acceptable because an unplanned criticality event or significant radiological emergency is improbable when the reactor is in the secured condition and fuel movement is not in progress.

Table 3.5-1, footnote (b) requires that any movement of fuel elements be stopped immediately if the criticality alarm and area radiation monitor is inoperable during movement of a fuel element. This action shall not preclude completion of movement of a fuel element to a safe position. This action significantly reduces the potential for a radiological emergency while the monitor is not operable.

SR 4.5.1 (1) requires a quarterly channel test of the criticality alarm and area radiation monitoring system. This test shall include the reactor building siren and receipt of the alarm by the automatic alarm monitoring company.

SR 4.5.1(2) requires a biennial channel calibration of the criticality alarm and area radiation monitoring system.

LCO 3.5.1 (2) requires that radiation sensitive badges be located at several locations within the reactor building for radiation analysis at all times. These badges are used for routine monitoring of facility radiation levels.

SR 4.5.1(3) requires quarterly reading and analysis of the radiation sensitive badges placed at strategic locations within the reactor building.

LCO 3.5.2 GASEOUS EFFLUENT MONITORING

LCO 3.5.2 requires that a gas sample be withdrawn continuously from the roof vent above the reactor, or from the vicinity of the reactor pool, and pumped through a radioactive gas detector chamber within the limits specified in Table 3.5-1 for the building gaseous effluent monitor. This requirement applies whenever the reactor is in the reactor operating condition or reactor shutdown condition and during movement of irradiated fuel. The gaseous effluent monitor intake locations and the detector design are intended to provide early detection of a leaking fuel element. The building gaseous effluent monitoring system is described in USAR Section 7.4.2.

Table 3.5-1, footnote (a) requires that an alternate method of monitoring the parameter shall be established within 8 hours if the gaseous effluent monitor becomes inoperable. An alternate method may consist of a portable radiation monitoring instrument with an alarm that is monitored by the reactor console operator. The alternate monitoring method may be used for up to 7 days from the time the required monitored is discovered to be inoperable. If the alarm and area radiation monitor is not restored to operable within 7 days, the reactor should be placed in the reactor secured condition within the following 4 hours.

a) When the reactor is not in the secured condition, use of an alternate monitoring method for performing this function for up to 7 days is acceptable because of the following: the control room is continuously manned by a reactor operator when the reactor is not in the secured condition; control room personnel will be alerted to the radiological condition by the alternate monitor; and, control room personnel can alert facility personnel in a manner equivalent to the alarm.

b) When the reactor is in the secured condition and no fuel movement is in progress, use of an alternate monitoring method for performing this function for up to 7 days is acceptable because both the potential for and consequences of a leaking fuel element are significantly reduced when the reactor is in the secured condition and fuel movement is not in progress.

Table 3.5-1, footnote (b) requires that any movement of fuel elements be stopped immediately if the gaseous effluent monitor is inoperable during movement of a fuel element. This action shall not preclude completion of movement of a fuel element to a safe position. This action significantly reduces the potential for a radiological emergency while the monitor is not operable.

SR 4.5.2 (1) requires a channel test of the gaseous effluent monitoring system be performed prior to exiting the reactor secured condition on the first reactor startup each day.

SR 4.5.2(2) requires a biennial channel calibration of the gaseous effluent monitoring system.

LCO 3.5.3 PARTICULATE EFFLUENT MONITORING

LCO 3.5.3 requires that a particulate sample be withdrawn continuously from the reactor room and collected on filter paper whenever the reactor is in the reactor operating condition or reactor shutdown condition and during movement of irradiated fuel. This monitor is primarily used for environmental monitoring. This LCO requires a continuous air sample and does not require operability of a specific air sampler. Therefore, a portable air sampler may be substituted for the installed sampler at any time.

SR 4.5.3(1) requires verification that that the reactor room particulate sampler be verified to be operating daily each day the reactor is operated.

SR 4.5.3(2) requires that the reactor room particulate sample be counted monthly.

LCO 3.5.4 POST ACCIDENT RADIATION MONITORING

LCO 3.5.4 requires that radiation detector packets containing a series of threshold detectors be maintained at all times at several locations within the reactor building for post-accident radiation analysis.

SR 4.5.4 requires annual verification that radiation detector packets containing threshold detectors are in place.

LCO 3.5.5 PORTABLE RADIATION MONITORING INSTRUMENTS

LCO 3.5.5 requires that the following portable radiation monitoring instruments be operable and onsite at all times:

a) A portable survey instrument for measuring beta-gamma dose rates in the range of 0.01 mr/hr to 50 r/hr.

b) A portable instrument for measuring fast and thermal neutron dose rates from 0.1 mr/hr to 1.0 r/hr.

SR 4.5.5 requires quarterly calibrations of each required portable radiation monitoring instrument.

LCO 3.5.6 RADIOACTIVE EFFLUENT LIMITS

LCO 3.5.6 is a restatement that normal releases of radioactive effluents from the ARRR must not exceed 10 CFR 20 limits. Additionally, Ar-41 production must be determined annually.

SR 4.5.6 requires an annual administrative verification that normal releases of radioactive effluents from reactor operation did not exceed 10 CFR 20 limits.

3.6 EXPERIMENTS

Technical Specification restrictions on types of experiments and experimental facilities, restrictions on the maximum reactivity values of experiments, and, restrictions on experimental materials (including explosives), limit the effect that the failure of an experiment could have on the facility to less than those postulated for other events described in USAR Chapter 13, Accident Analyses.

LCO 3.6.1 EVALUATION AND APPROVAL OF EXPERIMENTS

LCO 3.6.1 requires that experiments shall be evaluated and approved in accordance with the requirements of Technical Specification 6.5.1 prior to insertion of the experiment into the reactor. Technical Specification 6.5.1 requires that all new experiments or class of experiments or changes to experiments must be reviewed by the RSC and approved in writing by the reactor supervisor prior to initiation. This review must include a safety review that evaluates the experiment in the most reactive condition and an analysis of the possible hazards associated with the performance of the experiment.

SR 4.6.1 requires that the reactor operator verify that experiments have been evaluated and approved in accordance with the requirements of Technical Specification 6.5.1 prior to insertion of the experiment into the reactor.

LCO 3.6.2 REACTIVITY LIMITS DURING EXPERIMENTS

LCO 3.6.2 requires that evaluations of experiments in accordance with the Technical Specification 6.5.1 demonstrate each of the following:

a) The total combined reactivity that could be added to the core simultaneously by an experiment is ≤ 1.46% Δk/k (≤ $2.00).

b) The total combined reactivity that could be added to the core simultaneously by an experiment not rigidly fixed in place is ≤ 0.73% Δk/k (≤ $1.00).

c) The reactivity insertion rate for experiments having moving parts is < 0.073% Δk/k /second (< $0.10/second) except that moving parts worth < 0.0365% Δk/k /second (< $0.05/second) may be oscillated or removed at higher frequencies.

d) The limits in LCO 3.1.2, Excess Reactivity, and LCO 3.1.3, Shutdown Margin, shall not be exceeded including the reactivity worth of the experiment when evaluated in the most reactive condition.

These Technical Specifications limits on excess reactivity and shutdown margin include the reactivity worth of the experiment when evaluated in the most reactive condition. These limits protect the reactor from power excursions and ensure that the reactor can always be safely shut down even during the conduct of experiments. The LCO 3.6.2 limits on reactivity introduced by experiments are acceptable because collectively they ensure that the maximum reactivity insertion from an experiment is significantly less than the stepwise insertion of all of the core's excess reactivity (i.e. 2.19% Δk/k ($3.00)). Insertion of all of the core's excess reactivity is assumed in the analysis of the ejection of all three control rods as described in USAR Section 13.2.1, Insertion of Excess Reactivity. Additionally, the LCO 3.6.2 limits on reactivity ensure that sufficient SDM exists to allow the reactor to be made subcritical by an automatic or manual reactor scram and that the reactor will remain subcritical without further operator action even if the most reactive rod remains in its most reactive position.

There are no specific surveillance requirements for LCO 3.6.2 because SR 4.6.1(1) will ensure that the requirements of LCO 3.6.2 are met.

LCO 3.6.3 **SPECIAL NUCLEAR MATERIAL (SNM) INCLUDED IN EXPERIMENTS**

LCO 3.6.3 limits the amount of SNM introduced into the reactor for experiments to either: 5 grams of SNM in the form of solid samples; or, 3 grams of SNM in the form of liquid. LCO 3.6.3 further requires that liquid special nuclear materials shall be doubly encapsulated. LCO 3.6.3 allows only very small amounts of SNM to be introduced into the reactor for experiments to limit the potential reactivity effects. Additionally, the small quantities of SNM ensure that any release of gaseous and volatile fission products from this material is bounded by the release associated with the maximum credible accident addressed in USAR Chapter 13, Accident Analyses. Therefore, an experiment malfunction involving SNM will not result in consequences more severe than those listed in USAR Chapter 13 for the maximum credible accident.

SR 4.6.3 requires that the reactor operator verify that experiments that introduce SNM into the reactor meet the requirements of LCO 3.6.3 prior to insertion of the experiment into the reactor.

LCO 3.6.4 **MATERIALS USED DURING EXPERIMENTS**

LCO 3.6.4 prohibits experiments using materials that could result in any of the following: contamination of the reactor pool causing corrosive action on the reactor components or experiments; cause excessive production of airborne radioactivity; or, produce an uncontained violent chemical reaction. These issues, if applicable, are evaluated in accordance with the requirements specified in LCO 3.6.1. The restriction on materials that could produce an uncontained violent chemical reaction do not prohibit the radiography of explosives specifically permitted by the Technical Specifications.

LCO 3.6.5 EXPLOSIVE MATERIALS IN EXPERIMENTS

LCO 3.6.5 provides the following restrictions on explosive materials:

a) A maximum of five pounds equivalent TNT may be placed in the radiography facilities at one time.

b) A maximum of 1 pound equivalent TNT may be placed in the radiation field at one time. As described in USAR Section 13.3.3, the consequences of an explosive device of the type radiographed at the ARRR detonating during radiography has been evaluated. The maximum energy release is 490,000 calories for 1 pound of explosive. The energy released by detonation is equally divided between the shock wave and the internal blast pressure. The inertial and tensile strength of the explosive housing resists the impulsive force of the explosion up to the point of yield and fragmentation of the working end of the explosive assembly. The kinetic velocity of the fragments impinging on the inner wall of the neutron radiography facility is rapidly decelerated by the walls of the facility. Massive shielding surrounding the neutron radiography facility greatly attenuates the force of the explosion. As described in USAR Section 13.3.4, no damage to the control rods or the reactor would result from an explosion of this magnitude.

c) A maximum of 0.5 pound equivalent TNT per foot may be contained in long devices. As described in USAR 13.3.1, this limit is based on tests that demonstrated that detonation of one of these devices is not likely to result in detonation of adjacent devices either during radiography, when in storage or during shipment.

d) The maximum total exposure for explosive devices is 3×10^{11} neutrons/cm^2 and 3×10^3 roentgens of gammas. As described in USAR Section 10.2.1, Limits on Explosive Materials Experiments, this limit on radiation exposure for explosives is based on a study that determined that thermal neutron doses of 10^{15} neutrons/cm^2 and gamma doses of 5×10^6 roentgens on explosives have been experimentally evaluated and produce no detrimental effects on the explosives.

e) Explosive devices that have or provide a thrust in a definite direction upon ignition must be positioned so as to be aimed away from the reactor and components to minimize potential damage from an inadvertent detonation.

SR 4.6.5 requires that administrative controls verify, as applicable, that the requirements of LCO 3.6.5 are being implemented.

LCO 3.6.6 CONFIGURATION AND POTENTIAL FAILURE MECHANISMS

LCO 3.6.6 prohibits each of the following items:

a) Installation of experiments that could shadow the nuclear instrumentation detectors.

b) Installation of experiments in such a manner that a failure could interfere with the insertion of a reactor control rod.

c) Performance of experiments involving equipment whose failure could result in fuel element damage.

These issues are considered in the review and approval of experiments required by LCO 3.6.1.

LCO 3.6.7 GLORY HOLE FACILITY

LCO 3.6.7 allows one dry glory hole facility to be installed in any reactor core position. As described in USAR Section 10.1.3, the glory hole design allows it to accept capsules with a maximum diameter of [redacted]. Insertion of the glory hole may result in a gas filled volume within the core. If the volume is filled with air, activation of argon may result in unacceptable levels of Argon-41 in the area above the reactor. Therefore, the glory hole must be purged with CO_2 to prevent the formation of excessive amounts of Argon-41. If operated with a shield plug, the glory hole must be purged prior to each insertion of the shield plug. If operated without a shield plug, the glory hole must be purged with CO_2 continuously. Use of a glory hole must be in accordance with experiments that were reviewed and approved in accordance with the requirements of LCO 3.6.1.

SR 4.6.7 requires continuous gas samples be taken near the glory hole opening and Argon-41 presence monitored when the glory hole facility is operated without a shield plug. This SR ensures that the CO_2 purge is both being performed and being effective. Corrective action must be taken to prevent the release of gaseous activity in excess of 10 CFR 20 limits.

LCO 3.6.8 VERTICAL TUBES

LCO 3.6.8 allows vertical irradiation tubes with diameters up to 6 inches may be attached to the thermal column. As described in USAR Section 10.1.4, a vertical tube is a 6 inch diameter dry beam tube that is used for low flux dry hole experiments such as research on a new neutron detector. A vertical tube may be attached to any of the four slotted beams, two on each side of the thermal column assembly. Extensions of this tube allow experiments to be placed immediately adjacent to the reactor core. When the vertical tube is not being used for an experiment, it is placed near the reactor water tank wall to prevent production of Ar-41 and radiation streaming. Use of a vertical tube must be in accordance with experiments that were reviewed and approved in accordance with the requirements of LCO 3.6.1.

When in use, the vertical tube must be continuously purged with CO_2 to prevent the formation of excessive amounts of Ar-41. Additionally, the installed gas sampling system must be selected to sample in the area above the reactor so that corrective action can be taken to prevent the release of gaseous activity in excess of 10 CFR 20 limits.

SR 4.6.8(1) requires daily verification that each vertical tube is being purged continuously with CO_2 when a vertical tube is inserted.

SR 4.6.8(2) requires that continuous gas samples be taken near the pool and Argon 41 presence monitored when a vertical tube is inserted.

LCO 3.6.9 OTHER IRRADIATION FACILITIES

LCO 3.6.9 allows use of each of the following experimental facilities: the central core irradiation facility as described in USAR Section 10.1.5; the triangular in-core irradiation facilities, as described in USAR Section 10.1.6; and, In-core irradiation capsules as described in USAR 10.1.7. Use of these experimental facilities must be in accordance with experiments that were reviewed and approved in accordance with the requirements of LCO 3.6.1.

LCO 3.6.10 LARGE COMPONENT IRRADIATION BOX

LCO 3.6.10 prohibits installation of the large component irradiation box in the reactor tank. The large component irradiation box was not typically used at the ARRR facility and its use is not addressed in the USAR. The USAR must be amended and a Technical Specification change must be approved by the NRC prior to installation of the large component irradiation box.

LCO 3.6.11 PNEUMATIC TRANSFER FACILITY

LCO 3.6.11 prohibits installation of a pneumatic transfer facility in any reactor core position. The pneumatic transfer facility was not typically used at the ARRR facility and its use is not addressed in the USAR. The USAR must be amended and a Technical Specification change must be approved by the NRC prior to installation of a pneumatic transfer facility.

3.7 FUEL STORAGE AND TRANSFER

LCO 3.7.1 FUEL HANDLING TOOLS

LCO 3.7.1 requires that the fuel handling tools be secured with a locking mechanism under the cognizance of the Reactor Supervisor except when the fuel handling tools are authorized for use. This requirement prevents unauthorized movement of fuel and minimizes the potential for theft or diversion of a fuel element.

SR 4.7.1 requires weekly verification that the fuel handling tools are secured with a locking mechanism under the cognizance of the Reactor Supervisor when not authorized for use. This verification may be performed by visual verification that fuel handling tools are in proper storage location and re-locking. This SR may also be satisfied by using a tamper proof seal and performing a weekly verification that the lock and tamper proof seal are intact.

LCO 3.7.2 FUEL STORAGE IN THE REACTOR TANK

LCO 3.7.2 allows fuel elements to be stored in the storage racks located in the reactor tank if k_{eff} for the storage rack remains ≤ 0.9 for all conditions of moderation and reflection using light water. This limit is acceptable because it provides a very conservative margin for the prevention of an unplanned criticality in the storage racks.

SR 4.7.2 requires a verification be performed prior to any addition or movement of fuel in the reactor tank that the planned addition or movement of fuel in the reactor tank will not result in a geometric array where keff for the storage rack is > 0.9 for any conditions of moderation and reflection using light water. This verification may consist of an administrative review of the design specifications of the storage racks against the design specifications of the fuel that will be stored in the storage racks.

LCO 3.7.3 FUEL STORAGE IN THE FUEL STORAGE PITS

LCO 3.7.3 (1) allows up to 19 fuel elements containing up to 700 grams of U-235 to be stored in each fuel storage pit. These limits eliminate the potential for an unplanned criticality in the fuel storage pits and ensure that decay heat levels will be minimal.

LCO 3.7.3 (2) requires that each fuel storage pit containing a fuel element be secured with a lock and chain except during fuel transfer operations involving the fuel element being stored in the associated pit. This requirement prevents unauthorized opening of the fuel storage pit, prevents unauthorized movement of fuel, and minimizes the potential for theft or diversion of a fuel element. Locking the fuel storage pit when spent fuel is being stored will ensure the requirements of 10 CFR 1601, Control Of Access To High Radiation Areas, and 10 CFR 20.1801, Security Of Stored Material, are met when applicable.

SR 4.7.3(1) requires verification prior to any addition or movement of fuel in the fuel storage pits that that the requirements of LCO 3.7.3 (1) will still be met after the movement of the fuel element.

SR 4.7.3(2) requires quarterly verification that each fuel storage pit containing a fuel element is secured with a lock and chain except during fuel transfer operations involving the fuel element being stored in the associated pit.

LCO 3.7.4 FUEL NOT IN THE REACTOR AND NOT IN STORAGE

LCO 3.7.4 allows only one fuel element in the facility which is not either (1) in the core lattice; or, (2) in the storage racks in the reactor tank; or, (3) in a fuel storage pit. This requirement ensures ARRR staff resources for safety and security can be focused on the fuel element being moved and minimizes the potential for theft or diversion of a fuel element. This LCO is intended to allow fuel elements to be moved within the ARRR facility or the transportation of fuel elements onto or off the site and is not intended to allow storage of a fuel element in other than an approved storage location.

SR 4.7.4 requires verification that any movement or receipt of fuel at the facility will not result in more than one fuel element in the facility which is not in the reactor core lattice or in storage.

LCO 3.7.5 FUEL IN SHIPPING CONTAINERS

LCO 3.7.5 allows a fuel element to be contained in an approved fuel shipping container within the limitations of LCO 3.7.4 (i.e., only one fuel element not in the core lattice or in an approved storage location). This LCO is intended to allow fuel elements to be moved within the ARRR facility or the transportation of fuel elements onto or off the site and is not intended to allow storage of a fuel element in other than an approved storage location.

SR 4.7.5 requires verification that a fuel shipping container is approved shall be made prior to use as a shipping container.

LCO 3.7.6 FUEL TRANSFER IN THE REACTOR TANK

LCO 3.7.6 requires that all fuel transfers in the reactor tank be conducted by a minimum staff of three members, and shall include a licensed senior operator and a licensed operator. Fuel transfers in the reactor tank have the potential to result in changes in core reactivity and could create a radiation hazard. Therefore, requiring the presence of a senior reactor operator ensures the presence of an individual with the appropriate level of training to supervise this evolution. A licensed operator is required to manipulate the controls of any evolution that changes core reactivity and to provide an additional individual trained in reactor physics, radiological controls, and plant procedures to assist the senior reactor operator. LCO 3.7.6 also requires that the staff members monitor fuel transfers in the reactor tank using appropriate radiation monitoring instrumentation because of the potential that a fuel transfer could create a radiation hazard. This requirement is consistent with regulations and industry practice.

LCO 3.7.7 FUEL TRANSFER OUTSIDE THE REACTOR TANK

LCO 3.7.7 requires that fuel transfers outside the reactor tank but within the facility shall be supervised by a licensed Operator. Fuel transfers outside the reactor tank never involve more than one fuel element because of restrictions established by LCO 3.7.4 and does not have the potential to cause a criticality event because of the restrictions in LCO 3.7.3. Requiring the presence of a reactor operator ensures the presence of an individual trained in reactor physics, radiological controls, and plant procedures with the appropriate level of training to perform this evolution.

3.8 REACTOR FACILITY

LCO 3.8.1 REACTOR BUILDING ALARM

LCO 3.8.1 requires that the reactor building alarm be operable and monitored at all times to detect unauthorized entry into the ARRR building. This site security alarm has a 24-hour battery capacity and radio backup. The battery recharges continuously when electrical power is available. The alarm is monitored by an alarm monitoring company which will respond in accordance with the requirements of the site security plan if the alarm is actuated.

As explained in USAR Section 12.10, Security Planning, the building alarm is needed because the ARRR is licensed to possess Special Nuclear Material of Low Strategic Significance as defined in 10 CFR 73.2. Therefore, the ARRR is required to implement a security plan that includes a requirement to monitor the controlled access areas with an intrusion alarm to detect unauthorized penetrations or activities and assure that a watchman or offsite response force will respond to all unauthorized penetrations or activities.

SR 4.8.1 requires that the reactor building alarm be tested monthly. This test includes verification that the alarm monitoring company receives the appropriate signal when the alarm is actuated. This test also verifies that the backup battery and radio are functioning.

LCO 3.8.2 EXPLOSIVE MATERIAL AT THE REACTOR FACILITY

LCO 3.8.2 specifies that explosive materials may be brought into the facility for the purpose of being radiographed. As described in USAR Section 13.3, Storage and Radiography of Explosive Devices at the ARRR Facility, protecting operating personnel and the public from the effects of inadvertent detonation of explosive material is accomplished using a combination of methods. Procedural controls and the training and experience of personnel handling explosives minimize the potential for an unplanned detonation. Additionally, limits are placed on the type and amount of explosive material to minimize the consequences of an inadvertent detonation.

a) The requirement that explosive material must be stored in designated areas within the reactor facility is consistent with standard industrial safety practices and regulations for facilities handling "Low Explosives" in 27 CFR 555.202(b).

b) The requirement that Individual explosive devices must be less than 1000 grains equivalent TNT ensures that the consequences of an unplanned detonation are minimal. As described in USAR Section 13.3.1, Damage from Detonation of Stored Explosives, an explosion of 1000 grains of TNT in the unclad condition in a storage container results in a peak pressure of about 25 psi at the outside surface of the storage container and about 3 psi at 10 feet from the storage container. An explosion of this size will cause minimal damage to the ARRR facility.

c) The requirement that only solid or encased explosive materials may be brought into the facility minimizes the potential for an inadvertent detonation by preventing the long term buildup of loose explosive powders in the facility.

d) The requirement that any explosive devices containing loose explosive powders be completely encased also minimizes the potential for an inadvertent detonation by preventing the long term buildup of loose explosive powders in the facility.

e) The requirement that the maximum quantity of explosive material that may be possessed at one time be limited to 50 pounds equivalent TNT is based on Federal Explosives Law and Regulations, ATF P 5400.7 (09/00), Table 55.218. This regulation requires a 60 foot minimum distance between the barricaded explosive magazine and the nearest public street which is the employee parking lot. Additionally, as described in USAR Section 13.3.1, Damage from Detonation of Stored Explosives, an explosion of 50 pounds equivalent of clad TNT results in a peak pressure of about 40 psi at 10 feet. Pressure release from clad explosives would be considerably less. An explosion of this size will cause minimal damage to the ARRR facility.

f) The requirement that unshielded high frequency generating equipment not be operated within 50 feet of any explosive device minimizes the potential for an inadvertent detonation by eliminating potential sources of detonation.

g) The requirement that personnel handling the explosive devices are trained and familiar with the devices being radiographed is consistent with regulations and standard industrial practice for facilities handling explosives.

h) The requirement that personnel handling the explosive devices shall use special equipment such as non-sparking tools and shoes, protective clothing, safety shields, and grounded benches as required for the explosives being handled is consistent with regulations and standard industrial practice for facilities handling explosives.

i) The requirement that only explosive devices that do not exceed a maximum of ten pounds equivalent TNT may be removed from the storage areas at one time works in conjunction with LCO 3.6.5(1) which limits the amount of explosive material contained in devices that may be placed in the radiography facilities at a time to five pounds equivalent TNT. The combination of requirements ensures that only explosives in the radiography facility and those in the process of being prepared for radiography or returned to storage after radiography are removed from the designated storage areas.

j) The requirement that only explosive devices that will be radiographed within 8 hours, including preparation time, be removed from storage ensures that only explosives in the process of being prepared for radiography are removed from the designated storage areas. The 8 hour limit is necessary because some explosive devices may require as much as 8 hours for preparation (setup) and completion of radiography.

k) The allowance that restrictions for removal of explosive materials from designated storage areas do not apply to explosives packaged for shipment is acceptable because explosives packaged to meet Department of Transportation (DOT) requirements for shipment on commercial aircraft or over public highways do not represent a significant safety concern for the facility. Total explosive material onsite is limited to 50 pounds equivalent TNT in accordance with this LCO.

l) The requirement that a daily accountability log be maintained to show the amount of explosive material in the reactor facility is intended to minimize the potential for theft or diversion of explosive material and to provide documentation that the other requirements of this LCO are being met. The requirement that the accountability log contain a description of the explosive material and the location within the facility assumes that this is the minimum amount of information needed to satisfy the intent of the log.

SR 4.8.2 requires that the administrative controls that implement the requirements of LCO 3.8.2 be reviewed annually to ensure that ARRR procedural and technical requirements for handling explosive materials remain consistent with current regulations and industrial safety practices.

5

AEROTEST OPERATIONS, INC.

Aerotest Radiography and Research Reactor (ARRR)

DESCRIPTION OF PROPOSED CHANGES
TO THE
ARRR TECHNICAL SPECIFICATIONS

Revision 0

DOCKET No. 50-228
License No. R-98

Aerotest Operations, Inc.
3455 Fostoria Way
San Ramon, CA 94583

Aerotest Operations, Inc.

Description and Justification for
Proposed Changes to the ARRR Technical Specifications

Introduction:

Aerotest Operations, Inc. is proposing extensive changes to the Aerotest Radiography and Research Reactor (ARRR) Technical Specifications (TS) as part of the request for the renewal of the ARRR operating license. The proposed changes to the TS add numerous new technical requirements consistent with the recommendations in American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, "The Development of Technical Specifications for Research Reactors," ANS, La Grange Park, Illinois, 1990 (ANS 15.1).

In addition to the changes needed to adopt the recommendations in ANS 15.1, the proposed TS present each of the Limiting Conditions for Operation (LCOs) in three parts: (1) a statement of the technical requirements of the LCO; (2) an applicability statement that specifies under what operating conditions the LCO must be met; and, (3) Surveillance Requirements (SRs) that specify the periodic verifications or testing necessary to ensure the LCO is being met.

The proposed TS also include a specific requirement that action must be initiated to place the reactor and facility in a condition where the LCO is no longer applicable or the reactor is in the secured condition whenever an LCO is not met. If the LCO is applicable at all times, the proposed TS require that the Reactor Supervisor or designated alternate identify any additional compensatory actions required to place the reactor and facility in a safe condition. In some instances, the LCO includes actions that, if completed within the specified completion time, allow the LCO to be considered met.

Aerotest Operations, Inc. is also proposing a separate Bases document that, in conjunction with the proposed Updated Safety Analysis Report (USAR), provides background, an explanation, and the technical justification for each of the requirements in the proposed TS.

This document provides a description of the new requirements in the proposed TS and identifies the differences between the proposed TS and current Technical Specifications (CTS).

Attachment 1 to this document is a copy of the ARRR CTS that has been annotated to provide a detailed cross reference between the ARRR proposed TS and the CTS. The annotated copy of the CTS also shows significant changes being proposed to existing requirements.

Attachment 2 to this document is a copy of ARRR proposed TS that has been annotated to provide a detailed cross reference between the proposed TS and the CTS.

1.0 DEFINITIONS

Aerotest Operations, Inc. proposes adding definitions for the following terms to the ARRR Technical Specifications.

Channel	A definition for channel that is consistent with the ARRR design and recommendations in ANS 15.1 is being added to the TS. This definition is needed to ensure that requirements for the Reactor Protection System, Reactor Monitoring System, and Area and Effluent Radiation Monitoring Instrumentation in proposed TS Tables 3.2-1, 3.2-2, and 3.5-1 are properly applied.
Channel Calibration	A definition for channel calibration that is consistent with the ARRR design and recommendations in ANS 15.1 is being added to the TS. This definition is needed to ensure that requirements for the proposed TS Tables 3.2-1, 3.2-2, and 3.5-1 are properly applied.
Channel Check	A definition for channel check that is consistent with the ARRR design and recommendations in ANS 15.1 is being added to the TS. The proposed Technical Specification Bases provide the additional clarification that a channel check for a single channel instrument is satisfied by verification that the sensor or the signal processing equipment has not drifted outside its Technical Specification limit. A channel check will detect gross channel failure and is intended to ensure the instrumentation continues to operate properly between each channel calibration. Additionally, the Bases clarify that performance of a channel check is typically performed in conjunction with the TS requirement for the periodic verification that the parameter being monitored is within required limits.
Channel Test	A definition for channel check that is consistent with the ARRR design and recommendations in ANS 15.1 is being added to the TS. This definition is needed to ensure that requirements for testing instruments in TS Table 3.2-1, 3.2-2 and 3.5-1 are properly applied.
Control Rod	A definition for control rod is that is consistent with the ARRR design and recommendations in ANS 15.1 is being added to the TS.
Excess Reactivity	A definition for excess reactivity that, when used in conjunction with the definition proposed for Reference Core Condition, is consistent with the assumptions used in USAR Section 13.2, Accident Events and Scenarios, for the analysis of accidents involving the insertion of excess reactivity is being added to the TS.
Exclusion Area	A definition for exclusion area is being added to the TS. The proposed definition restates requirements for the exclusion area consistent with the definition in 10 CFR 100. This definition is needed for the application of TS 5.1.1 which specifies that the exclusion area must be established by a steel, locked perimeter fence that surrounds the ARRR facility. TS 5.1.1 also specifies that the restricted area, as defined in 10 CFR 20, consists of the entire exclusion area.

Explosive Material	A definition for explosive material using a generic industrial definition is being added to the TS. This definition, in conjunction with specific descriptions in the ARRR USAR, is needed to ensure that restrictions established in the Technical Specifications are applied appropriately.
Glory Hole	A definition for the glory hole is being added to the TS. This definition, in conjunction with descriptions in the ARRR USAR, is needed to ensure that restrictions established in the Technical Specifications are applied appropriately.
Moveable Experiment	A definition for moveable experiments is being added to the TS. This definition, in conjunction with descriptions in the ARRR USAR, is needed to ensure that restrictions established in the Technical Specifications are applied appropriately.
Rated Thermal Power (RTP)	A definition for Rated Thermal Power is being added to the TS. This definition is needed to ensure that the existing requirements in ARRR License Condition 2.C.(1), Maximum Power Level, that steady state power levels shall not be in excess of 250 kW (thermal).
Reactivity (Dollars ($))	A definition for reactivity that clarifies the relationship between reactivity measured in dollars and reactivity measured in Δk/k is being added to the TS. This definition is needed because assumptions used in TRIGA fuel performance and accident analyses and requirements in the ARRR CTS sometimes define reactivity in units of dollars and sometimes define reactivity in units of Δk/k. This definition establishes the relationship between reactivity in units of dollars and reactivity in units of Δk/k by specifying the value of the effective fraction of delayed neutrons (β) at 0.0073. This definition ensures a consistent understanding and application of assumptions regarding reactivity used in analyses described in USAR Chapter 13, Accident Analyses, and requirements established in both the existing and the proposed TS. The value of 0.0073 for β, the effective fraction of delayed neutrons, is a value typically used for TRIGA reactors and maintains or is more conservative than existing requirements and limits for reactivity. The proposed TS and USAR establish all requirements expressing reactivity in both dollars and Δk/k.
Reactivity Worth of An Experiment	A definition for the reactivity worth of an experiment is being added to the TS. This definition, in conjunction with descriptions in the ARRR USAR, is needed to ensure that restrictions established in the TS are applied consistently.
Reactor Operator	A definition for reactor operator is being added to the TS. This definition specifies that the term reactor operator applies only to an individual who is licensed to manipulate the controls of a reactor.

Reactor Secured

CTS Definition 1.1, Shutdown, is-being replaced with the term Reactor Secured to define the operating condition with the minimum potential for an accident or transient. The proposed definition of Reactor Secured is consistent with guidelines proposed in ANSI/ANS 15.1. The proposed definition differs from CTS Definition 1.1, Shutdown, by including two new conditions:

1. There is insufficient moderator available in the reactor to attain criticality; or;
2. There is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection.

The remaining portions of the proposed definition for Reactor Secured maintain the existing requirements in CTS Definition 1.1, Shutdown, except that the proposed definition is more conservative by requiring that the reactor be subcritical by greater than 0.73% $\Delta k/k$ (i.e., \$1.00) versus 0.7% $\Delta k/k$ required in CTS Definition 1.1. This change ensures that the new definition of Reactivity (Dollars), which defines the effective fraction of delayed neutrons (β) at 0.0073, does not inadvertently cause the proposed requirements for reactor secured to be less conservative than the requirements in CTS 1.1 for reactor shutdown.

Reactor Shutdown

In conjunction with the change described above which replaces the CTS term "Shutdown" with the new term "Reactor Secured," the term reactor shutdown is being redefined to describe a condition when the reactor is not in the reactor secured condition (i.e., called shutdown in the CTS) but remains subcritical by at least 0.73% $\Delta k/k$ (\$1.00) when in the reference core condition with the reactivity worth of all installed experiments included and the reactor is not in the reactor secured condition. This revised definition of Reactor Shutdown is consistent with recommendation in ANSI/ANS 15.1.

Reference Core Condition

A definition for Reference Core Condition that is consistent with recommendations in ANSI/ANS 15.1 is being added to the TS. This definition establishes standard or baseline conditions for ensuring requirements for core reactivity (e.g., shutdown margin) are properly understood and consistently applied.

Shutdown Margin (SDM)

A definition for Shutdown Margin that maintains existing requirements in CTS 5.3.1 is being added to the TS. The proposed definition is consistent with the guidelines proposed in ANSI/ANS 15.1.

2.0 SAFETY LIMIT (SL)

2.1 MAXIMUM FUEL ELEMENT TEMPERATURE

Aerotest Operations, Inc. proposes adding Safety Limit 2.1 that specifies that the temperature in any fuel element in the ARRR shall not exceed 500 °C under any condition of operation. Safety limits for nuclear reactors are defined in 10 CFR 50.36 as limits for important process variables that are necessary to reasonably protect the integrity of physical barriers that guard against the uncontrolled release of radioactivity. The purpose of this safety limit and basis for selecting 500 °C as the limit are explained in the Technical Specification Bases. No other safety limits are applicable to the ARRR design.

3.0 LIMITING CONDITIONS FOR OPERATION

3.0 GENERAL REQUIREMENTS

LCO 3.0.1, Abnormal Operation, maintains the existing requirement in current Technical Specification (CTS) 10.3.

LCO 3.0.2, Defects in Fuel Elements, Control Rods, or Control Circuitry, maintains the existing requirement in CTS 10.3.

LCO 3.0.3, Actions for LCO Not Met, specifies that LCOs must be met whenever the reactor is in the condition specified in the associated applicability statement. However, if an LCO is not met, the ARRR may be operating outside of design requirements or analysis assumptions specified in the Updated Safety Analysis Report (USAR). LCO 3.0.3 identifies the compensatory actions required upon discovery of a failure to meet an LCO. The Technical Specification Bases provide an explanation of the reasons for and details for the implementation of each of the actions specified in LCO 3.0.3.

3.1 REACTOR CORE PARAMETERS

LCO 3.1.1, Rated Thermal Power (RTP), maintains the existing requirement in License Condition 2.C (1) that the total reactor core heat transfer rate to the reactor coolant must not exceed 250 kW. LCO 3.1.1 adds the clarification that this power level is "as measured by the calibrated power channels."

SR 4.1.1 adds a new requirement that a reactor thermal power calibration must be performed annually and that neutron flux channel 3 and channel 4 must reflect the results of this thermal power calibration.

Addition of the clarification that the limit on thermal power is "as measured by the calibrated power channels" recognizes that the only method for the operator to verify that limits on thermal power are not being exceeded is by monitoring neutron flux using channel 3 and channel 4. This change is acceptable because there is no change from the understanding and implementation of the current requirement in License Condition 2.C (1). The addition of SR 4.1.1, a new requirement for comparing the results of a calorimetric heat balance calculation to the indication from neutron flux channels 3 and 4 and adjusting channel output accordingly, provides assurance that License Condition 2.C (1) will be met. As discussed in the Technical Specification Bases, operating experience indicates that an annual thermal power calibration of the neutron flux channels is sufficient to maintain required accuracy. Additionally, a new requirement for the control room operator to periodically monitor and compare the output from the redundant neutron flux channels (i.e., channel checks required by SR 4.2.6(1)) ensures that LCO 3.1.1 is being met at all times. The channel check requirement to compare the output of the redundant channels will ensure that any significant deviation between channels is promptly identified and investigated.

LCO 3.1.2, Excess Reactivity, maintains the existing requirement in CTS 5.1.2. LCO 3.1.2 uses the term "with the reactor in the reference core condition" that is defined in Technical Specification 1.0, to replace the CTS term "above cold, clean critical." LCO 3.1.2 maintains the existing limit of \$3.00 for excess reactivity with the clarification that this limit is equal to 2.19% $\Delta k/k$ based on an effective beta of 0.0073.

SR 4.1.2 adds a new requirement that excess reactivity be verified to be within the limits of LCO 3.1.2 annually and following any significant change to the core or any control rod. An explanation of this requirement and justification of the frequency for performing this verification is presented in the Technical Specification Bases.

LCO 3.1.3, Shutdown Margin (SDM), maintains the existing requirement in CTS 5.3.1. LCO 3.1.3 substitutes the term "Shutdown Margin" that is defined in Technical Specification 1.0, for the CTS explanation of the requirement (i.e., reactor shall be subcritical by a minimum margin of

0.50 dollars when the maximum worth rod is fully withdrawn from the core). LCO 3.1.3 maintains the existing limit of $0.50 for excess reactivity with the clarification that this limit is equal to 0.365% Δk/k based on an effective beta of 0.0073.

SR 4.1.3 adds a new requirement that SDM be verified to be within the limits of LCO 3.1.3 annually and following any significant change to the core or any control rod. An explanation of this requirement and justification of the frequency for performing this verification is presented in the Technical Specification Bases.

LCO 3.1.4, Reactivity Coefficient, maintains the existing requirement in CTS 5.1.3 that the reactivity decrement at full power is ≥ $0.80 when measured with respect to source range power level. LCO 3.1.4 maintains the existing limit of $0.80 for the reactivity decrement with the clarification that this limit is equal to 0.584% Δk/k based on an effective beta of 0.0073. Related requirements in CTS 5.1.3 for the bath temperature coefficient and prompt fuel temperature coefficient are maintained as design requirements in proposed TS 5.3.1(4)(a).

SR 4.1.4 adds a new requirement that the reactivity decrement be verified to be within the limits of LCO 3.1.4 annually and following any significant change to the core or any control rod. An explanation of this requirement and justification of the frequency for performing this verification is presented in the Technical Specification Bases.

LCO 3.1.5, Core Configuration, maintains the existing requirement in CTS 5.1.1 for the number of TRIGA fuel elements and the maximum mass of U-235 that may be included in the core.

SR 4.1.5(1) adds a new requirement for a core inventory following the addition of any fuel elements to the reactor core to verify that the reactor core lattice contains no more than 90 TRIGA type fuel elements prior to reactor startup

SR 4.1.5(2) adds a new requirement for an administrative review to verify that the reactor core lattice contains ≤ 3.30 kg of U-235 prior to reactor startup following the addition of any fuel elements to the reactor core.

SR 4.1.5(3) adds a new requirement that a minimum of 20% of the fuel elements in the reactor core lattice must be examined by visual inspection annually in accordance with guidelines established by the fuel manufacturer.

SR 4.1.5(4) adds a new requirement that each fuel element in the core must have been examined by visual inspection within the previous five years. Typically, an administrative review performed in conjunction with the annual performance of SR 4.1.5(3) ensures this requirement is met.

3.2 REACTOR CONTROL AND SAFETY SYSTEMS

LCO 3.2.1, Control Rods, maintains the existing requirement in CTS 5.3.2. LCO 3.2.1 adds the clarification that the three operable control rods include 1 safety rod, 1 shim rod and 1 regulating rod. This clarification ensures that the specific operating requirements associated with these specific control rods established in USAR Chapter 4, Reactor, and TS 3.2.5, Reactor Sequence and Interlocks, are met. As explained in the Bases, a control rod is considered operable if it is capable of being fully inserted both manually and within the limits of LCO 3.2.2 following an RPS scram signal. LCO 3.2.1 provides clarification that this requirement is applicable only when the reactor is in the reactor operating condition. This applicability is acceptable because automatic or manual insertion of control rods do not provide any safety function when the reactor is already subcritical (i.e., subcritical by at least 0.73% $\Delta k/k$ ($1.00)).

No specific SR is required to verify LCO 3.2.1 is met because free movement of each control rod for insertion is verified by the performance of SR 4.2.2 (1) and whenever the reactor is shutdown.

LCO 3.2.2, Scram Time, maintains the existing requirement in CTS 5.3.3. LCO 3.2.2 provides clarification that this requirement is applicable only when the reactor is in the reactor operating condition. This applicability is acceptable because automatic or manual insertion of control rods do not provide any safety function when the reactor is already subcritical (i.e., subcritical by at least 0.73% $\Delta k/k$ ($1.00)).

SR 4.2.2 adds a new requirement that scram times be measured and verified to be within the limits of LCO 3.2.2 semiannually and following any significant change to the core or any significant work on the control rods or the control rod drive system.

LCO 3.2.3, Reactivity Insertion Rate, maintains the existing requirement in CTS 5.3.2. LCO 3.2.3 provides clarification that this requirement is applicable only when the reactor is in the reactor operating condition or reactor shutdown condition. This clarification is acceptable because rod withdrawal is not possible when in the reactor secured condition which negates the need for limits on reactivity insertion rates achieved by rod withdrawal.

SR 4.2.3(1) adds a new requirement that the reactivity worth of each control rod must be measured annually and following significant changes to the core or any control rod. The annual frequency for verification of control rod reactivity worth is consistent with the expected slow change in reactivity worth over time.

SR 4.2.3(2) adds a new requirement that the withdrawal speed of each control rod is measured semiannually and the maximum rate of reactivity addition verified to be within the limits of LCO 3.2.3.

LCO 3.2.4, Reactivity Insertion Monitoring, maintains the existing requirement in CTS 6.6.

SR 4.2.4 maintains the existing requirement in CTS 6.6 that subcritical multiplication levels be plotted in accordance with the requirements of LCO 3.2.4 from start of control rod withdrawal until reactor criticality.

LCO 3.2.5, Reactor Sequence and Interlocks, maintains the existing requirements in CTS 6.4 and CTS 6.5. CTS 6.4 specifies that reactor sequence and interlocks must be operable "while fuel is in the core." LCO 3.2.5 specifies that reactor sequence and interlocks must be operable only when the reactor is in the reactor operating condition or reactor shutdown condition. This change is acceptable because rod withdrawal is not possible when in the reactor secured condition which negates the need for limits on control rod withdrawal sequence.

SR 4.2.5 adds a new requirement for annual verification of the operability of each of the interlocks required by LCO 3.2.5. Annual verification is acceptable because the interlocks are highly reliable. Additionally, the interlocks are backups to the primary methods for preventing inadvertent reactivity additions which are operating procedures and operator training.

LCO 3.2.6, Reactor Protection System (RPS) Instrumentation, and Table 3.2-1 maintain the existing requirements in CTS 6.2 and CTS Table 2. Functions in CTS Table 2 that were capable of actuating a reactor scram were maintained in Table 3.2-1, Reactor Protection System. Functions in CTS Table 2 that monitored reactor parameters but are not capable of actuating a reactor scram were maintained in Table 3.2-2, Reactor Monitoring System. Functions in CTS Table 2 for area radiation monitoring and effluent monitoring were maintained in Table 3.5-1, Criticality Alarm, and Area and Effluent Radiation Monitoring Instrumentation. Design details for nuclear instrumentation channels in CTS Table 1 are being relocated to USAR Chapter 7, Instrumentation and Control Systems.

Table 3.2-1, footnote (a) maintains the allowance in CTS Table 1, footnote (a) and **Table 3.2-1, footnote (b)** maintains the allowance in CTS Table 1, footnote (b).

Table 3.2-1, footnotes (c) and (d) were added to establish required actions and completion times for inoperable RPS channels. Specifically, Table 3.2-1, footnote (c) establishes a requirement that an inoperable RPS channel must be restored to operable or the reactor must be placed in the secured condition within 8 hours of the discovery that the channel is inoperable. The allowable out of service time of 8 hours for any one RPS channel is acceptable because none of the RPS channels are assumed to function to prevent, detect, or mitigate the consequences of any of the accidents or transients discussed in USAR Chapter 13, Accident Analyses. Additionally, one or more alternate methods exist for the operator to monitor the parameter with the inoperable instrumentation and, if necessary, initiate a manual reactor scram.

The TS Bases for LCO 3.2.6 provides a detailed justification for the actions that allow operation to continue for a limited time with an inoperable RPS channel.

SR 4.2.6(1) adds a new requirement for a channel check of each of the applicable channels listed in Table 3.2-1 prior to exiting the reactor secured condition on the first reactor startup each day.

SR 4.2.6(2) adds a new requirement for a channel test of each of the applicable channels listed in Table 3.2-1 be performed semiannually.

SR 4.2.6(3) adds a new requirement for a channel calibration of each of the applicable channels listed in Table 3.2-1 be performed prior to initial use.

LCO 3.2.7, Reactor Monitoring System (RMS) Instrumentation, and Table 3.2-2 maintain the existing requirements in CTS 6.2 and CTS Table 2. Functions in CTS Table 2 that were capable of actuating a reactor scram were maintained in Table 3.2-1, Reactor Protection System. Functions in CTS Table 2 that monitored reactor parameters but are not capable of actuating a reactor scram were maintained in Table 3.2-2, Reactor Monitoring System. Functions in CTS Table 2 for area radiation monitoring and effluent monitoring were maintained in Table 3.5-1, Criticality Alarm, and Area and Effluent Radiation Monitoring Instrumentation.

Table 3.2-2, footnote (a) is being added to the TS to establish required actions and completion times for inoperable RMS channels. Specifically, these footnotes specify that an alternate method of monitoring the parameter must be established within 24 hours from the discovery that a bridge crane location or reactor water radioactivity channel is inoperable. The alternate monitoring method may be substituted for the required channel for maximum of 30 days from the time the channel is discovered to be inoperable. The TS Bases for LCO 3.2.7 provides a detailed justification for the actions that allow operation to continue for a limited time with an inoperable RMS channel.

Table 3.2-2, footnote (b) is being added to the TS to require that primary coolant conductivity be verified using an alternate method within 7 days of the discovery that the conductivity channel is inoperable and every 7 days thereafter. The TS Bases for LCO 3.2.7 provides a detailed justification for the actions that allow operation to continue for a limited time with an inoperable RMS channel.

Table 3.2-2, footnote (c) is being added to the TS to require that reactor demineralizer water flow be verified using an alternate method immediately upon discovery that the demineralizer water flow channel is inoperable and every 24 hours thereafter. The TS Bases for LCO 3.2.7 provides a detailed justification for the actions that allow operation to continue for a limited time with an inoperable RMS channel.

SR 4.2.7(1) is being added to the TS to require a channel check of each of the applicable channels listed in Table 3.2-2 prior to exiting the reactor secured condition on the first reactor startup each day.

SR 4.2.7(2) is being added to the TS to require channel tests of each of the applicable channels listed in Table 3.2-2 be performed semiannually except for the reactor water radioactivity channel which must be tested prior to exiting the reactor secured condition on the first reactor startup each day.

SR 4.2.7(3) is being added to the TS to require channel calibrations of each of the applicable channels listed in Table 3.2-2 be performed prior to initial use except for the reactor water radioactivity channel which shall be calibrated biennially.

3.3 COOLANT SYSTEMS

LCO 3.3.1, Reactor Tank Water Level, maintains the existing requirement in CTS 4.1 that reactor tank water level must be maintained ≥ 16 feet above the top of the active core. LCO 3.3.1 adds a new clarification that this requirement is applicable only when the reactor is in the reactor operating condition or reactor shutdown condition. This change will allow water level to be reduced to less than 16 feet above the top of the active core when the reactor is in the secured condition if required for maintenance. This change is acceptable because the need for radiological shielding above the core is significantly reduced when the reactor is not critical. Additionally, the need for the water above the core to mitigate the consequences of a significant release of activity from a fuel clad defect is significantly reduced when the reactor is secured.

Although a specific requirement for periodic verification that reactor tank water level is within required limits was not added to the TS, performance of the channel checks, channel tests, and channel calibrations for the RPS functions that monitor reactor tank water level in SR 4.2.6.1, SR 4.2.6.2, and SR 4.2.6.3 are sufficient to verify that LCO 3.3.1 is being met.

LCO 3.3.2, Primary Coolant Water Temperature, maintains the existing requirement in CTS 4.1 that reactor tank water temperature must be maintained ≥ 60 °F and ≤ 130 °F. LCO 3.3.2 adds a new clarification that this requirement is applicable only when the reactor is in the reactor operating condition or reactor shutdown condition. This change will clarify that there is no requirement to maintain reactor water temperature when the reactor is in the secured condition. This change is acceptable because maintaining the reactor tank water temperature ≥ 60 °F satisfies an assumption that ensures the bath temperature coefficient of reactivity remains negative and is not applicable when the reactor is secured. Maintaining the reactor tank water temperature ≤ 130 °F prevents damage to ion exchanger resin in the demineralizer system which is not a reactor safety issue.

Although a specific requirement for periodic verification that reactor tank water temperature is within required limits was not added to the TS, performance of the channel checks, channel tests, and channel calibrations for the RPS functions that monitor reactor tank water level in SR 4.2.6.1, SR 4.2.6.2, and SR 4.2.6.3 are sufficient to verify that LCO 3.3.2 is being met.

LCO 3.3.3, Primary Coolant Water Quality, clarifies the existing requirement in CTS 4.2 that corrective action shall be taken to avoid exceeding a pH of 7.5 or a conductivity of 5 μmho/cm. LCO 3.3.3 specifies that reactor tank water must be maintained within the existing limits. LCO 3.3.3 provides clarification that if this requirement is not met, action must be initiated within 24 hours to restore primary coolant water pH and conductivity to within the specified limits. This change is acceptable because conductivity and pH not within the specified limits do not represent an immediate safety concern, especially at the low temperatures at which the ARRR is operated. The adverse effects of not meeting the specified limits are accelerated corrosion and the resulting accumulation of activation products. However, these adverse effects are time dependent and occur very slowly. Therefore, actions to identify the cause and prevent recurrence of poor water quality may be delayed but should be initiated within 24 hours and completed within a reasonable period of time. Corrective action typically involves identification and elimination of the source of contamination of the reactor coolant and/or replacement of ion exchange resin.

SR 4.3.3(1) maintains the existing requirement in CTS 4.2 for monthly verification that reactor coolant pH and conductivity are within the limits of LCO 3.3.3.

SR 4.3.3(2) adds a new requirement that the radioactivity level of the primary coolant be analyzed annually. There is no Technical Specification limit for reactor water radioactivity. However, reactor water radioactivity is monitored by a radiation monitor required by LCO 3.2.7 and Table 3.2-2 to provide an early indication of a fuel element cladding failure. Sampling and counting the reactor tank water within 24 hours and every day thereafter is an adequate alternate for the reactor water radioactivity channel. Annual analysis of the radioactivity level of the primary coolant is intended to identify any unusual radioactivity in the coolant.

3.4 VENTILATION SYSTEMS

LCO 3.4.1, Control Room, maintains the existing requirement in CTS 3.1.3 that the control room be maintained at a positive pressure with respect to the reactor room. LCO 3.4.1 adds the clarification that this requirement is applicable only when in the reactor operating condition or reactor shutdown condition and during movement of irradiated fuel. This clarification is acceptable because a significant release of activity from failed fuel is improbable when the reactor is in the secured condition.

SR 4.4.1 adds a new requirement for quarterly verification that the control room is being maintained at a positive air pressure with respect to the reactor room.

LCO 3.4.2, Reactor Building Circulation Fans, maintains the existing requirement in CTS 3.1.1 that all air conditioning systems and building circulating fans in or normally open to the reactor high bay area have the capability to be shut off from a single control in the control room. LCO 3.4.2 adds the clarification that this requirement is applicable only when in the reactor operating condition or reactor shutdown condition and during movement of irradiated fuel. This clarification is acceptable because a significant release of activity from failed fuel is improbable when the reactor is in the secured condition.

SR 4.4.2 adds a new requirement for quarterly verification of proper operation of the emergency shutoff from the control room of all air conditioning systems and building circulating fans that are in or normally open to the high bay area.

3.5 CRITICALITY ALARM, RADIATION AND RADIOACTIVE EFFLUENT MONITORING

LCO 3.5.1(1), Criticality Alarm and Area Radiation Monitoring, and Table 3.5-1 maintain the existing requirements in CTS 7.1 and CTS Table 2.

Table 3.5-1, footnote (a) adds a new allowance that an alternate method of monitoring the parameter may be established within 8 hours if the criticality alarm and area radiation monitor becomes inoperable. The Bases for LCO 3.5.1 provides a description and justification for this allowance.

Table 3.5-1, footnote (b) adds a new requirement that any movement of fuel elements be stopped immediately if the criticality alarm and area radiation monitor is inoperable during movement of a fuel element. The Bases for LCO 3.5.1 provides a description and explanation of this requirement.

SR 4.5.1 (1) adds a new requirement for a quarterly channel test of the criticality alarm and area radiation monitoring system.

SR 4.5.1(2) adds a new requirement for a biennial channel calibration of the criticality alarm and area radiation monitoring system.

LCO 3.5.1 (2) adds a new requirement that radiation sensitive badges be located at strategic locations within the reactor building for radiation analysis at all times. These badges are used for routine monitoring of facility radiation levels.

SR 4.5.1(3) adds a new requirement for quarterly reading and analysis of the radiation sensitive badges placed at strategic locations within the reactor building.

LCO 3.5.2, Gaseous Effluent Monitoring, maintains the existing requirement in CTS 6.2 and Table 2. Table 3.5-1, footnote (a) provides a new allowance that an alternate method of monitoring the parameter shall be established within 8 hours if the gaseous effluent monitor becomes inoperable. The Bases for LCO 3.5.2 provides a description and justification for this allowance.

SR 4.5.2 (1) adds a new requirement that a channel test of the gaseous effluent monitoring system be performed prior to exiting the reactor secured condition on the first reactor startup each day.

SR 4.5.2(2) adds a new requirement for a biennial channel calibration of the gaseous effluent monitoring system.

LCO 3.5.3, Particulate Effluent Monitoring, adds a new requirement that a particulate sample be withdrawn continuously from the reactor room and collected on filter paper whenever the reactor is in the reactor operating condition or reactor shutdown condition and during movement of irradiated fuel. This monitor is primarily used for environmental monitoring.

SR 4.5.3(1) adds a new requirement for daily verification that the reactor room particulate sampler is operating each day the reactor is operated.

SR 4.5.3(2) adds a new requirement that the reactor room particulate sample be counted monthly.

LCO 3.5.4, Post Accident Radiation Monitoring, maintains the existing requirement in CTS 7.6.

SR 4.5.4 adds a new requirement for annual verification that radiation detector packets containing threshold detectors are in place.

LCO 3.5.5, Portable Radiation Monitoring Instruments, maintains the existing requirements in CTS 7.4 and CTS 7.5.

SR 4.5.5 adds a new requirement for quarterly calibrations of each required portable radiation monitoring instrument.

LCO 3.5.6, Radioactive Effluent Limits, is a restatement of existing requirements in 10 CFR 20 that normal releases of radioactive effluents from the ARRR must not exceed the specified limits.

SR 4.5.6(1) adds a new requirement that Ar-41 production must be determined annually.

SR 4.5.6(2) adds a new requirement for an annual administrative verification that normal releases of radioactive effluents from reactor operation did not exceed 10 CFR 20 limits.

3.6 EXPERIMENTS

LCO 3.6.1, Evaluation and Approval of Experiments, maintains the existing requirement in CTS 9.2.

SR 4.6.1 adds an explicit requirement that the reactor operator verify that experiments have been evaluated and approved in accordance with the requirements of TS 6.5.1 prior to insertion of the experiment into the reactor as required by LCO 3.6.1.

LCO 3.6.2, Reactivity Limits during Experiments, maintains the existing requirements in CTS 9.3, CTS 9.4, CTS 9.10, CTS 5.1.2, and CTS 5.3.1.

LCO 3.6.3, Special Nuclear Material (SNM) included in Experiments, maintains the existing requirement in CTS 9.9.

SR 4.6.3 adds an explicit requirement that the reactor operator verify that experiments meet the requirements of LCO 3.6.3 prior to insertion of the experiment into the reactor.

LCO 3.6.4, Materials Used during Experiments, maintains the existing requirement in CTS 9.7.

LCO 3.6.5, Explosive Materials in Experiments, maintains the existing requirements in CTS 9.11.4, CTS 9.11.4.1, CTS 9.11.4.2, CTS 9.12.3, and CTS 9.12.4.

SR 4.6.5 adds an explicit requirement that Aerotest Operations, Inc. establishes and implements administrative controls that ensure the requirements of LCO 3.6.5 are being met.

LCO 3.6.6, Configuration and Potential Failure Mechanisms, maintains the existing requirements in CTS 9.5, 9.6 and 9.8.

LCO 3.6.7, Glory Hole Facility, maintains the existing requirements in CTS 8.3.1 and CTS 8.3.2. LCO 3.6.7 clarifies the requirement in CTS 8.3.2 that "The glory hole shall be purged with CO_2 to prevent formation of excessive amounts of Ar-41" with the following requirements:

(a) A glory hole facility operated with a shield plug shall be purged prior to each insertion of the shield plug.

(b) A glory hole facility operated without a shield plug shall be purged with CO_2 continuously.

SR 4.6.7 maintains the existing requirement in CTS 8.3.2.

LCO 3.6.8, Vertical Tubes, maintains the existing requirements in CTS 8.6.1 and CTS 8.6.2.

SR 4.6.8(1) maintains the existing requirement in CTS 8.6.2 with the added clarification that the purge must be continuous.

SR 4.6.8(2) adds a new requirement that gas samples must be taken near the pool continuously and Argon 41 presence shall be monitored when the vertical tube is inserted.

LCO 3.6.9, Other Irradiation Facilities, maintains the existing requirements in CTS 8.7.1, CTS 8.7.2, and CTS 8.7.3.

LCO 3.6.10, Large Component Irradiation Box, replaces all of the requirements in CTS 8.1 regarding the use of the large component irradiation box with an LCO that prohibits installation of the large component irradiation box in the reactor tank. The large component irradiation box is not being used at the ARRR facility, its future use is not anticipated, and its use is not addressed in the USAR. The USAR must be amended and a Technical Specification change must be approved by the NRC prior to installation of the large component irradiation box.

LCO 3.6.11, Pneumatic Transfer Facility, replaces all of the requirements in CTS 8.2 regarding the use of the Pneumatic Transfer Facility with an LCO that prohibits installation of a pneumatic transfer facility in any reactor core position. The pneumatic transfer facility is not being used at the ARRR facility, its future use is not anticipated, and its use is not addressed in the USAR. The USAR must be amended and a Technical Specification change must be approved by the NRC prior to installation of a pneumatic transfer facility.

3.7 FUEL STORAGE AND TRANSFER

LCO 3.7.1, Fuel Handling Tools, maintains the existing requirement in CTS 11.3 with the clarifications that there is more than one fuel handling tool and that the tools are secured with a locking device rather than in a locked cabinet as specified in the CTS.

SR 4.7.1 adds a new requirement for weekly verification that the requirements of LCO 3.7.1 are being met.

LCO 3.7.2, Fuel Storage in the Reactor Tank, in conjunction with TS 5.4.2, maintains the design requirements in CTS 11.2 for fuel elements stored in the reactor tank that are not part of the core with two changes:

(1) The limit for k_{eff} for fuel elements stored in the reactor tank that are not part of the core was changed from 0.8 to 0.9; and,

(2) A clarification was added that this limit applies to all conditions of light water moderation.

These changes are acceptable because the proposed limit of 0.9 for k_{eff} provides a very substantial margin for the prevention of an inadvertent criticality involving the fuel being stored in the reactor tank. The proposed limit for k_{eff} of 0.9 under all conditions of light water moderation is significantly more conservative than the limit of 0.95 that would be applicable if 10 CFR 50.68, Criticality Accident Requirements, were applicable. No mechanism exists that could result in either an intentional or inadvertent addition of reactivity to the fuel being stored in the reactor tank. The ARRR does not take any credit for the criticality alarm required by CTS 7.1 and CTS Table 2; however, requirements for the criticality alarm are maintained in LCO 3.5.1 and Table 3.5-1. This change makes the ARRR requirements consistent with those adopted by facilities similar to the ARRR. No design changes to the existing design of the storage racks in the reactor tank are contemplated.

SR 4.7.2 adds an explicit requirement for verification that the requirements of LCO 3.7.2 will be met prior to any addition or movement of fuel in the reactor tank.

LCO 3.7.3, Fuel Storage in the Fuel Storage Pits, maintains the existing requirements in CTS 11.1 that allows up to 19 fuel elements containing up to 700 grams of U-235 to be stored in each fuel storage pit. LCO 3.7.3 also maintains the existing requirement in CTS 11.1 that each fuel storage pit containing a fuel element must be secured with a lock and chain.

SR 4.7.3(1) adds an explicit requirement for verification that LCO 3.7.3 requirements will be met for the number of fuel elements and mass of U-235 to be stored in each fuel storage pit prior to any addition or movement of fuel in the fuel storage pit.

SR 4.7.3(2) adds a requirement for quarterly verification that requirements for locking the fuel storage pits in LCO 3.7.3 are met.

LCO 3.7.4, Fuel not in the Reactor and not in Storage, maintains the existing requirement in CTS 11.5.

SR 4.7.4 adds an explicit requirement for verification that the requirements of LCO 3.7.4 will be met prior to any movement or receipt of fuel at the ARRR facility.

LCO 3.7.5, Fuel in Shipping Containers, specifies that fuel may be contained in an approved fuel shipping container and, when an approved fuel shipping container is used, then the licensed limitations on k_{eff} for the container shall apply. This requirement replaces the less specific requirement in CTS 11.3 that a shielded fuel transfer cask must be used for the transfer of highly radioactive fuel elements.

SR 4.7.5 adds an explicit requirement for verification that the requirements of LCO 3.7.5 are met prior to the use of a fuel shipping container.

LCO 3.7.6, Fuel Transfer in the Reactor Tank, maintains the existing requirement in CTS 11.4.

LCO 3.7.7, Fuel Transfer Outside the Reactor Tank maintains the existing requirement in CTS 11.4.

3.8 REACTOR FACILITY

LCO 3.8.1, Reactor Building Alarm, maintains the existing requirement in CTS 3.2.

SR 4.8.1 maintains the existing requirement in CTS 3.2.

LCO 3.8.2, Explosive Material at the Reactor Facility, maintains the existing requirements in CTS 9.11 and CTS 9.12 with two exceptions:

1. CTS 9.11.1 specifies that individual explosive devices must be "encased in metallic sheathing." This requirement is replaced by LCO 3.8.2(3) which specifies, "Only solid or encased explosive materials may be brought into the facility," and LCO 3.8.2(4) which specifies, "Any explosive device containing loose explosive powders shall be completely encased." The CTS requirement that explosives be "encased in metallic sheathing" is based on out-of-date design assumptions regarding small explosive devices. The out-of-date design assumptions are that most small explosive devices consist of explosive powers that must be completely encased in sheathing to prevent the long term buildup of loose explosive powders in a facility and that various metals were the preferred material for making the sheathing. Since the time the ARRR Technical Specifications were written, the explosives industry has evolved to include many new explosives that are blended with plastics to form high density structural components that do not require separate sheathing. Additionally, when sheathing is required, plastic and other non-metallic cases have become common in products such as initiators, detonators, etc. This change is acceptable because the use of either explosives blended with plastics to form high density structural components or encasements made of plastic and other non-metallic material is effective in preventing the long term buildup of loose explosive powders in a facility. These proposed changes were discussed with knowledgeable representatives of the Lawrence Livermore National Laboratory and a commercial explosive company, Teledyne RISI. The proposed requirements are consistent with safety practices at these facilities.

2. CTS 9.11.3.1 specifies that only explosive devices that will be radiographed within 4 hours, including preparation time, may be removed from storage. CTS 9.11.3.2 requires the maintenance of an accountability log for explosives. These requirements were established to ensure accountability of the explosive material by allowing only explosives in the process of being prepared for radiography to be removed from the designated storage areas. The requirements ensure that all explosive material are accounted for each day and that all explosive material not packaged for shipment was returned to storage areas at the end of each workday. LCO 3.8.2(9) maintains these requirements except that the time that explosives may be removed from storage is extended from 4 hours to 8 hours. LCO 3.8.2(10) clarifies the requirement for accountability of explosives by requiring that the accountability log is updated daily. The 8 hour limit is necessary because some explosive devices may require as much as 8 hours for preparation

(setup) and completion of radiography. This change is acceptable because the 8 hour limit ensures that all explosives are returned to designated areas and accounted for each day.

SR 4.8.2 maintains the existing requirements in CTS 9.11 and CTS 9.12.

4.0 SURVEILLANCE REQUIREMENTS

4.0 GENERAL REQUIREMENTS

SR 4.0.1 is added to clarify that SRs must be met whenever the associated LCO is required to be met. The SR 4.0.1 Bases provide the additional clarification that, unless otherwise specified, an SR must be completed successfully before the reactor can be placed in a condition where the associated LCO is applicable. If an SR is not met or is not performed at the required frequency, it must be assumed that the associated LCO is not met.

SR 4.0.1 provides a new allowance that may be used for extending the interval between performances of SRs to facilitate flexible scheduling consistent with guidelines provided in ANSI/ANS 15.1. This allowance is acceptable because the short extension to the SR performance interval does not significantly degrade the reliability that results from performing the SR at its specified frequency. This is based on the recognition that most SRs satisfy required acceptance criteria when the SR is performed. The Bases clarify that the allowance in SR 4.0.2 is not intended to be used to routinely increase the interval between required performances of an SR.

5.0 DESIGN FEATURES

5.1 SITE AND FACILITY DESCRIPTION

TS 5.1.1 maintains the existing design requirement in CTS 2.2.

TS 5.1.2 maintains the existing design requirement in CTS 2.2.

TS 5.1.3 maintains the existing design requirement in CTS 2.3. However, the statement that the principal activities within the ARRR exclusion area include use of a hot cell and chemistry laboratory were deleted because the hot cell was removed from the ARRR building in 1969 and the chemistry laboratory is only occasionally used.

TS 5.1.4 maintains the existing design requirement in CTS 3.1.

TS 5.1.5 maintains the existing design requirement in CTS 3.1.2. A description of the function of the gravity ventilators was added to USAR Sections 6.1 and 9.1.

TS 5.1.6 maintains the existing design requirement in CTS 3.2. Requirements in CTS 3.2 for monitoring and testing the alarm system were maintained in LCO 3.8.1 and SR 4.8.1.

5.2 REACTOR COOLANT SYSTEM

TS 5.2, Reactor Coolant System, was added to the Technical Specifications to provide a brief description of the reactor coolant system consistent with the guidelines for Technical Specification content in ANSI/ANS 15.1 and Technical Specifications for facilities similar to ARRR. A detailed description of the ARRR reactor coolant system is provided in USAR Chapter 5, Reactor Coolant Systems.

5.3 REACTOR CORE AND FUEL

TS 5.3.1(1), Reactor Core, was added to the Technical Specifications to provide a brief description of the essential elements of the reactor core consistent with the guidelines for Technical Specification content in ANSI/ANS 15.1 and Technical Specifications for facilities similar to ARRR. A detailed description of the ARRR reactor core is provided in USAR Chapter 4, Reactor.

TS 5.3.1(2), which describes the need to add fuel elements to compensate for fuel burnup, was added to the Technical Specifications to provide a brief description of the ARRR practice of periodically adding fuel elements. This statement is consistent with the description of ARRR operating practice described in USAR Chapter 4, Reactor. Related limits that the ARRR reactor may not contain more than 90 TRIGA type fuel elements or 3.30 kilograms of U-235 in CTS 5.1.1 are maintained in LCO 3.1.5 and SR 4.1.5.

TS 5.3.1(3) maintains the existing design requirements in CTS 5.2.1.

TS 5.3.1(4) maintains the existing design requirements in CTS 5.1.3 and CTS 5.1.4.

TS 5.3.2, Reactor Fuel, was added to the Technical Specifications to provide design requirements of the TRIGA fuel elements used in the ARRR reactor core. This description is consistent with the guidelines for Technical Specification content in ANSI/ANS 15.1 and Technical Specifications for facilities similar to ARRR. A detailed description of the ARRR reactor fuel is provided in USAR Chapter 4, Reactor.

TS 5.3.3, Control Rods, was added to the Technical Specifications to provide design requirements of the control rods used in the ARRR reactor core and is consistent with existing requirements in CTS 5.3.2. This description is consistent with the guidelines for Technical Specification content in ANSI/ANS 15.1 and Technical Specifications for facilities similar to ARRR. A detailed description of the ARRR control rods is provided in USAR Chapter 4, Reactor.

5.4 FISSIONABLE MATERIAL STORAGE

TS 5.4.1 maintains the existing design requirements in CTS 11.1. Related limits that each ARRR spent fuel pit may not contain more than 19 elements or 700 grams of U-235 are maintained in LCO 3.7.3 and SR 4.7.3.

TS 5.4.2 maintains the existing design requirements in CTS 11.2 except that the limit for keff for fuel elements stored in the reactor tank that are not part of the core was changed from 0.8 to 0.9 and the qualification was added that this limit applies to all conditions of light water moderation. No changes to the existing design of the storage racks are contemplated. This change is acceptable because the proposed limit for keff of 0.9 provides a very substantial margin for the prevention of an inadvertent criticality involving the fuel being stored in the reactor tank. No mechanism exists that could result in either an intentional or inadvertent addition of reactivity to the fuel being stored in the reactor tank. The proposed limit for keff of 0.9 under all conditions of light water moderation is significantly more conservative than the limit of 0.95 that would be applicable if 10 CFR 50.68, Criticality Accident Requirements, were applicable. The ARRR does not take any credit for the criticality alarm required by CTS 7.1 and CTS Table 2; however, requirements for the criticality alarm are maintained in LCO 3.5.1 and Table 3.5-1. This change makes the ARRR requirements consistent with those adopted by facilities similar to the ARRR.

5.5 EXPERIMENTAL FACILITIES

TS 5.5.1 maintains the existing design description and requirements for the neutron radiography facility in CTS 8.4.

TS 5.5.2 maintains the existing design descriptions and requirements for the thermal column and vertical irradiation tubes in CTS 8.5 and CTS 8.6.1.

TS 5.5.3 maintains the existing design description and requirements for the other irradiation facilities in CTS 8.7.1 and CTS 8.7.2.

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION

TS 6.1.1 and Figure 6.1, Aerotest Operations, Inc, Organization Chart, are being added to the TS to describe requirements for the organizational structure and chain of responsibility for the ARRR plant consistent with the guidelines provided in ANSI/ANS 15.1. In addition to providing more detail, Figure 6.1 is consistent with the assignment of responsibility shown in the CTS with the added clarification that the Reactor Supervisor reports to the President of Aerotest Operations, Inc. through the General Manager. Information describing Aerotest Operations corporate structure and corporate reporting relationships was deleted because this information is more appropriately addressed in the license.

TS 6.1.2, Responsibility, is being added to provide a statement of responsibilities for management personnel and to provide an explicit allowance for the designation of qualified alternates for any position in Figure 6.1, Aerotest Operations, Inc, Organization Chart, consistent with the guidelines provided in ANSI/ANS 15.1.

TS 6.1.3(1)(a) maintains the existing requirement in CTS 10.1.

TS 6.1.3(1)(b) maintains the existing requirement in CTS 10.1 with the added clarification that the second person at the facility must be capable of performing prescribed written instructions. Additionally, TS 6.1.3(1)(b) provides an explicit allowance that unexpected absence for as long as two hours to accommodate a personal emergency is acceptable provided immediate action is taken to obtain a replacement. This allowance applies to TS 6.1.3(1)(b) only and would allow reactor operation to continue during the temporary absence of the second person at the facility provided that TS 6.1.3(1)(a) is met. This allowance is consistent with the guidelines provided in ANSI/ANS 15.1.

TS 6.1.3(1)(c) establishes a new requirement that a designated senior reactor operator must be readily available and on call with specific restrictions about what constitutes readily available and on call. This requirement ensures the prompt availability of a senior reactor operator to evaluate and respond to any unplanned event and is consistent with the guidelines provided in ANSI/ANS 15.1.

TS 6.1.3(2) establishes a new requirement that a senior reactor operator must be at the facility during initial startup and approach to power and during relocation of any in-core experiment with reactivity worth ≥ 0.73% $\Delta k/k$ (≥ \$1.00). TS 6.1.3(2), in conjunction with TS 3.7.6, maintains the existing requirement in CTS 11.4 that a senior reactor operator must be present during fuel or control rod relocations within the reactor core region. These requirements ensure that a senior reactor operator is at the facility during any activity that could result in a large change in core reactivity and are consistent with the guidelines provided in ANSI/ANS 15.1.

TS 6.1.3(3) establishes a new requirement that a list of reactor facility personnel by name and telephone number must be readily available in the control room for use by the operator. This list must include management personnel, radiation safety personnel, and other operations personnel. This requirement ensures the prompt availability of an appropriate mix of ARRR personnel to evaluate and respond to any unplanned event and is consistent with the guidelines provided in ANSI/ANS 15.1.

TS 6.1.4, Selection and Training of Personnel, establishes a new requirement that the selection, training, and requalification of reactor operators must meet or exceed the requirements of American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4-1988, Sections 4 through 6. TS 6.1.4 establishes a specific requirement that is consistent with current practice.

6.2 REVIEW AND AUDIT FUNCTION

TS 6.2.1 (1) through (5), Reactor Safeguards Committee (RSC) Composition and Qualifications, maintains the existing requirement in CTS 12.1.3. TS 6.2.1 adds the clarification that RSC members and alternates are appointed by and report to the President, Aerotest Operations, and that RSC members must collectively represent a broad spectrum of expertise in reactor, radiological, industrial, and explosive safety. Additionally, TS 6.2.1(5) stipulates that the chairman and at least two individuals who are qualified in these technologies must be members from outside the operating organization.

TS 6.2.1 (6) adds a new requirement that the Reactor Supervisor and Radiological Safety Officer (both members of the operating organization shown on Figure 6.1) must be members of the RSC. TS 6.2.1 (6)(a) and TS 6.2.1 (6)(b) maintain existing requirements in CTS 12.1.1, 12.1.2, 12.1.4, and 12.1.5 that specify the responsibilities and minimum qualifications for the Reactor Supervisor and Radiological Safety Officer.

TS 6.2.1 (7) adds a new allowance that alternates may serve during the absence of regular RSC members if the alternates are qualified and approved.

TS 6.2.2, RSC Charter and Rules, maintains the existing requirement in CTS 12.1.3 that the RSC must meet on the call of the chairman and must meet at least annually. TS 6.2.2 provides the clarification that the RSC must meet more frequently as circumstances warrant, consistent with effective monitoring of facility activities. TS 6.2.2 also provides the clarification that official RSC action requires a quorum of not less one-half of the membership where the operating staff does not constitute a majority. Finally, TS 6.2.2 requires that dissemination, review, and approval of RSC meeting minutes must be conducted in a timely manner.

TS 6.2.3, RSC Review Function, and TS 6.2.4, RSC Audit Function, maintain all of the existing requirements in CTS 12.1.3.1 through 12.1.3.6 and add additional and more specific review, audit, and reporting requirements that are consistent with the existing RSC charter and ANSI/ANS 15 guidelines.

6.3 RADIATION SAFETY

TS 6.3.1, Radiation Safety Program, clarifies that the ARRR must maintain a radiation safety program that complies with the requirements of 10 CFR 20 and adds a new requirement that this program must conform to the guidelines of American National Standard for Radiation Protection at Research Reactor Facilities, ANSI/ANS-15.11-1993.

TS 6.3.2, Radiological Safety Officer (RSO), together with TS 6.2.1(6)(b), clarifies the duties and reporting relationship of the RSO consistent with existing requirements in CTS 12.1.2.

6.4 PROCEDURES

TS 6.4.1, Written Procedures, specifies that written procedures be "prepared, reviewed, and approved" for startup, operation, and shutdown of the reactor and for loading, unloading and movement of fuel within the reactor consistent with the guidelines in ANSI/ANS 15.1. TS 6.4.1 maintains the existing requirements in CTS 12.2.1.1, 12.2.1.2 and 12.2.1.5 that written procedures be provided and followed for these activities. TS 6.4.1 eliminates specific reference to procedures for actions in response to specific and foreseen potential malfunctions including responses to alarms, leaks and abnormal reactivity changes which are now implicitly addressed as operating procedures. TS 6.4.1 adds new requirements for written procedures for personnel radiation protection, administrative controls for operations and maintenance and for the conduct of irradiations and experiments, and implementation of required plans such as emergency or security plans.

TS 6.4.2, Procedure Approval, maintains the existing requirement in CTS 12.1.3.2 that the RSC or designated alternates review and approve procedures and adds an explicit requirement that the Reactor Supervisor or designated alternate also review and approved procedures and that all such reviews and approvals must be documented in a timely manner.

TS 6.4.3, Procedure Changes, maintains the existing requirement in CTS 12.2.2.

TS 6.4.4, Deviation from Procedures, adds an explicit allowance that temporary deviations from the procedures may be made by a Senior Reactor Operator in order to deal with special or unusual circumstances or conditions. Such deviations must be documented and reported to the Reactor Supervisor or designated alternates. This allowance is consistent with the guidelines in ANSI/ANS 15.1.

6.5 EXPERIMENTS

TS 6.5.1, Experiment Review and Approval, maintains the existing requirements in CTS 12.1.3.2, CTS 9.1 and CTS 9.2 with a new explicit requirement that the Reactor Supervisor must also review and approve experiments. TS 6.5.1 adds a new requirement that review and approval of experiments must adhere to the guidelines for review and documentation that ARRR established in the Aerotest Experiment Type Review (AETR) form.

TS 6.5.2, Experiment Performance, adds explicit new requirements regarding the responsibility of the Reactor Supervisor and ARRR staff to ensure that experiments conform to approved experiment types and restrictions.

TS 6.5.3, Changes to Experiments, maintains the existing requirement in CTS 12.1.3.2 that changes to experiments must be approved by the RSC and provides an explicit allowance that minor changes that do not significantly alter the experiment may be approved by the Reactor Supervisor or designated alternate. This change is needed to allow the Reactor Supervisor or designated alternate to correct minor discrepancies in test procedures discovered during performance.

6.6 REQUIRED ACTIONS

TS 6.6.1, Action to be Taken in Case of Safety Limit (SL) Violation, and **TS 6.6.2**, Action to be Taken for Reportable Event described 6.7.3, add new requirements for responding reportable events that are consistent with the guidelines in ANSI/ANS 15.1.

6.7 REPORTS

TS 6.7.1, Operating Reports, adds a new requirement for Aerotest to make annual operating reports to the NRC covering the operation of the facility during the previous calendar year and that this report must include: a summary of changes, tests, and experiments as required by 10 CFR 50.59; annual occupational exposure reports as required by 10 CFR 20.2206; Annual Material Status Reports as required by NUREG/BR-0007; and, Nuclear Material Transaction Reports as required by NUREG/BR-0006.

TS 6.7.2, Changes to the Facility or Organization, adds a new requirement for a written report to the NRC within 30 days of a permanent change in the facility organization involving the President, Aerotest Operations, Inc. or the General Manager. TS 6.7.2 maintains the existing requirement in ARRR License Condition 2.D.3 for prompt reporting of any significant changes in the transient or accident analysis as described in the USAR.

TS 6.7.3, Reportable Events, adds new requirements for reports to the NRC involving violation of a safety limit, release of activity from the site above allowed limits, violation of limiting safety system settings, violation of LCOs, equipment failures, uncontrolled reactivity changes, abnormal degradation of reactor fuel or cladding, and inadequate procedures. TS 6.7.3 is consistent the existing requirements in ARRR License Condition 2.D.1 and 2.D.2 except that TS 6.7.3 allows 14 days versus 10 days in ARRR License Condition 2.D.1 for reporting conditions that could have prevented a system from performing a safety function and 30 days in ARRR License Condition 2.D.2 for reporting that the ARRR was operated at a substantial variance to performance specifications in the Hazards Summary Report or the Technical Specifications. TS 6.7.3 is consistent with guidelines in ANSI/ANS 15.1.

6.8 RECORDS

TS 6.8, Records, adds new requirements for records retention that are consistent with guidelines in ANSI/ANS 15.1. In addition to the new requirements, TS 6.8 maintains all of the existing requirements in ARRR License Condition 2.E. TS 6.8 also maintains the existing requirements in CTS 12.3 regarding records of the irradiation or radiography of explosives except that TS 6.8 provides the clarification that these records must be retained for at least 7 years.

AEROTEST OPERATIONS, INC.

Aerotest Radiography and Research Reactor (ARRR)

DESCRIPTION OF PROPOSED CHANGES TO THE ARRR TECHNICAL SPECIFICATIONS

ATTACHMENT 1

Attachment 1 is a copy of the ARRR current Technical Specifications (CTS) that has been annotated to provide a detailed cross reference between the ARRR proposed Technical Specifications (TS) and the CTS. The annotated copy of the CTS also shows significant changes being proposed to existing requirements.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

AEROTEST OPERATIONS, INC.

DOCKET NO. 50-228

AEROTEST RADIOGRAPHY AND RESEARCH REACTOR (ARRR)

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 1
License No. R-98

1. The Atomic Energy Commission (the Commission) has found that:

   A. The application for transfer of license filed by OEA, Inc. for Aerotest Operations, Inc. (the licensee) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. Construction of the facility has been substantially completed in conformity with Construction Permit No. CPRR-86, and the application, as amended, the provisions of the Act and the rules and regulations of the Commission;

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

   D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

   E. Aerotest Operations, Inc. is technically and financially qualified to possess, use, and operate the facility in accordance with the rules and regulations of the Commission;

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

F. The issuance of this operating license will not be inimical to the common defense and security or to the health and safety of the public, and does not involve a significant hazards consideration;

G. The receipt, possession, and use of byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30 and 70, including Sections 30.33, 70.23, and 70.31;

H. The licensee is qualified to be the holder of the license; and

I. The transfer of the license is otherwise consistent with applicable provisions of law, regulations, and orders issued by the Commission pursuant thereto.

2. Facility Operating License No. R-98, previously issued to Aerojet-General Corporation, is hereby transferred to Aerotest Operations, Inc. and the license is reissued in its entirety to read as follows:

A. This license applies to the Aerotest Radiography and Research Reactor (ARRR), previously called the Aerojet General Nuclear Industrial Reactor (AGNIR), a pool-type nuclear reactor owned by the OEA, Inc. The facility is located at the Aerotest Operations site near San Ramon, California, and is described in the application dated September 14, 1964 (the application), and in supplements thereto, including the application for transfer of license dated April 24, 1974.

And the Updated Final Safety Analysis Report, dated March 2005.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Aerotest Operations, Inc.:

(1) Pursuant to Section 104c of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities", to possess, use, and operate the reactor at the designated location in San Ramon, California, in accordance with the procedures and limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, "Special Nuclear Material", to receive, possess, and use up to 5.0 kilograms of contained uranium 235 in connection with operation of the reactor; and

(3) Pursuant to the Act and 10 CFR Part 30, "Licensing of Byproduct Material", (1) to receive, possess, and use a 2 curie americium-beryllium neutron startup source, and (2) to possess, but not to separate, such byproduct material as may be produced by operation of the reactor.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

LCO 3.1.1
Def: RTP

The licensee is authorized to operate the facility at steady state power levels not in excess of 250 kilowatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 8 Amendment No. 9.

D. Reports

In addition to reports otherwise required under this license and applicable regulations:

(1) The licensee shall report in writing to the Commission within 10 days of its observed occurrence any incident or condition relating to the operation of the facility which prevented or could have prevented a nuclear system from performing its safety function as described in the Technical Specifications or in the Hazards Summary Report.

Updated Final Safety Analysis Report.

(2) The licensee shall report to the Commission in writing within 30 days of its observed occurrence any substantial variance disclosed by operation of the facility from performance specifications contained in the Hazards Summary Report or the Technical Specifications.

(3) The licensee shall report to the Commission in writing within 30 days of its occurrence any significant change in transient or accident analysis, as described in the Hazards Summary Report.

E. Records

DELETED

~~In addition to those otherwise required under this license and applicable regulations, the licensee shall keep the following:~~

TS 6.8.5

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

TS 6.8.5

(1) ~~Reactor operating records, including power levels.~~

(2) ~~Records of in-pile irradiations.~~

(3) ~~Records showing radioactivity released or discharged into the air or water beyond the effective control of the licensee as measured at the point of such release or discharge.~~

(4) ~~Records of emergency reactor scrams, including reasons for emergency shutdowns.~~

F. This amended license is effective as of the date of issuance and shall expire at midnight April 16, 2005.

FOR THE ATOMIC ENERGY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

Attachment:
Change No. 8 to the Technical
Specifications

Amendment No. 9

Date of Issuance: October 22, 1974

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

2 Delete Page

ATTACHMENT TO LICENSE AMENDMENT NO. 1

CHANGE NO. 8 TO APPENDIX A OF TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. R-98

DOCKET NO. 50-228

Revise Appendix A as follows:

1. Change the name of the reactor to "Aerotest Radiography and Research Reactor (ARRR)".
2. Paragraph 2.1 - end the sentence after the words "exclusion area".
3. Paragraph 2.2 and 2.3 - change the abbreviation "AGNIR" in the first sentence to read "ARRR".
4. Paragraph 12.1.1 - revise the paragraph to read as follows:

"12.1.1 The Reactor Supervisor shall have responsibility of the reactor facility. In all matters pertaining to reactor operations and to these Technical Specifications, the Reactor Supervisor shall be responsible to the President, Aerotest Operations, Inc., a wholly-owned subsidiary of Explosive Technology, Inc. The President, Aerotest Operations, Inc. shall report to the Board of Directors of Aerotest Operations, Inc. which includes the Presidents of both OEA, Inc. and Explosive Technology, Inc."

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

ATTACHMENT TO LICENSE AMENDMENT NO. 1

CHANGE NO. 8 TO APPENDIX A OF TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. R-98

DOCKET NO. 50-228

Revise Appendix A as follows:

1. Change the name of the reactor to "Aerotest Radiography and Research Reactor (ARRR)".
2. Paragraph 2.1 - end the sentence after the words "exclusion area".
3. Paragraph 2.2 and 2.3 - change the abbreviation "AGNIR" in the first sentence to read "ARRR".
4. Paragraph 12.1.1 - revise the paragraph to read as follows:

   "12.1.1 The Reactor Supervisor shall have responsibility of the reactor facility. In all matters pertaining to reactor operations and to these Technical Specifications, the Reactor Supervisor shall be responsible to the President, Aerotest Operations, Inc., a wholly-owned subsidiary of Explosive Technology, Inc. The President, Aerotest Operations, Inc. shall report to the Board of Directors of Aerotest Operations, Inc. which includes the Presidents of both OEA, Inc. and Explosive Technology, Inc."

Annotated to show differences between Current Technical Specifications (CTS) and Proposed Technical Specifications (PTS).

APPENDIX A

LICENSE NO. R-98

TECHNICAL SPECIFICATIONS FOR THE

AEROTEST RADIOGRAPHY AND RESEARCH REACTOR(ARRR)

1.0 Definitions

~~1.1~~ ~~Shutdown~~ Reactor Secured

Add Parts 1 and 2

TS 1.0

The reactor, with fixed experiments in place, shall be considered to be shut down (not in operation) whenever all of the following conditions have been met: (a) the console key is in the "off" position and the key is removed from the console and under the control of a licensed operator (or stored in a locked storage area); (b) sufficient control rods are inserted so as to assure the reactor is subcritical by a margin greater than 0.7% 0.73% delta k/k ~~cold, clean critical condition~~; (c) no work is in progress involving refueling operations or maintenance of its control rod mechanisms. when in Reference Core Cond.

~~1.2~~ Reactor ~~Operation~~ Operating

Reactor operation shall mean any condition wherein the reactor is not ~~shut down~~ in the reactor shutdown or reactor secured condition. TS 1.0

~~1.3~~ Operable

A system or component shall be considered operable when it is capable of performing its required function ~~in its normal manner~~. TS 1.0

~~1.4~~ Operating

A component or system is operating if it is performing its required function ~~in its normal manner~~. TS 1.0

~~1.5~~ Experiment

Replace Definition

~~Experiment shall mean any apparatus, device, or material installed in the core or experimental facilities (except for underwater lights, fuel storage racks and the like) which is not a normal part of these facilities.~~ TS 1.0

~~1.6~~ Experimental Facilities

Experimental facilities shall mean Glory Hole, vertical tubes, ~~pneumatic transfer systems~~, central thimble, beam tubes, thermal column, and in-pool irradiation facilities. two triangular exposure locations; TS 1.0

~~1.7~~ Reactor ~~Safety Circuits~~

Reactor ~~safety circuits~~ Protection System (RPS) shall mean those circuts, including their associated TS 1.0

Add New Definitions; See List Next Page

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

New Definitions Added in the Proposed Technical Specifications

Channel	A channel is the combination of sensor, line, amplifier, and output devices which are connected for the purpose of measuring the value of a parameter.
Channel Calibration	A channel calibration is an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include a channel test.
Channel Check	A channel check is a qualitative verification of acceptable performance by observation of channel behavior, or by comparison of the channel with other independent channels or systems measuring the same variable.
Channel Test	A channel test is the introduction of a signal into the channel for verification that it is operable.
Control Rod	A control rod is a device fabricated from neutron absorbing material that is used to establish neutron flux changes and to compensate for routine reactivity losses. A control rod can be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged.
Excess Reactivity	Excess reactivity is that amount of reactivity that would exist if all reactivity control devices were moved to the maximum reactive condition from the point where the reactor is in the reference core condition exactly critical ($k_{eff} = 1$).
Exclusion Area	Exclusion area means that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities, including exclusion or removal of personnel and property from the area.
Explosive Material	Explosive materials are any chemical compound, mixture, or device, the primary or common purpose of which is to function by explosion; or, any device containing a detonating charge that is used for initiating detonation in an explosive, or which can be caused to deflagrate when confined.
Glory Hole	A dry glory hole facility is an aluminum tube of 1.5 in. outside diameter which will fit into any fuel element hole and extends from above the top wooden reactor shield to the lower grid plate. The tube is not filled with water and is used to lower material to be irradiated through the tube into the core region.
Moveable Experiment	A moveable experiment is one where it is intended that all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating
Rated Thermal Power (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 250 kW.

New Definitions Added in the Proposed Technical Specifications

Reactivity (Dollars ($))	Reactivity may be expressed in units of dollars and cents where reactivity in Dollars ($) equals reactivity ($\Delta k/k$) divided by β, the fraction of delayed neutrons, which is equal to 0.0073. Therefore, a reactivity of $1.00 equals 0.73% $\Delta k/k$ and reactivity of 1.00% $\Delta k/k$ equals $1.37.
Reactivity Worth of An Experiment	The reactivity worth of an experiment is the value of the reactivity change that results from the experiment being inserted into or removed from its intended position.
Reactor Operator	A reactor operator is an individual who is licensed to manipulate the controls of a reactor.
Reactor Secured	Reactor secured is the condition with fuel in the reactor and any fixed experiments in place, when: 1. There is insufficient moderator available in the reactor to attain criticality; or; 2. There is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection; or; 3. All of the following conditions exist: (See CTS Definition "Shutdown")
Reactor Shutdown	Reactor shutdown is the condition when the reactor is subcritical by at least 0.73% $\Delta k/k$ ($1.00) when in the reference core condition with the reactivity worth of all installed experiments included and the reactor is not in the reactor secured condition.
Reference Core Condition	The condition of the core when it is critical, at ambient temperature, and the reactivity worth of xenon is negligible (<0.219% $\Delta k/k$ ($0.30)), (i.e. cold, clean, and critical).
Shutdown Margin (SDM)	Shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition and with the most reactive rod in its most reactive position, and that the reactor will remain subcritical without further operator action.

Or provide information for initiation of manual protective action.

Channels

input circuits which are designed to initiate a reactor scram.

Automatic reactor protection

2.0 Reactor Site

2.1 The reactor and associated equipment is located within an exclusion area. TS 5.1.1

2.2 A steel, locked perimeter fence shall surround the ARRR facility, forming an exclusion area. The minimum distance from the center of the reactor pool to the boundary of the exclusion area fencing shall be 50 feet. The restricted area, as defined in 10 CFR 20, shall consist of the entire exclusion area. TS 5.1.1 TS 5.1.2

2.3 The principal activites carried on within the exclusion area shall be those associated with the operation of the ARRR reactor ~~and the use of a hot cell and chemistry laboratory~~. TS 5.1.3 DOC 5.1.3

3.0 Reactor Building

3.1 The reactor shall be housed in a steel building capable of meeting the following functional requirements: TS 5.1.4

3.1. all circulating fans and air conditioning systems except the system which supplies air to the control room shall have the capability to be shut off from a single control in the control room. TS 3.4.2

3.1.2 ventilation shall be achieved by gravity ventilators located on the roof of the building, and TS 5.1.5

3.1.3 a positive air pressure shall be maintained in the control room with respect to the reactor room. TS 3.4.1

3.2 An alarm system shall be installed to detect unauthorized entry into the reactor building. The alarm system shall be monitored constantly and its annunciation shall be tested monthly. TS 3.8.1 4.8.1 5.1.6

4.0 Reactor Pool (Primary System)

4.1 The minimum depth of water above the top of the active core shall be 16 ft. The maximum bulk water temperature shall be 130°F and the minimum 60°F. TS 3.3.1 3.3.2

4.2 The pH and conductivity of the primary coolant shall be measured at least once each month. Corrective action shall be taken to avoid exceeding a pH of 7.5 or a conductivity of 5 umho/cm. TS 3.3.3 4.3.3

initiated within 24 hours of DOC 3.3.3

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

5.0 Reactor Core

5.1 Fuel Elements

5.1.1 The reactor shall contain no more than 90 TRIGA type fuel elements. The core shall be loaded with not more than 3.30 kg of U-235. TS 3.1.5

5.1.2 The maximum excess reactivity above ~~cold, clean critical~~ Reference Core Condition with or without experiments in place, shall ~~be 3 dollars~~, 2.19% ΔK/K TS 3.1.2

5.1.3 The bath temperature coefficient and the prompt fuel temperature coefficient shall be negative at all operating temperatures and ~~the minimum reactivity decrement at full power~~ shall be 80 cents when measured with respect to source power level. 0.584 % ΔK/K TS 5.3.1 (4)(a) TS 3.1.4

5.1.4 The coolant void coefficient shall be negative across the active core. Maximum in-core operating void shall be 10% of the coolant core volume as defined by a cylinder bounded by the grid plates. TS 5.3.1 (4)(b) TS 5.3.1 (4)(c)

5.2 Reflector Elements

5.2.1 The overall reflector elements' dimensions shall be the same as the fuel elements. TS 5.3.1 (3)

5.3 Control Elements

5.3.1 The reactor shall be subcritical by a minimum margin of 0.50 dollar 0.365 % ΔK/K when the maximum worth rod is fully withdrawn from the core. TS 3.1.3 Def: SDM

5.3.2 The maximum rate of reactivity addition for the control rods shall be 11 cents/second. 0.080% ΔK/K There shall be a minimum of three operable control elements. TS 3.2.3 3.2.1

5.3.3 The total time for insertion of the control rods following receipt of a scram signal by the safety system shall be a maximum of 600 milliseconds. TS 3.2.2

6.0 Reactor Safety Systems

6.1 ~~The reactor safety system shall include sensing devices and associated circuits which automatically actuate visual and audible alarms and, when certain pre-set limits are exceeded, scram the reactor. The systems shall be fail-safe(de-energizing shall cause a scram) Table 1 describes the minimum requirements of the safety system.~~

Relocated to USAR 7.0, 7.2, 7.3 & 7.4

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

Table 3.2-1 (RPS)
3.2-2 (RMS)
3.5-1 (Rad Mon)

TS 3.2.6 (RPS)
3.2.7 (RMS)
3.5.1 (Rad Mon)
3.5.2 (Eff Mon)

6.2 The nuclear, process and radiation monitoring instrumentation shall provide the functions and have the set point ranges and associated annunciations listed in Table 2 of these specifications.

6.3 The safety system shall be designed such that no single component failure or circuit fault shall simutaneously disable both the automatic and manual scram circuits. Relocated to USAR 7.0

6.4 Reactor sequence, interlocks and safety circuits shall remain operable while fuel is in the core except that one channel may be removed for maintenance purposes when the reactor is shut down. TS 3.2.5 3.2.5(4)

6.5 Interlocks shall prevent safety rod withdrawal unless all of the following conditions exist: TS 3.2.5(1)

6.5.1 The master switch is in the ON position; TS 3.2.5(1)(a)

6.5.2 The safety system has been reset; except as permitted by Table 3.2-1, footnotes (a) & (b) TS 3.2.5(1)(b)

6.5.3 All four nuclear instruments channels are in the OPERATE mode; TS 3.2.5(1)(c)

6.5.4 The startup channel count rate is greater than 2 cps. 20 CPH TS 3.2.5(1)(d)

It shall not be possible to withdraw more than the safety rod until it has reached the upper limit interlock, at which time either the shim or regulating rod may be moved, but only one at a time. TS 3.2.5(2) 3.2.5(3)

6.6 During a critical experiment a subcritical multiplication plots shall be obtained from at least three instrumentation channels. These channels may be used in addition to the normal operating instrumentation in Table 1. 3.2-1 TS 3.2.4

6.7 Process instrumentation with readout in the control room shall be operating to permit continous indication of pool water temperature and conductivity. Alarms shall be operable to indicate low water flow, low pool water and improper location of the crane bridge. TS 3.2.6 (RPS) TS 3.2.7 (RMS)

7.0 <u>Radiation Monitoring</u>

7.1 A fixed gamma monitor employing Geiger tube detectors shall be located on the wall connecting the control room and the reactor room. This monitor shall serve as both an area radiation monitor and a criticality alarm and will annunciate through an automatic monitoring system to the San Ramon, California, Fire Department and actuate a siren within the reactor building on high radiation level. The monitor shall have a minimum range of 0 to 20 mr/hr. The annunciation and the siren actuation shall be tested monthly. TS 3.5.1 Relocated to USAR 7.4.1 TS 4.5.1

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

Reactor Operating, Reactor Shutdown, Movement of irradiated fuel

7.2 During reactor operation a gas sample shall be continuously withdrawn from the roof vent above the reactor, or from the vicinity of the reactor core, and pumped through a radioactive gas detector chamber. The gas chamber shall be monitored by a beta-gamma detector which shall have a continuous readout in the control room. An annunciator shall indicate when the gas exceeds 2 mr/hr. TS 3.5.2

Relocated to USAR

Table 3.5-1

7.3 A fission product water monitor shall be attached to the process water cleanup system loop adjacent to the demineralizer and shall provide continuous indication in the control room. High radiation levels within the demineralizer or pool water shall annunciate an audible alarm on the reactor console. The range of the monitor shall be from 0.1 to 100 mr/hr. TS 3.2.7

Table 3.2-2

7.4 Portable survey instruments for measuring beta-gamma dose rates in the range of 0.01 mr/hr to 50 r/hr shall be available to the facility. TS 3.5.5(1)

7.5 Portable instruments for measuring fast and thermal neutron dose rates from 0.1 mrem/hr to 1.0 rem/hr shall be available at the facility. TS 3.5.5(2)

7.6 Radiation detector packets containing a series of threshold detectors shall be placed at several locations within the reactor building for post-accident radiation analysis. TS 3.5.4

8.0 Experimental Facilities

8.1 Large-Component Irradiation Box — not be installed — DOC 3.6.10

8.1.1 A large-component irradiation box shall ~~have a maximum volume of 20 cu. feet. The box shall encompass not more than 120° arc of the core and shall be designed so that it can be placed no closer than 5 cm to the outer row of active fuel elements.~~ TS 3.6.10

8.1.2 ~~The platform shall be positioned remotely relative to the reactor core by a positive drive and shall be captive to the stand which is bolted to the floor of the tank. Positive mechanical stops shall prevent moving the experiment box into the active reactor core. CO_2 shall be used for purging and to maintain a slight positive pressure in the box relative to the pool water pressure.~~

8.1.3 ~~To remove or install the experiment box, the platform shall be moved two or more feet away from the reactor core. The box shall then be lowered onto the platform and bolted in place with remote handling equipment. The voided box shall be purged of air prior to exposure to neutrons.~~

8.2 Pneumatic Transfer Facility TS 3.6.11

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

shall not be installed. DOC 3.6.11

8.2.1 ~~A pneumatic transfer facility may be located in any reactor core position. The facility shall be operated with dry CO_2 and exhausted through a filter ventilation system, which is monitored for radioactivity.~~ TS 3.6.11

8.2.2 ~~The in-core portion of the transfer facility shall have a maximum void volume of 34 cu. in. in the active fuel region. A manual control shall be provided which is capable of overriding the automatic timer control.~~

8.3 Glory Hole Facility

8.3.1 A dry glory hole facility may be located in any reactor core position. The glory hole shall accept capsules to a maximum of 1.35 in. in diameter. TS 3.6.7

Relocated to Bases

8.3.2 The glory hole shall be purged with CO_2 to prevent formation of excessive amounts of argon-41. Gas samples shall be taken near the pool when the glory hole facility is operated without a shield plug to insure adequate monitoring of radioactive gases. Ar-41 TS 4.6.7

8.4 Neutron Radiography Facility

8.4.1 The beam tube shall consist of a two-section tapered tube having a rectangular cross section. The upper and lower sections of the tube shall be equipped with a fill and drain line. TS 5.5.1

8.4.2 All components contacting the pool water shall be fabricated from aluminum or stainless steel. TS 5.5.1

8.4.3 The beam catcher shield shall consist of a movable radiation shield. TS 5.5.1

8.5 Thermal Column

8.5.1 The thermal column shall be positioned remotely on steel locating pins immediately adjacent to the reactor core. TS 5.5.2(2)

8.5.2 The thermal column shall be composed of a three-foot cube of graphite encased in aluminum containg five rows of 1.5 in. diameter irradiation holes. The rows shall be placed 6 inches apart and contain seven holes per row. Slotted beams shall be provided to allow experiments to be attached directly to the thermal column. TS 5.5.2

8.6 Vertical Tube

8.6.1 Vertical irradiation tubes, having diameters up to 6 in. may be attached to the thermal column. TS 3.6.8 5.5.2(4)

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

8.6.2 The vertical tube shall be continuously purged with CO_2 to prevent the formation of excess amounts of argon-41. TS 3.6.8(1)

8.7 Other Irradiation Facilities

8.7.1 The central 7 fuel elements of the reactor may be removed from the core and a central irradiation facility installed provided the cross-section area of the facility does not exceed 16 in^2. TS 3.6.9(1) 5.5.3

8.7.2 Two triangular exposure facilities are available which shall allow the insertion of circular experiemnts to a maximum of 2.35 in. diameter or triangular experiments to a maximum of 3.0 in. on a side. TS 3.6.9(2) 5.5.3

8.7.3. Irradiation capsules in the shape of dummy fuel elements shall have a maximum inner void volume of 34 cu. in. in the active fuel region. TS 3.6.9(3)

9.0 Experiment Limitations

9.1 Experiments shall be evaluated in the most reactive condition. TS 4.6.1 6.5.1(4)

9.2 The documentation of experiments, which shall be reviewed and approved prior to insertion in the reactor, shall include at least: TS 4.6.1 6.5.1(1)

9.2.1 The purpose of the experiment;

9.2.2 A description of the experiment; and

9.2.3 An analysis of the possible hazards associated with the performance of the experiment.

9.3 The value of the reactivity worth of any single independent experiment shall not exceed 2 dollars. If such experiments are connected or otherwise related so that their combined reactivity could be added to the core simultaneously, their combined reactivity shall not exceed 2 dollars. 1.46 % ΔK/K TS 3.6.2(1)

9.4 The reactivity worth of any single independent experiment not rigidly fixed in place shall not exceed 1 dollar. If such experiments are connected or otherwise related so that their combined reactivity could be added to the core simultaneously, their combined reactivity worth shall not exceed 1 dollar. 0.73% ΔK/K TS 3.6.2(2)

9.5 No experiment shall be installed in the reactor in such a manner that it could shadow the nuclear instrumentation system monitors. TS 3.6.6(1)

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

9.6 No experiment shall be installed in the reactor in such a manner that a failure could interfere with the insertion of a reactor control element. TS 3.6.6(2)

9.7 No experiment shall be performed involving materials which could: TS 3.6.4

9.7.1 Contaminate the reactor pool causing corrosive action on the reactor components or experiments;

9.7.2 Cause excessive production of airborne radioactivity; or

9.7.3 Produce an uncontained violent chemical reaction.

9.8 Experiments shall not be performed involoving equipment whose failure could result in fuel element damage. TS 3.6.6(3)

9.9 The amount of special nuclear material contained in an experiment shall be limited to 5 grams in the form of solid samples or 3 grams in the form of liquid. Liquid special nuclear materials shall be doubly encapsulated. TS 3.6.3

0.073% ΔK/K

9.10 Experiments having moving parts shall be designed to have reactivity insertion rates less than 10 cents/sec except that moving parts worth less than 5 cents may be oscillated or removed at higher frequencies. TS 3.6.2(3)

0.0365% ΔK/K

9.11 Solid explosive materials may be brought into the facility for the purpose of being radiographed in the neutron radiography facilities located above the pool; provided that the following conditions are met; TS 3.6.5

9.11.1 Individual explosive devices shall be limited to 1000 grains equivalent TNT encased in metallic sheathing. TS 3.8.2(2) 3.8.2(3)

solid or encased DOC 3.8.2

9.11.2 The maximum quantity of explosive material that may be possessed at one time shall be limited to 50 pounds equivalent TNT. TS 3.8.2(5)

9.11.3 Explosive material shall be stored in designated areas within the reactor facility. TS 3.8.2(1)

9.11.3.1 Only the explosive devices to be radiographed within 4 hrs. not to exceed a maximum of ten pounds equivalent TNT, may be removed from the storage area at one time for radiographing, including preparation but excluding packaged shipments. 8 DOC 3.8.2 TS 3.8.2(9)

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

daily

9.11.3.2 An accountability log shall be maintained to show the amount of explosive material in the reactor facility at all times, and shall contain a description of the explosive, and the location within the facility(e.g.,storage, radiographing facility, or shipping dock). TS 3.8.2(10)

9.11.4 The maximum amount of explosive material contained in devices that may be placed in the radiography facilities at a time shall be limited to five pounds equivalent TNT. TS 3.6.5(1)

9.11.4.1 Explosive material in the radiation field at one time shall be limited to 1 pound equivalent TNT. TS 3.6.5(2)

9.11.4.2 Explosive material contained in long device(s) shall be limited to 0.5 pound equivalent TNT per foot. TS 3.6.5(3)

9.12 Personnel handling the explosive devices shall be trained and familiar with the devices being radiographed. TS 3.8.2(7) 3.8.2(8)

9.12.1 Personnel handling the explosive devices shall use special equipment, such as nonsparking tools and shoes, protective clothing, safety shields and grounded benches as required for the explosives being handled.

9.12.2 Unshielded high frequency generating equipment shall not be operated within 50 feet of any explosive device. TS 3.8.2(6)

9.12.3 The explosive device shall be subjected to a total exposure not to exceed 3×10^{11} neutrons/cm^2 and 3×10^3 roentgens of gammas. TS 3.6.5(4)

9.12.4 Explosive devices that, upon ignition, have or provide a thrust in a definite direction shall be positioned so as to be aimed away from the reactor and components. TS 3.6.5(5)

10.0 General Operating Limitations reactor operating or reactor shutdown

10.1 Reactor operation shall be permitted only when two or more personnel are in the reactor building, at least one of whom is a licensed Operator. TS 6.1.3(1)(a) 6.1.3(1)(b)

10.2 The reactor shall not be operated wherever there are significant defects in fuel elements, control rods or control circuitry. TS 3.0.2

10.3 Upon occurrence of abnormal operation of the reactor, including its controls, safety systems and auxiliary systems, action shall be taken immediately to secure the safety of the facility and determine the cause of the abnormal behavior. TS 3.0.1

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

11.0 Fuel Storage and Transfer

11.1 The fuel storage pits located in the floor of the reactor room shall accommodate a maximum of 19 fuel elements(700 gm U-235)in storage racks dry or flooded with water. The fuel storage pits shall be secured with a lock and chain except during fuel transfer operations. TS 3.7.3 5.4.1

11.2 Additional fuel storage racks may be located in the reactor tank. Each of these storage facilities shall be so designed that for all conditions of moderation k_{eff} shall not exceed a value of 0.8 0.9 TS 3.7.2 DOC 3.7.2
light water

11.3 A fuel handling tool ~~shall be used~~ in transferring fuel elements of low radioactivity between the storage pits and the reactor; a shielded fuel transfer cask shall be used for the transfer of highly radioactive fuel elements. The fuel handling tool(s) shall remain ~~in a locked cabinet~~ under the cognizance of the Reactor Supervisor when not authorized for use. secured with a lock TS 5.4.4 TS 3.7.1

11.4 All fuel transfers in the reactor tank shall be conducted by a minimum staff of three men, and shall include a licensed Senior Operator and a licensed Operator. The staff members shall monitor the operation using appropriate radiation monitoring instrumentation. Fuel transfers outside the reactor tank but within the facility shall be supervised by a licensed Operator. TS 3.7.6 TS 3.7.7

11.5 Not more than one fuel element shall be allowed in the facility which is not in storage or in the core lattice. TS 3.7.4

12.0 Administrative Requirements

12.1 Organization

12.1.1 The Reactor Supervisor shall have responsibility of the reactor facility. In all matters pertaining to reactor operations and to these Technical Specifications, the Reactor Supervisor shall be responsible to the President, Aerotest Operations, Inc., ~~a wholly-owned subsidiary of Explosive Technology, Inc. The President, Aerotest Operations, Inc. shall report to the Board of Directors of Aerotest Operations, Inc. which includes the Presidents of both OEA, Inc. and Explosive Technology, Inc.~~ TS 6.2.1 (6)(a)(i) 6.1.1 6.1.2 Figure 6-1 DOC 6.2.1

12.1.2 The Radiological Safety Officer shall review and approve all procedures and experiments involving radiological safety. He shall enforce rules, regulations and procedures relating to radiological safety, conduct routine radiation surveys and is responsible to the Manager, Aerotest Operations. TS 6.2.1 (6)(b) 6.3.2
General

12.1.3 The Reactor Safeguards Committee shall be composed of not less than five members, of whom no more than three are members of the operating organization. The committee TS 6.2.1 (1)

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

shall meet on call of the chairman and they shall meet at least annually. The committee shall be responsible for, but not limited to the following: TS 6.2.2(i)

~~12.1.3.1 Reviewing and approving nuclear safety standards associated with the use of the facility;~~ DOC 6.2.3

~~12.1.3.2 Reviewing and approving all proposed experiments and procedures and changes thereto, and modifications to the reactor and its associated components.~~

~~12.1.3.3 Determining whether proposed experiments, procedures or modifications involve unreviewed safety questions, as defined in 10 CFR 50, Part 50.59(c), and are in accordance with these Technical Specifications;~~

~~12.1.3.4 Conducting periodic audits of procedures, reactor operations and maintenance, equipment performance, and records;~~

~~12.1.3.5 Reviewing all reported abnormal occurrences and violations of these Technical Specifications, evaluating the cause of such events and the corrective action taken and recommending measures to prevent reoccurrence and;~~

~~12.1.3.6 Reporting their findings and recommendations concerning the above to the Manager, Aerotest Operations~~

12.1.4 The Reactor Supervisor shall have a Bachelor's degree in Engineering or Physical Science and shall have a minimum of 4 years experience in the operation of a nuclear facility during which he shall have demonstrated competence in supervision and reactor operations. He shall hold a Senior Reactor Operator License for the facility. TS 6.2.1(c)(a) (iii) (iv) (v)

12.1.5 The Radiological Safety Officer shall have a Bachelor's degree in Biological or Physical Science and shall have a minimum of 2 years experience in personnel and environmental radiation monitoring programs at a nuclear facility. Certification as a Health Physicist by the Health Physics Society is acceptable in lieu of the education and experience requirements given above. TS 6.2.1(c)(b)(v) (vi) (vii)

12.2 Procedures

12.2.1 Detailed written procedures shall be provided and followed for the following reactor operations: TS 6.4.1

12.2.1.1. Normal startup, operation and shutdown of ~~the complete facility and of all systems and components involving nuclear safety of the facility~~ the reactor TS 6.4.1(i)(a)

12.2.1.2. Refueling operations loading, unloading, movement of fuel TS 6.4.1(i)(b)

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

12.2.1.3 ~~Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms, suspected primary system leaks and abnormal reactivity changes.~~

12.2.1.4 ~~Paragraph omitted.~~

12.2.1.5 Preventive or corrective maintenance operations which could have an effect on the safety of the reactor. TS 6.4.1(1)(d)

12.2.2 Temporary procedures which do not change the intent of previously approved procedures may be utilized on approval by a licensed Senior Reactor Operator and one other qualified individual. Such procedures shall be subsequently reviewed by the Reactor Safeguards Committee. TS 6.4.3(1)

12.3 <u>Records</u>

In addition to those records required under the facility license and applicable regulations, the following records shall be kept for four years when explosive materials are to be irradiated or radiographed: TS 6.8.5(2)

12.3.1 The type and quantity of material irradiated.

12.3.2 Date, time of day, and length of exposure.

12.3.3 Total neutron and gamma exposure level.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

Relocated to USAR 7.2

TABLE 1

NUCLEAR INSTRUMENTATION

Channel (No.)	Detector	Minimum Sensitivity	Information	Minimum Range	Information to Logic Element (Scram)
Startup (1)	BF_3 Proportional Counter	4.5 counts/sec per n/cm^2-sec	Neutron flux, period	Source level to 1 watt	Period scram;[a] low count rate scram
Log N (2)	Compensated ion chamber	4×10^{-14} amp/n/cm^2-sec	Power level, period	10^{-2} watts to 120% full power	Period scram
Linear Level Safety (3)	Uncompensated ion chamber	4.4×10^{-14} amp/n/cm^2-sec	Power level	30 watts to 120% full power	High and low level[b] scrams
Linear Level Safety (4)	Compensated ion chamber	4.4×10^{-14} amp/n/cm^2-sec	Power level	10^{-1} watts to 120% full power	High and low level scram

(a) Scrams on Channel 1 are by-passed when signal on Channel 2 exceeds a fixed setting similarly the high voltage is removed from the detector and the detector is shorted.

(b) Low level scram is bypassed on Channel 3 and 4 when Channel 2 is below a fixed setting.

Table 3.2-1, footnote (b)
footnote (a)

3.2-1 (RPS)
3.2:2 (RMS)
3.5-1 (Rad Monitoring)

TABLE 2

SAFETY SYSTEM FUNCTIONS

Sensor or Trip Device	No. of Switches or Sensors	Annunciator and Scram Set Point	Annunciator and Alarm Set Point
Short Period; Chs. 1, 2	2	≥ 3 sec.	≥ 3 sec
High Neutron Flux Level; Chs. 3, 4	2	≤ 98% of full scale and not greater than 120% full power RTP	
High Temperature of Coolant Water	1	≤ 130°F	≤ 130 °F
Low Pool Water Level	1		≥ 16 feet above core ≤ 1 ~~ft max decrease~~
Seismic Disturbance	1	IV on modified Mercalli Scale max.	
Bridge Crane Location	1	NA	When located off storage position
Low Neutron Detector Voltage; Chs. 2, 3, 4	3	≥ 500 volts	≥ 500 volts
Low Source Level; Ch. 1	1	≥ 2 cps 120 cpm	
Loss of Instrument Power; Ch. 2	1	x	X
Low Neutron Flux; Ch. 3 & 4	2	≥ 5% of full scale	≥ 5% of full scale
Area Radiation Monitor	1	NA	≤ 10 mr/hr
Water Radioactivity	1	NA	≤ 20 mr/hr
Demineralizer Water Flow	1	NA	≥ 4 gpm
Building Gas Effluent Monitor	1	NA	≤ 2 mr/hr
Master Key Switch	1	Not on "ON" position	OFF
Manual Scram Button Bar	1	Button Bar Depressed	Bar Depressed
Primary Coolant Conductivity	1	NA	Indication Only

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

AEROTEST OPERATIONS, INC.

Aerotest Radiography and Research Reactor (ARRR)

DESCRIPTION OF PROPOSED CHANGES TO THE ARRR TECHNICAL SPECIFICATIONS

ATTACHMENT 2

Attachment 2 is a copy of ARRR proposed Technical Specifications (TS) that has been annotated to provide a detailed cross reference between the proposed TS and the CTS.

AEROTEST OPERATIONS, INC.

Aerotest Radiography and Research Reactor (ARRR)

APPENDIX A:

TECHNICAL SPECIFICATIONS

Amendment 9 [Proposed R0]

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

TABLE OF CONTENTS

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

1.0 DEFINITIONS

DOC 1.0

Channel	A channel is the combination of sensor, line, amplifier, and output devices which are connected for the purpose of measuring the value of a parameter.
Channel Calibration	A channel calibration is an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include a channel test.
Channel Check	A channel check is a qualitative verification of acceptable performance by observation of channel behavior, or by comparison of the channel with other independent channels or systems measuring the same variable.
Channel Test	A channel test is the introduction of a signal into the channel for verification that it is operable.
Control Rod	A control rod is a device fabricated from neutron absorbing material that is used to establish neutron flux changes and to compensate for routine reactivity losses. A control rod can be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged.
Excess Reactivity	Excess reactivity is that amount of reactivity that would exist if all reactivity control devices were moved to the maximum reactive condition from the point where the reactor is in the reference core condition exactly critical (k_{eff} =1).
Exclusion Area	Exclusion area means that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities, including exclusion or removal of personnel and property from the area.
Experiment	Any operation, hardware, or target (excluding devices such as detectors, foils, etc.), that is designed to investigate non-routine reactor characteristics or that is intended for irradiation within the pool, on or in a beamport or irradiation facility, and is not rigidly secured to a core or shield structure so as to be a part of their design. CTS 1.5
Experimental Facilities	Experimental facilities mean the glory hole, vertical tubes, central thimble, beam tubes, thermal column, inpool irradiation facilities, and the two triangular exposure locations. CTS 1.6

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

Explosive Material	Explosive materials are any chemical compound, mixture, or device, the primary or common purpose of which is to function by explosion; or, any device containing a detonating charge that is used for initiating detonation in an explosive, or which can be caused to deflagrate when confined.	DOC 1.0
Glory Hole	A dry glory hole facility is an aluminum tube of 1.5 in. outside diameter which will fit into any fuel element hole and extends from above the top wooden reactor shield to the lower grid plate. The tube is not filled with water and is used to lower material to be irradiated through the tube into the core region.	
Moveable Experiment	A moveable experiment is one where it is intended that all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating	
Operable	Operable means a component or system is capable of performing its required function.	CTS 1.3
Operating	Operating means a component or system is performing its required function.	CTS 1.4
Rated Thermal Power (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 250 kW.	Lic Condition 2.C(1)
Reactivity (Dollars ($))	Reactivity may be expressed in units of dollars and cents where reactivity in Dollars ($) equals reactivity ($\Delta k/k$) divided by β, the fraction of delayed neutrons, which is equal to 0.0073. Therefore, a reactivity of $1.00 equals 0.73% $\Delta k/k$ and reactivity of 1.00% $\Delta k/k$ equals $1.37.	DOC 1.0
Reactivity Worth of An Experiment	The reactivity worth of an experiment is the value of the reactivity change that results from the experiment being inserted into or removed from its intended position.	
Reactor Operating	Reactor operating is any condition with fuel in the reactor when the reactor is not in the reactor secured condition or the reactor shutdown condition.	CTS 1.2
Reactor Operator	A reactor operator is an individual who is licensed to manipulate the controls of a reactor.	DOC 1.0
Reactor Protection System	Reactor protection systems are those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.	CTS 1.7

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

Reactor Secured	Reactor secured is the condition with fuel in the reactor and any fixed experiments in place, when: 1. There is insufficient moderator available in the reactor to attain criticality; or; 2. There is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection; or; 3. All of the following conditions exist: a. The console key switch is in the off position and the key is removed from the console and under the control of a licensed operator or stored in a locked storage area; and, b. Sufficient control rods are inserted so as to assure the reactor is subcritical by a margin greater than 0.73% $\Delta k/k$ (\$1.00) when in reference core condition; and, c. No work is in progress involving refueling operations or maintenance of control rod mechanisms.	DOC 1.0 DOC 1.0 CTS 1.1
Reactor Shutdown	Reactor shutdown is the condition when the reactor is subcritical by at least 0.73% $\Delta k/k$ (\$1.00) when in the reference core condition with the reactivity worth of all installed experiments included and the reactor is not in the reactor secured condition.	DOC 1.0
Reference Core Condition	The condition of the core when it is critical, at ambient temperature, and the reactivity worth of xenon is negligible (<0.219% $\Delta k/k$ (\$0.30)), (i.e. cold, clean, and critical).	DOC 1.0
Shutdown Margin (SDM)	Shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition and with the most reactive rod in its most reactive position, and that the reactor will remain subcritical without further operator action.	DOC.1.0

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

2.0 SAFETY LIMIT (SL)

2.1 MAXIMUM FUEL ELEMENT TEMPERATURE

The temperature in any fuel element in the ARRR TRIGA reactor shall not exceed 500 °C under any condition of operation. DOC 2.1

Technical Specification 6.6.1 lists the actions that shall be taken following a safety limit violation.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

3.0 LIMITING CONDITIONS FOR OPERATION

3.0 GENERAL REQUIREMENTS

3.0.1 Abnormal Operation:

Upon occurrence of abnormal operation of the reactor, including its controls, safety systems and auxiliary systems, action shall be taken immediately to secure the safety of the facility and determine the cause of the abnormal behavior. CTS 10.3

3.0.2 Defects in Fuel Elements, Control Rods, or Control Circuitry:

The reactor shall not be operated wherever there are significant defects in fuel elements, control rods, or control circuitry. CTS 10.2

3.0.3 Actions for LCO Not Met:

LCOs must be met whenever the reactor is in the condition specified in the associated applicability statement. If an LCO is not met, the following actions shall be taken: DOC 3.0.3

(1) The failure to meet an LCO shall be reported to the Reactor Supervisor or designated alternate immediately.

(2) If the LCO includes actions required when the LCO is not met, the LCO shall be considered met if the actions are completed within the specified completion time.

(3) If the LCO does not include actions required when the LCO is not met, action shall be initiated to place the reactor and facility in a condition where the LCO is no longer applicable or the reactor is in the reactor secured condition.

(4) If the LCO is applicable at all times, the Reactor Supervisor or designated alternate shall identify any additional compensatory actions required to place the reactor and facility in a safe condition. Actions to meet the LCO shall continue until the LCO is met.

(5) If the reactor is placed in the reactor shutdown condition, the reactor shall not be placed in the reactor operating condition until authorized by the Reactor Supervisor or designated alternate.

(6) The Reactor Supervisor or designated alternate shall make a determination if the event is reportable in accordance with Technical Specification 6.7.3 and, if necessary, initiate requirements of Technical Specification 6.6.2 for a reportable event.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

3.1 REACTOR CORE PARAMETERS

3.1.1 Rated Thermal Power (RTP):

Reactor thermal power shall not exceed 250 kW as measured by the calibrated power channels. Lic Cond 2.C(1)

Applicability: At all times.

3.1.2 Excess Reactivity:

The maximum excess reactivity with the reactor in reference core condition, with and without experiments in place, shall be 2.19% $\Delta k/k$ ($3.00). CTS 5.1.2

Applicability: At all times.

3.1.3 Shutdown Margin (SDM):

SDM, with and without experiments in place, shall be $\geq$ 0.365% $\Delta k/k$ ($\geq$ $0.50). CTS 5.3.1

Applicability: At all times.

3.1.4 Reactivity Coefficients:

Reactivity coefficients shall be maintained such that the reactivity decrement at full power is $\geq$ 0.584% $\Delta k/k$ ($\geq$ $0.80) when measured with respect to source range power level. CTS 5.1.3

Applicability: Whenever the reactor is in the reactor operating condition or reactor shutdown condition.

3.1.5 Core Configuration:

Core configuration shall be maintained within the following limits: CTS 5.1.1

(1) The reactor core lattice shall contain $\leq$ 90 cylindrical TRIGA type fuel elements.

(2) The reactor core lattice shall be loaded with $\leq$ 3.30 kg of U-235.

Applicability: At all times.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

3.2 REACTOR CONTROL AND SAFETY SYSTEMS

3.2.1 Control Rods:

Three control rods (1 safety rod, 1 shim rod and 1 regulating rod) shall be operable. CTS 5.3.2

Applicability: Whenever the reactor is in the reactor operating condition.

3.2.2 Scram Time:

The total time for complete insertion of the control rods shall be ≤ 600 milliseconds following receipt of a scram signal by the safety system channels listed in Table 3.2-1. CTS 5.3.3

Applicability: Whenever the reactor is in the reactor operating condition.

3.2.3 Reactivity Insertion Rate:

The maximum rate of reactivity addition by the control rods shall be +0.080% Δk/k/second (+$0.11/second). CTS 5.3.2

Applicability: Whenever the reactor is in the reactor operating condition.

3.2.4 Reactivity Insertion Monitoring: CTS 6.6

Subcritical multiplication levels shall be plotted using input from a minimum of three instrumentation channels. Nuclear instrumentation channels listed in Table 3.2-1 may be used to satisfy requirements for one or more channels.

Applicability: Prior to criticality during experiments with the potential to affect core reactivity.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

3.2.5 Reactor Sequence and Interlocks: CTS 6.4 6.5

(1) Interlocks shall prevent withdrawal of the safety rod unless all of the following conditions exist:

(a) *The master switch is in the ON position.* CTS 6.5.1

(b) The safety system has been reset except for those components bypassed by the Channel 1 disconnect in accordance with Table 3.2-1 footnotes (a) and (b). CTS 6.5.2

(c) All four nuclear instrument channels are in the operate mode. CTS 6.5.3

(d) The startup channel count rate is > 120 counts per minute unless Channel 1 is bypassed in accordance with Table 3.2-1 footnote (a). CTS 6.5.4

(2) Interlocks shall prevent withdraw of the shim rod and the regulating rod unless the safety rod is withdrawn to its upper limit. CTS 6.5

(3) Interlocks shall prevent simultaneous withdraw of the shim rod and the regulating rod. CTS 6.5

(4) Any one interlock in LCO 3.2.5 may be removed from service for maintenance when the reactor is in the reactor shutdown condition. CTS 6.4

Applicability: Whenever the reactor is in the reactor operating condition or reactor shutdown condition.

3.2.6 Reactor Protection System Instrumentation:

(1) Each Reactor trip system, alarm, annunciator, and associated monitoring instrument channel listed in Table 3.2-1 shall be operable. CTS 6.2, Table 2

(2) The minimum number of operable channels, limiting safety system settings, and alarm and annunciator set points shall be within the limits specified in Table 3.2-1.

Applicability: Whenever the reactor is in the reactor operating condition.

3.2.7 Reactor Monitoring System Instrumentation:

(1) Each Reactor monitoring system, alarm, annunciator, and associated monitoring instrument channel listed in Table 3.2-2 shall be operable. CTS 6.2, Table 2

(2) The minimum number of operable channels and alarm and annunciator set points shall be within the limits specified in Table 3.2-2.

Applicability: Whenever the reactor is in the reactor operating condition.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

CTS Table 2

Table 3.2-1
Reactor Protection System Instrumentation

Channel	Minimum No. of Required Channels (c)(d)	Required Function(s)	Limiting Safety System Setting	Annunciator and Alarm Set Point
Neutron Flux, Channel 1	1(a)	Reactor Trip: Short Period Low Source Level	 ≥ 3 seconds ≥ 120 cpm	 ≥ 3 seconds ≥ 120 cpm
Neutron Flux, Channel 2	1	Reactor Trip: Short Period Loss of Inst Power Low Detector Voltage	 ≥ 3 seconds X ≥ 500 volts	 ≥ 3 seconds X ≥ 500 volts
Neutron Flux, Channel 3	1(b)	Reactor Trip: High Neutron Flux Low Neutron Flux Low Detector Voltage	 ≤ 98% of full scale and ≤120% RTP ≥ 5% of full scale ≥ 500 volts	 ≤ 98% of full scale and ≤120% RTP ≥ 5% of full scale ≥ 500 volts
Neutron Flux, Channel 4	1(b)	Reactor Trip: High Neutron Flux Low Neutron Flux Low Detector Voltage	 ≤ 98% of full scale and ≤120% RTP ≥ 5% of full scale ≥ 500 volts	 ≤ 98% of full scale and ≤120% RTP ≥ 5% of full scale ≥ 500 volts
Reactor Tank Water Level Low	1	Reactor Trip	≥ 16 feet above top of core	≥ 16 feet above top of core
Seismic Disturbance	1	Reactor Trip	IV on Modified Mercalli Scale max	IV on Modified Mercalli Scale max
Reactor Tank Water Temp High	1	Reactor Trip	≤ 130 °F	≤ 130 °F
Manual Scram Bar	1	Reactor Trip	Bar depressed	N/A
Master Key Switch	1	Reactor Trip	"OFF" position	N/A

(a) Channel 1 scrams are bypassed when Channel 2 exceeds a fixed setting of approximately 1×10^{-10} amps. When bypassed, Channel 1 detector is shorted and high voltage is removed. CTS Table 1

(b) Low level scrams are bypassed on Channels 3 and 4 when Channel 2 is below a fixed setting of approximately 1×10^{-10} amps.

(c) When any one required channel is inoperable, the reactor shall be placed in the reactor secured condition within 8 hours. DOC 3.2.c

(d) When more than one required channel is inoperable, the reactor shall be placed in the reactor secured condition within 1 hour. 3.2.c

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

Table 3.2-2
Reactor Monitoring System Instrumentation

CTS Table 2

Channel	Minimum No. of Required Channels	Required Function(s)	Annunciator and Alarm Set Point
Bridge Crane Location	1[a]	Annunciator/Alarm	When located off storage position
Primary Coolant Conductivity	1[b]	Control Room Indication	Indication only
Reactor Demineralizer Water Flow	1[c]	Control Room Annunciator/Alarm	≥ 4 gpm
Reactor Water Radioactivity	1[a]	Control Room Indication/ Annunciator/Alarm	≤ 20 mr/hr

(a) When a Bridge Crane Location or Reactor Water Radioactivity channel is inoperable, an alternate method of monitoring the parameter shall be established within 24 hours. The alternate monitoring method may be substituted for the required channel for a maximum of 30 days from the time the channel is inoperable. Otherwise, the reactor shall be placed in the reactor secured condition within following 4 hours. DOC 3.2.7

(b) When a Primary Coolant Conductivity channel is inoperable, primary coolant conductivity shall be verified using an alternate method within 7 days and every 7 days thereafter. The alternate method may be substituted for the required channel for a maximum of 30 days from the time the channel is inoperable. Otherwise, the reactor shall be placed in the reactor secured condition within following 4 hours. DOC 3.2.7

(c) When a Reactor Demineralizer Water Flow channel is inoperable, reactor demineralizer water flow shall be verified using an alternate method immediately and every 24 hours thereafter. The alternate method may be substituted for the required channel for a maximum of 30 days from the time the channel is inoperable. Otherwise, the reactor shall be placed in the reactor secured condition within following 4 hours. DOC 3.2.7

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

3.3 COOLANT SYSTEMS

3.3.1 Reactor Tank Water Level:

The depth of water above the top of the active core shall be ≥ 16 feet. CTS 4.1

Applicability: Whenever the reactor is in the reactor operating condition or reactor shutdown condition.

3.3.2 Primary Coolant Water Temperature:

The primary coolant bulk water temperature shall be ≥ 60 °F and ≤ 130 °F CTS 4.1

Applicability: Whenever the reactor is in the reactor operating condition or reactor shutdown condition.

3.3.3 Primary Coolant Water Quality:

Primary Coolant Water Quality shall be maintained within the following limits: CTS 4.2

(1) pH shall be ≤ 7.5; and,

(2) Conductivity shall be ≤ 5 μmho/cm.

Applicability: At all times.

If LCO 3.3.3 is not met, action shall be initiated within 24 hours to restore primary coolant water pH and conductivity to within the specified limits.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

3.4 VENTILATION SYSTEMS

3.4.1 Control Room:

The control room shall be maintained at a positive pressure with respect to the reactor room. CTS 3.1.3

Applicability: Whenever the reactor is in the reactor operating condition or reactor shutdown condition and during movement of irradiated fuel.

3.4.2 Reactor Building Circulation Fans:

All air conditioning systems and building circulating fans in or normally open to the high bay area shall have the capability to be shut off from a single control in the control room. CTS 3.1.1

Applicability: Whenever the reactor is in the reactor operating condition or reactor shutdown condition and during movement of irradiated fuel.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

3.5 CRITICALITY ALARM, RADIATION AND RADIOACTIVE EFFLUENT MONITORING

3.5.1 Criticality Alarm and Area Radiation Monitoring:

(1) A fixed gamma monitor employing Geiger tube detectors shall be located on the wall connecting the control room and the reactor room within the limits specified in Table 3.5-1 for the Criticality Alarm and Area Radiation Monitor. CTS 7.1

(a) This monitor shall annunciate through an automatic monitoring system; and,

(b) This monitor shall actuate a siren within the reactor building on high radiation level.

(2) An appropriate number of radiation sensitive badges shall be placed at strategic locations within the reactor building to obtain a valid representative sample for radiation analysis. DOC 3.5.1(2)

Applicability: At all times.

3.5.2 Gaseous Effluent Monitoring:

A gas sample shall be continuously withdrawn from the roof vent above the reactor, or from the vicinity of the reactor pool, and pumped through a radioactive gas detector chamber within the limits specified in Table 3.5-1 for the Building Gaseous Effluent Monitor. CTS 7.2 Table 2

Applicability: Whenever the reactor is in the reactor operating condition or reactor shutdown condition and during movement of irradiated fuel.

3.5.3 Particulate Effluent Monitoring:

A particulate sample shall be continuously withdrawn from the reactor room and collected on filter paper. DOC 3.5.3

Applicability: Whenever the reactor is in the reactor operating condition or reactor shutdown condition and during movement of irradiated fuel.

3.5.4 Post Accident Radiation Monitoring:

Two radiation detector packets containing a series of threshold detectors shall be placed at strategic locations within the reactor building for post-accident radiation analysis. CTS 7.6

Applicability: At all times.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

3.5.5 Portable Radiation Monitoring Instruments:

The following portable radiation monitoring instruments shall be operable and onsite as follows:

(1) A portable survey instrument for measuring beta-gamma dose rates in the range of 0.01 mr/hr to 50 r/hr. CTS 7.4

(2) A portable instrument for measuring fast and thermal neutron dose rates from 0.1 mr/hr to 1.0 r/hr. CTS 7.5

Applicability: At all times.

3.5.6 Radioactive Effluent Limits:

Normal releases of radioactive effluents from reactor operation shall not exceed 10 CFR 20 limits. DOC 3.5.6

Applicability: At all times.

Table 3.5-1
Criticality Alarm, and Area and Effluent Radiation Monitoring Instrumentation

CTS Table 2

Channel	Minimum No. of Required Channels	Required Function(s)	Annunciator and Alarm Set Point
Criticality Alarm and Area Radiation Monitor	1(a)(b)	Annunciator/ Alarm for LCO 3.5.1	≤ 10 mr/hr
Building Gaseous Effluent Monitor	1(a)(b)	Annunciator/ Alarm for LCO 3.5.2	≤ 2 mr/hr

(a) When the Criticality Alarm and Area Radiation Monitor or Building Gaseous Effluent Monitor is inoperable, alternate methods of monitoring the parameter shall be established within 8 hours. The alternate monitoring method may be substituted for the required channel for maximum of 7 days from the time the channel is inoperable. Otherwise, the reactor shall be placed in the reactor secured condition within following 4 hours. DOC 3.5.1

(b) When the Criticality Alarm and Area Radiation Monitor or Building Gaseous Effluent Monitor is inoperable during movement of a fuel element, movement of the fuel element shall be stopped immediately. This action shall not preclude completion of movement of a fuel element to a safe position. DOC 3.5.1

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

3.6 EXPERIMENTS

3.6.1 Evaluation and Approval of Experiments:

Experiments shall be evaluated and approved in accordance with the requirements of Technical Specification 6.5.1. CTS 9.2

Applicability: Prior to insertion of the experiment into the reactor.

3.6.2 Reactivity Limits during Experiments:

Evaluations of experiments performed in accordance with the requirements of Technical Specification 6.5.1 shall demonstrate that the following reactivity limits will be maintained during the experiment when in its most reactive condition:

(1) The reactivity worth of any single independent experiment shall be ≤ 1.46% Δk/k (≤ $2.00). If such experiments are connected or otherwise related so that their combined reactivity could be added to the core simultaneously, their combined reactivity shall be ≤ 1.46% Δk/k ($2.00). CTS 9.3

(2) The reactivity worth of any single independent experiment not rigidly fixed in place shall be ≤ 0.73% Δk/k (≤ $1.00). If such experiments are connected or otherwise related so that their combined reactivity could be added to the core simultaneously, their combined reactivity worth shall be ≤ 0.73% Δk/k (≤ $1.00). CTS 9.4

(3) Experiments having moving parts shall be designed to have reactivity insertion rates < 0.073% Δk/k /second (< $0.10/second) except that moving parts worth < 0.0365% Δk/k /second (< $0.05/second) may be oscillated or removed at higher frequencies. CTS 9.10

(4) Limits in LCO 3.1.2, Excess Reactivity, and LCO 3.1.3, Shutdown Margin, shall not be exceeded including the reactivity worth of the experiment when evaluated in the most reactive condition. CTS 5.1.2
5.3.1

Applicability: Prior to initiation of the experiment.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

3.6.3 Special Nuclear Material (SNM) included in Experiments: CTS 9.9

SNM introduced into the reactor for experiments shall be limited as follows:

(1) The amount of special nuclear material contained in an experiment shall be limited to either of the following:

(a) 5 grams of SNM in the form of solid samples; or,

(b) 3 grams of SNM in the form of liquid.

(2) Liquid special nuclear materials shall be doubly encapsulated.

Applicability: At all times.

3.6.4 Materials Used during Experiments: CTS 9.7

No experiment shall be performed involving materials which could:

(1) Contaminate the reactor pool causing corrosive action on the reactor components or experiments;

(2) Cause excessive production of airborne radioactivity; or,

(3) Produce an uncontained violent chemical reaction.

Applicability: At all times.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

3.6.5 Explosive Materials in Experiments:

Solid explosive materials may be brought into the facility for the purpose of being radiographed in the neutron radiography facilities located above the pool provided that the following conditions are met:

(1) The maximum amount of explosive material contained in devices that may be placed in the radiography facilities at a time shall be limited to five pounds equivalent TNT. CTS 9.11.4

(2) Explosive material in the radiation field at one time shall be limited to 1 pound equivalent TNT. CTS 9.11.4.1

(3) Explosive material contained in long devices shall be limited to 0.5 pound equivalent TNT per foot. CTS 9.11.4.2

(4) The explosive devices shall be subjected to a total exposure not to exceed 3×10^{11} neutrons/cm^2 and 3×10^3 roentgens of gammas. CTS 9.12.3

(5) Explosive devices that have or provide a thrust in a definite direction upon ignition shall be positioned so as to be aimed away from the reactor and components. CTS 9.12.4

Applicability: At all times.

3.6.6 *Configuration and Potential Failure Mechanisms:*

(1) No experiment shall be installed in the reactor in such a manner that it could shadow the nuclear instrumentation detectors. CTS 9.5

(2) No experiment shall be installed in the reactor in such a manner that a failure could interfere with the insertion of a reactor control rod. CTS 9.6

(3) Experiments shall not be performed involving equipment whose failure could result in fuel element damage. CTS 9.8

Applicability: At all times.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

3.6.7 Glory Hole Facility:

One dry glory hole facility may be located in any reactor core position with the following restrictions:

(1) The glory hole shall accept capsules with a maximum diameter of 1.35 inches. CTS 8.3.1

(2) The glory hole shall be purged with CO_2 to prevent the formation of excessive amounts of Argon-41 as follows: CTS 8.3.2

(a) A glory hole facility operated with a shield plug shall be purged prior to each insertion of the shield plug. DOC 8.3.2

(b) A glory hole facility operated without a shield plug shall be purged with CO_2 continuously. CTS 8.3.2

Applicability: Whenever a glory hole facility is located in any reactor core position and the reactor is in the reactor operating condition.

3.6.8 Vertical Tubes:

Vertical irradiation tubes with diameters up to 6 inches may be attached to the thermal column with the following restrictions:

(1) Each vertical tube shall be purged with CO_2 continuously. CTS 8.6.1

(2) Gas samples shall be taken near the pool continuously and Argon-41 presence monitored when the vertical tube is inserted. CTS 8.6.2

Applicability: Whenever a Vertical Tubes is attached to the thermal column and the reactor is in the reactor operating condition.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

3.6.9 Other Irradiation Facilities:

The following Irradiation Facilities may be used within the limitations specified:

(1) The central 7 fuel elements of the reactor may be removed from the core and a central irradiation facility installed provided the cross-sectional area of the facility does not exceed 16 square inches. CTS 8.7.1

(2) Two triangular exposure facilities are available which shall allow the insertion of circular experiments to a maximum of 2.35 inches diameter or triangular experiments to a maximum of 3.0 inches on a side. CTS 8.7.2

(3) Irradiation capsules in the shape of dummy fuel elements shall have a maximum inner void volume of 34 cubic inches in the active fuel region. CTS 8.7.3

Applicability: At all times.

3.6.10 Large Component Irradiation Box:

The large component irradiation box shall not be installed in the reactor tank. CTS 8.1

Applicability: At all times.

3.6.11 Pneumatic Transfer Facility:

A pneumatic transfer facility shall not be installed in any reactor core position. CTS 8.2

Applicability: At all times.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

3.7 FUEL STORAGE AND TRANSFER

3.7.1 Fuel Handling Tools:

The fuel handling tools shall be secured with a locking mechanism under the cognizance of the Reactor Supervisor when not authorized for use. CTS 11.3

Applicability: At all times except when the fuel handling tools are in use.

3.7.2 Fuel Storage in the Reactor Tank:

Fuel may be stored in fuel storage racks located in the reactor tank within the following limitation: CTS 11.2

(1) Fuel in the reactor tank but not part of the reactor core lattice shall be stored in a geometric array where $k_{eff} \leq 0.9$ for all conditions of moderation and reflection using light water. DOC 3.7.2

Applicability: At all times.

3.7.3 Fuel Storage in the Fuel Storage Pits:

Fuel may be stored in the fuel storage pits within the following limitations: CTS 11.1

(1) Each fuel storage pit shall hold ≤19 fuel elements and ≤ 700 grams of U-235.

(2) Each fuel storage pit containing a fuel element shall be secured with a lock and chain except during fuel transfer operations involving the fuel element being stored in the associated pit.

Applicability: At all times fuel is stored in any fuel storage pit.

3.7.4 Fuel not in the Reactor and not in Storage:

Not more than one fuel element shall be allowed in the facility which is not in storage or in the core lattice. CTS 11.5

Applicability: At all times.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

3.7.5 Fuel in Shipping Containers:

(1) Fuel may be contained in an approved fuel shipping container within the limitations of LCO 3.7.4. CTS 11.3

(2) When an approved fuel shipping container is used, then the licensed limitations on k_{eff} for the container shall apply. DOC 3.7.5

Applicability: Whenever a fuel shipping container is in use.

3.7.6 Fuel Transfer in the Reactor Tank:

All fuel transfers in the reactor tank shall be conducted by a minimum staff of three members, and shall include a licensed Senior Operator and a licensed Operator. The staff members shall monitor the operation using appropriate radiation monitoring instrumentation. CTS 11.4

Applicability: During fuel transfers in the reactor tank.

3.7.7 Fuel Transfer Outside the Reactor Tank:

Fuel transfers outside the reactor tank but within the facility shall be supervised by a licensed Operator. CTS 11.4

Applicability: During transfers of fuel outside the reactor tank.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

3.8 REACTOR FACILITY

3.8.1 Reactor Building Alarm:

The reactor building alarm shall be operable and monitored continuously. CTS 3.2

Applicability: At all times.

3.8.2 Explosive Material at the Reactor Facility:

Explosive materials may be brought into the facility for the purpose of being radiographed provided that the following conditions are met: CTS 9.11

(1) Explosive material shall be stored in designated areas within the reactor facility. CTS 9.11.3

(2) Individual explosive devices shall be limited to 1000 grains equivalent TNT. CTS 9.11.1

(3) Only solid or encased explosive materials may be brought into the facility. DOC 3.8.2(3)

(4) Any explosive device containing loose explosive powders shall be completely encased. DOC 3.8.2(4)

(5) The maximum quantity of explosive material that may be possessed at one time shall be limited to 50 pounds equivalent TNT. CTS 9.11.2

(6) Unshielded high frequency generating equipment shall not be operated within 50 feet of any explosive device. CTS 9.12.2

(7) Personnel handling the explosive devices shall be trained and familiar with the devices being radiographed. CTS 9.12

(8) Personnel handling the explosive devices shall use special equipment such as non-sparking tools and shoes, protective clothing, safety shields, and grounded benches as required for the explosives being handled. CTS 9.12.1

(9) Only the explosive devices that do not exceed a maximum of ten pounds equivalent TNT and that will be radiographed within 8 hours, including preparation time, may be removed from the storage areas at one time. This restriction does not apply to explosives packaged for shipment. CTS 9.11.3.1

(10) A daily accountability log shall be maintained to show the amount of explosive material in the reactor facility and shall contain a description of the explosive material, and the location within the facility (e.g., storage, radiography facility, or shipping dock). CTS 9.11.3.2

Applicability: At all times.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

4.0 SURVEILLANCE REQUIREMENTS

4.0 GENERAL REQUIREMENTS

4.0.1 Surveillance Requirements:

SRs shall be met whenever the associated LCO is required to be met. DOC 4.01

4.0.2 Surveillance Requirement Frequency:

SRs shall be performed at the frequency specified for the SR as follows: DOC 4.02

(a) Five-year (interval not to exceed six years)
(b) Biennial (interval not to exceed two and one-half years)
(c) Annual (interval not to exceed fifteen months)
(d) Semiannual (interval not to exceed seven and one-half months)
(e) Quarterly (interval not to exceed four months)
(f) Monthly (interval not to exceed six weeks)
(g) Weekly (interval not to exceed ten days)
(h) Daily (must be done during the calendar day).

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

4.1 REACTOR CORE PARAMETERS

4.1.1 Rated Thermal Power (RTP):

(1) A reactor thermal power calibration shall be performed annually. Neutron Flux Channel 3 and Channel 4 shall reflect the results of this thermal power calibration. DOC 3.1.1

4.1.2 Excess Reactivity:

(1) Excess reactivity shall be verified to be within the limits of LCO 3.1.2 annually and following any significant change to the core or any control rod. DOC 3.1.2

4.1.3 Shutdown Margin (SDM):

(1) SDM shall be verified to be within the limits of LCO 3.1.3 annually and following any significant change to the core or any control rod. DOC 3.1.3

4.1.4 Reactivity Coefficients:

(1) The reactivity decrement at full power shall be verified to be within the limits of LCO 3.1.4 annually and following any significant change to the core or any control rod. DOC 3.1.4

4.1.5 Core Configuration:

(1) A core inventory shall verify that the reactor core lattice contains ≤ 90 TRIGA type fuel elements prior to reactor startup following the addition of any fuel elements to the reactor core. DOC 3.1.5

(2) An administrative review shall verify that the reactor core lattice contains ≤ 3.30 kg of U-235 prior to reactor startup following the addition of any fuel elements to the reactor core.

(3) A minimum of 20% of the fuel elements in the reactor core lattice shall be examined by visual inspection annually.

(4) Each fuel element in the core shall have been examined by visual inspection within the previous five years.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

4.2 REACTOR CONTROL AND SAFETY SYSTEMS

4.2.1 Operable Control Rods:

Free movement of each control rod for insertion is verified by the performance of SR 4.2.2 (1). DOC 3.2.1

4.2.2 Scram Time:

(1) The control rod scram times shall be measured and verified to be within the limits of LCO 3.2.2 semiannually and following any significant change to the core or any significant work on the control rods or the control rod drive system. DOC 3.2.2

4.2.3 Reactivity Insertion Rate:

(1) The reactivity worth of each control rod shall be measured annually and following significant changes to the core or any control rod. DOC 3.2.3

(2) The withdrawal speed of each control rod shall be measured semiannually and the maximum rate of reactivity addition verified to be within the limits of LCO 3.2.3.

4.2.4 Reactivity Insertion Monitoring:

(1) Subcritical multiplication levels shall be plotted in accordance with the requirements of LCO 3.2.4 during experiments from start of control rod withdrawal until reactor criticality. DOC 3.2.4

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

4.2.5 Reactor Sequence and Interlocks:

(1) The function of each of the following interlocks shall be verified annually: DOC 3.2.5

(a) Safety rod withdrawal is prevented by each of the following:

(i) The master switch is not in the ON position; or,

(ii) The safety system has not been reset; or,

(iii) Any one of the four nuclear instrument channels not in the OPERATE mode; or

(iv) The neutron flux channel 1 count rate is ≤120 counts per minute unless bypassed when channel 2 exceeds a fixed setting of approximately 1×10^{-10} amps.

(b) Withdrawal of the shim rod and withdrawal of the regulating rod is prevented until the safety rod is withdrawn to its upper limit.

(c) Simultaneous withdrawal of the shim rod and the regulating rod is blocked.

4.2.6 Reactor Protection System Instrumentation:

(1) Channel checks of each of the applicable channels listed in Table 3.2-1 shall be performed prior to exiting the reactor secured condition on the first reactor startup each day. DOC 3.2.6

(2) Channel tests of each of the applicable channels listed in Table 3.2-1 shall be performed semiannually.

(3) Channel calibrations of each of the applicable channels listed in Table 3.2-1 shall be performed prior to initial use.

4.2.7 Reactor Monitoring System Instrumentation:

(1) Channel checks of each of the applicable channels listed in Table 3.2-2 shall be performed prior to exiting the reactor secured condition on the first reactor startup each day. DOC 3.2.7

(2) Channel tests of each of the applicable channels listed in Table 3.2-2 shall be performed semiannually except for the reactor water radioactivity channel which shall be tested prior to exiting the reactor secured condition on the first reactor startup each day.

(3) Channel calibrations of each of the applicable channels listed in Table 3.2-2 shall be performed prior to initial use except for the reactor water radioactivity channel which shall be calibrated biennially.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

4.3 COOLANT SYSTEMS

4.3.1 Reactor Tank Water Level:

Periodic verification of reactor water tank level is addressed in SR 4.2.6.1, SR 4.2.6.2, and SR 4.2.6.3.

4.3.2 Primary Coolant Water Temperature:

Periodic verification of primary coolant water temperature is addressed in SR 4.2.6.1, SR 4.2.6.2, and SR 4.2.6.3.

4.3.3 Primary Coolant Water Quality:

(1) The pH and conductivity of the primary coolant shall be measured verified to be within the limits of LCO 3.3.3 monthly. CTS 4.2

(2) The radioactivity level of the primary coolant shall be analyzed annually. DOC 3.3.3

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

4.4 VENTILATION SYSTEMS

4.4.1 Control Room:

(1) The control room shall be verified to be at a positive air pressure with respect to the reactor room quarterly. DOC 3.4.1

4.4.2 Reactor Building Circulation Fans:

(1) Proper operation of the emergency shutoff from the control room of all air conditioning systems and building circulating fans in or normally open to the high bay area shall be verified quarterly. DOC 3.4.2

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

4.5 CRITICALITY ALARM, RADIATION AND RADIOACTIVE EFFLUENT MONITORING

4.5.1 Criticality Alarm and Area Radiation Monitoring: DOC 3.5.1

(1) A channel test of the channel listed in Table 3.5-1 shall be performed quarterly. This test shall include the reactor building siren and receipt of the alarm by the automatic alarm monitoring company.

(2) A channel calibration of the channel listed in Table 3.5-1 shall be performed biennially.

(3) Radiation sensitive badges placed at strategic locations within the reactor building shall be analyzed quarterly.

4.5.2 Gaseous Effluent Monitoring: DOC 3.5.2

(1) A channel test of the channel listed in Table 3.5-1 shall be performed prior to exiting the reactor secured condition on the first reactor startup each day.

(2) A channel calibration of the channel listed in Table 3.5-1 shall be performed biennially.

4.5.3 Particulate Effluent Monitoring: DOC 3.5.3

(1) The reactor room particulate sampler shall be verified to be operating daily each day the reactor is operated.

(2) The reactor room particulate sample shall be counted monthly.

4.5.4 Post Accident Radiation Monitoring: DOC 3.5.4

(1) The two radiation detector packets containing threshold detectors shall be verified to be in place annually.

4.5.5 Portable Radiation Monitoring Instruments:

(1) Channel calibrations of each portable radiation monitoring instrument required by LCO 3.5.5 shall be performed quarterly. DOC 3.5.5

4.5.6 Radioactive Effluent Limits:

(1) Ar-41 production shall be determined annually. DOC 3.5.6

(2) Administrative verification that normal releases of radioactive effluents from reactor operation did not exceed 10 CFR 20 limits shall be performed annually.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

4.6 EXPERIMENTS

4.6.1 Evaluation and Approval of Experiments:

(1) The reactor operator shall verify that experiments have been evaluated and approved in accordance with the requirements of Technical Specification 6.5.1 prior to insertion of the experiment into the reactor. DOC 3.6.1

4.6.2 Reactivity Limits during Experiments:

No requirements. SR 4.6(1) will ensure that the requirements of LCO 3.6.2 are met.

4.6.3 Special Nuclear Material (SNM) included in Experiments:

(1) The reactor operator shall verify that experiments that introduce SNM into the reactor meet the requirements of LCO 3.6.3 prior to insertion of the experiment into the reactor. DOC 3.6.3

4.6.4 Materials Used during Experiments:

No requirements. SR 4.6(1) will ensure that the requirements of LCO 3.6.2 are met.

4.6.5 Explosive Materials in Experiments:

(1) Administrative controls shall verify, as applicable, that the requirements of LCO 3.6.5 are being implemented. DOC 3.6.5

4.6.6 Configuration and Potential Failure Mechanisms:

No requirements. SR 4.6(1) will ensure that the requirements of LCO 3.6.2 are met.

4.6.7 Glory Hole Facility:

(1) Continuous gas samples shall be taken near the glory hole opening and Argon-41 presence shall be monitored when the glory hole facility is operated without a shield plug. CTS 8.3.2

4.6.8 Vertical Tubes:

(1) Verification that each vertical tube is being purged continuously with CO_2 shall be performed daily. CTS 8.6.2

(2) Gas samples shall be taken near the pool continuously and Argon 41 presence shall be monitored when the vertical tube is inserted.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

4.6.9 Other Irradiation Facilities:

No requirements. SR 4.6(1) will ensure that the requirements of LCO 3.6.2 are met.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

4.7 FUEL STORAGE AND TRANSFER

4.7.1 Fuel Handling Tool:

(1) Verification that the fuel handling tools are secured with a locking mechanism under the cognizance of the Reactor Supervisor when not authorized for use shall be performed weekly. DOC 3.7.1

4.7.2 Fuel Storage in the Reactor Tank:

(1) Verification that any planned addition or movement of fuel in the reactor tank that is not part of the reactor core lattice will not result in a geometric array where keff is > 0.9 for all conditions of moderation and reflection using light water shall be made prior to any addition or movement of fuel in the reactor tank. DOC 3.7.2

4.7.3 Fuel Storage in the Fuel Storage Pits:

(1) Verification that any planned addition or movement of fuel stored in each fuel storage pit will not result in > 19 fuel elements or > 700 grams of U-235 in each fuel storage pit shall be made prior to any addition or movement of fuel in the fuel storage pits. DOC 3.7.3

(2) Verification that each fuel storage pit containing a fuel element is secured with a lock and chain except during fuel transfer operations involving the fuel element being stored in the associated pit shall be performed quarterly.

4.7.4 Fuel not in the Reactor and not in Storage:

(1) Verification that any movement or receipt of fuel at the facility will not result in more than one fuel element in the facility which is not in the reactor core lattice or in storage shall be made prior to any movement or receipt of fuel at the facility. DOC 3.7.4

4.7.5 Fuel in Shipping Containers:

(1) Verification that an outgoing fuel shipping container is approved shall be made prior to use. DOC 3.7.5

4.7.6 Fuel Transfer in the Reactor Tank:

No requirements.

4.7.7 Fuel Transfer Outside the Reactor Tank:

No requirements.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

4.8 REACTOR FACILITY

4.8.1 Reactor Building Alarm:

(1) The Reactor Building Alarm system shall be tested monthly. CTS 3.2

4.8.2 Explosive Material at the Reactor Facility:

(1) Administrative controls that implement the requirements of LCO 3.8.2 shall be reviewed annually. DOC 3.8.2

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

5.0 DESIGN FEATURES

5.1 SITE AND FACILITY DESCRIPTION

5.1.1 A steel, locked perimeter fence shall surround the ARRR facility forming an exclusion area. The reactor and associated equipment are located within an exclusion area. The restricted area, as defined in 10 CFR 20, shall consist of the entire exclusion area. CTS 2.2

5.1.2 The minimum distance from the center of the reactor pool to the boundary of the exclusion area fencing shall be 50 feet. CTS 2.2

5.1.3 The principal activities carried on within the exclusion area shall be those associated with the operation of the reactor. CTS 2.3

5.1.4 The reactor shall be housed in a steel building. CTS 3.1

5.1.5 Reactor building ventilation shall be achieved by gravity ventilators located on the roof of the building. CTS 3.1.2

5.1.6 An alarm system shall be installed to detect unauthorized entry into the reactor building. CTS 3 2

5.2 REACTOR COOLANT SYSTEM

5.2.1 The reactor shall be cooled by the pool water which circulates by natural convection. DOC 5.2

5.2.2 The pool water shall be cooled by a pumped cooling system consisting of a primary and secondary loop.

5.2.3 The water purity shall be maintained by a mixed bed demineralizer.

5.3 REACTOR CORE AND FUEL

5.3.1 Reactor Core: DOC 5.3.1

(1) The reactor core shall consist of standard TRIGA fuel elements, graphite reflector elements, 3 control rods and guide tubes, a neutron source, and irradiation facilities.

(2) Fuel elements may be added to compensate for fuel burn up within the limits in Technical Specification 3.1.5. DOC 5.3.1

(3) The overall reflector elements' dimensions shall be the same as the fuel elements. CTS 5.2.1

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

(4) Core and fuel design shall ensure that reactivity coefficients are maintained as follows:

(a) The bath temperature coefficient and the prompt fuel temperature coefficient are negative at all operating temperatures. CTS 5.1.3

(b) The coolant void coefficient is negative across the active core. 5.1.4

(c) Maximum in-core operating void is 10% of the coolant core volume as defined by a cylinder bounded by the grid plates. 5.1.4

5.3.2 Reactor Fuel:

The TRIGA fuel elements used in the ARRR shall have the following nominal characteristics at fabrication: DOC 5.3.2

Fuel alloy:	uranium-zirconium hydride
Enrichment:	≤ 20 wt % U-235
Cladding:*	0.030 inch thick (aluminum), or 0.020 inch thick (stainless steel)
Fuel matrix:	8 wt. % (aluminum), or ≤ 12 wt. % (stainless steel)
Hydrogen-to-zirconium atom ratio in the ZrH_x:	1.0 to 1.1 % (aluminum), or 1.6 to 1.7 % (stainless steel)
Fuel loading:	36 to 38 grams (aluminum), or ≤ 55 grams (stainless steel)

5.3.3 Control Rods: CTS 5.3.2

(1) The reactor shall have three control rods: safety, shim, and regulating.

(2) Each control rod shall contain boron carbide as a neutron poison, sealed in an aluminum or stainless steel tube.

5.4 FISSIONABLE MATERIAL STORAGE

5.4.1 The fuel storage pits located in the floor of the reactor room shall each accommodate a maximum of 19 fuel elements and a maximum of 700 grams of U-235 in storage racks. The fuel storage pits may be dry or flooded with water. CTS 11.1

5.4.2 Fuel may be stored in fuel storage racks located in the reactor tank. Fuel in the reactor tank but not part of the reactor core shall be stored in a geometric array where keff is ≤ 0.9 for all conditions of moderation and reflection using light water. CTS 11.2 DOC 5.4.2

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

5.4.3 Fuel may be contained in an approved fuel shipping container within the limitations in LCO 3.7.4, Fuel not in the Reactor and not in Storage, and LCO 3.7.5, Fuel in Shipping Containers. DOC 5.4.3

5.4.4 A fuel handling tool shall be used for transferring fuel elements of low radioactivity between the storage pits and the reactor. A shielded fuel transfer cask shall be used for the transfer of highly radioactive fuel elements between the storage pits and the reactor. CTS 11.3

5.5 EXPERIMENTAL FACILITIES

5.5.1 Neutron Radiography Facility:

(1) The beam tube shall consist of a two-section tapered tube having a rectangular cross section. CTS 8.4.1

(2) The upper and lower sections of the beam tube shall be equipped with a fill and drain line. CTS 8.4.1

(3) All components contacting the pool water shall be fabricated from aluminum or stainless steel. CTS 8.4.2

(4) The beam catcher shield shall consist of a movable radiation shield. CTS 8.4.3

5.5.2 Thermal Column:

(1) The thermal column shall be composed of a three-foot cube of graphite encased in aluminum containing five rows of 1.5 inch diameter irradiation holes placed 6 inches apart with seven holes per row. CTS 8.5.2

(2) The thermal column shall be positioned remotely on steel locating pins immediately adjacent to the reactor core. CTS 8.5.1

(3) Slotted beams shall be provided to allow experiments to be attached directly to the thermal column. CTS 8.5.2

(4) Vertical irradiation tubes, having diameters up to 6 inches, may be attached to the thermal column. CTS 8.6.1

5.5.3 Other Irradiation Facilities:

(1) The central 7 fuel elements of the reactor may be removed from the core within the limits specified in LCO 3.6.9(1). CTS 8.7.1

(2) Two triangular exposure facilities are available which shall allow the insertion of circular experiments to a maximum of 2.35 inches diameter or triangular experiments to a maximum of 3.0 inches on a side. CTS 8.7.2

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION

6.1.1 Structure:

The management and operation of the reactor facility shall be in accordance with the organizational structure indicated in Figure 6.1. CTS 12.1.1

6.1.2 Responsibility:

(1) Responsibility for the safe operation of the reactor facility shall be with the chain of command established in Figure 6.1. CTS 12.1.1

(2) Individuals at all management levels shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to all requirements of the operating license and Technical Specifications. DOC 6.1.2

(3) Responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon appropriate qualifications. DOC 6.1.2

6.1.3 Staffing:

(1) The minimum staffing when in the reactor operating or reactor shutdown condition shall be:

(a) A licensed Reactor Operator in the control room. CTS 10.1

(b) A second designated person present at the facility able to carry out prescribed written instructions. Unexpected absence for as long as two hours to accommodate a personal emergency may be acceptable provided immediate action is taken to obtain a replacement. CTS 10.1 DOC 6.1.3(1)(b)

(c) A designated Senior Reactor Operator shall be readily available on call. Readily available on call means an individual who: DOC 6.1.3(1)(c)

(i) Has been specifically designated and the designation known to the operator on duty.

(ii) Keeps the operator on duty informed of where he may be rapidly contacted and the phone number.

(iii) Is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g., 30 minutes or within a 15-mile radius).

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

(2) Events requiring the presence of a Senior Reactor Operator at the facility include:

(a) Initial startup and approach to power. DOC 6.1.3(2)

(b) All fuel or control rod relocations within the reactor core region. CTS 11.4

(c) Relocation of any in-core experiment with reactivity worth ≥ 0.73% $\Delta k/k$ (≥ \$1.00). DOC 6.1.3(2)

(3) A list of reactor facility personnel by name and telephone number shall be readily available in the control room for use by the operator. The list shall include: DOC 6.1.3(3)

(a) Management personnel;

(b) Radiation safety personnel; and,

(c) Other operations personnel.

6.1.4 Selection and Training of Personnel:

The selection, training, and requalification of reactor operators shall meet or exceed the requirements of American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4-1988, Sections 4-6. DOC 6.1.4

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

6.2 REVIEW AND AUDIT FUNCTION

6.2.1 Reactor Safeguards Committee (RSC) Composition and Qualifications:

(1) The RSC shall be composed of not less than five members, of whom no more than three are members of the operating organization. CTS 12.1.3

(2) RSC members and alternates shall be appointed by and report to the President, Aerotest Operations, Inc. DOC 6.2.1

(3) RSC members shall collectively represent a broad spectrum of expertise in reactor, radiological, industrial, and explosive safety. DOC 6.2.1

(4) Individuals on the RSC may be either from within or outside the operating organization. CTS 12.1.3

(5) RSC members from outside the operating organization shall include a chairman and at least two individuals who are qualified in appropriate technologies listed in Technical Specification 6.2.1(3). DOC 6.2.1(5)

(6) RSC members from within the operating organization shall include:

(a) A Reactor Supervisor, who:

(i) Shall have responsibility of the reactor facility; CTS 12.1.1

(ii) Shall be responsible to the General Manager, Aerotest Operations, Inc. in all matters pertaining to reactor operations and to the Technical Specifications; CTS 12.1.1 DOC 6.1.1

(iii) Shall have a Bachelor's degree in Engineering or Physical Science; CTS 12.1.4

(iv) Shall have a minimum of 4 years experience in the operation of a nuclear facility during which competence in supervision and reactor operations shall have been demonstrated; and, CTS 12.1.4

(v) Shall hold a Senior Reactor Operator's license for the facility. CTS 12.1.4

(b) A Radiological Safety Officer, who:

(i) Shall review and approve all procedures and experiments involving radiological safety; CTS 12.1.2

(ii) Shall enforce rules, regulations and procedures relating to radiological safety; CTS 12.1.2

(iii) Shall conduct routine radiation surveys; CTS 12.1.2

(iv) Shall be responsible to the General Manager, Aerotest Operations, Inc.;

(v) Shall have a Bachelor's degree in Biological or Physical Science; CTS 12.1.5

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

(vi) Shall have a minimum of 2 years experience in personnel and environmental radiation monitoring programs at a nuclear facility; CTS 12.1.5

(vii) May be certified as a Health Physicist by the Health Physics Society in lieu of the education and experience requirements given above. CTS 12.1.5

(7) Qualified and approved alternates may serve in the absence of regular members.

6.2.2 RSC Charter and Rules:

(1) The RSC shall meet on the call of the chairman and shall meet at least annually and more frequently as circumstances warrant, consistent with effective monitoring of facility activities. CTS 12.1.3

(2) Official RSC action requires a quorum of not less one-half of the membership where the operating staff does not constitute a majority. DOC 6.2.2

(3) The dissemination, review, and approval of RSC meeting minutes shall be conducted in a timely manner. DOC 6.2.2

6.2.3 RSC Review Function:

(1) The following items shall be reviewed by the RSC:

(a) Determinations that proposed changes in equipment, systems, test, experiments, or procedures do not involve an unreviewed safety question. DOC 6.2.3

(b) All new procedures and major revisions thereto having safety significance, proposed changes in reactor facility equipment, or systems having safety significance.

(c) All new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity.

(d) Proposed changes in Technical Specifications, license, or charter.

(e) Violations of Technical Specifications, license, or charter.

(f) Violations of internal procedures or instructions having safety significance. CTS 12.1.3.5

(g) Operating abnormalities having safety significance.

(h) Reportable occurrences listed in 6.7.2.

(i) Audit reports.

(2) A written report or minutes of the findings and recommendations of the RSC shall be submitted to the President, Aerotest Operations, Inc. and to the RSC members in a timely manner after the review has been completed. CTS 12.1.3.6 DOC 6.2

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

6.2.4 Audit Function:

(1) The RSC Chairman shall be responsible for proper implementation of the audit function. CTS 12.1.3.4

(2) The audit function shall include selective (but comprehensive) examination of operating records, logs, and other documents. DOC 6.2.4

(3) Discussions with cognizant personnel and observation of operations should be used also as appropriate.

(4) In no case shall the individual immediately responsible for the area perform an audit in that area.

(5) The following items shall be audited:

(a) Facility operations for conformance to the Technical Specifications and applicable license or charter conditions, at least once per calendar year (interval between audits not to exceed 15 months);

(b) The requalification program for the operating staff, at least once every other calendar year (interval between audits not to exceed 30 months);

(c) The results of actions taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operations that affect reactor safety, at least once per calendar year (interval between audits not to exceed 15 months);

(d) The reactor facility emergency plan and implementing procedures at least once every other calendar year (interval between audits not to exceed 30 months).

(6) Deficiencies uncovered that affect reactor safety shall immediately be reported to the President, Aerotest Operations, Inc.

(7) A written report of the findings of the audit shall be submitted to the President, Aerotest Operations and to the RSC members within three months after the audit has been completed. CTS 12.1.3.6 DOC 6.2

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

6.3 RADIATION SAFETY

6.3.1 Radiation Safety Program:

The radiation safety program shall comply with the requirements of 10 CFR 20 and the guidelines of American National Standard for Radiation Protection at Research Reactor Facilities, ANSI/ANS-15.11-1993. DOC 6.3.1

6.3.2 Radiological Safety Officer:

(1) The Radiological Safety Officer shall be assigned the responsibility for implementing the radiation protection program. CTS 12.1.2

(2) The Radiological Safety Officer shall report to the General Manager, Aerotest Operations, Inc. CTS 12.1.2

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

6.4 PROCEDURES

6.4.1 Written Procedures:

(1) Written procedures shall be prepared, reviewed, and approved for each of the following prior to initiating these activities: CTS 12.2.1

(a) Startup, operation, and shutdown of the reactor. CTS 12.2.1.1

(b) Loading, unloading, and movement of fuel within the reactor. CTS 12.2.1.2

(c) Maintenance of major components of systems that could have an effect on reactor safety. CTS 12.2.1.5

(d) Surveillance checks, calibrations, and inspections required by the Technical Specifications or those that may have an effect on reactor safety. CTS 12.2.1.5

(e) Personnel radiation protection, consistent with applicable regulations or guidelines. These procedures shall include management commitment and programs to maintain exposure and release as low as reasonably achievable (ALARA) in accordance with the guidelines of ANSI/ANS-15.11-1993. DOC 6.4.1

(f) Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity.

(g) Implementation of required plans such as emergency or security plans.

6.4.2 Procedure Approval:

Procedures shall be reviewed by the RSC and approved by the Reactor Supervisor or designated alternates and such reviews and approvals shall be documented in a timely manner. CTS 12.1.3.2

6.4.3 Procedure Changes:

(1) Temporary procedures which do not change the intent of previously approved procedures may be utilized on approval by a Senior Reactor Operator and one other qualified individual. Such procedures shall be subsequently reviewed by the RSC. CTS 12.2.2.

(2) Substantive changes to the previous procedures shall be made effective only after documented review by the RSC and approval by the Reactor Supervisor or designated alternates. CTS 12.2.2

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

6.4.4 Deviation from Procedures:

Temporary deviations from the procedures may be made by a Senior Reactor Operator in order to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported to the Reactor Supervisor or designated alternates.

DOC 6.4.4

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

6.5 EXPERIMENTS

6.5.1 Experiment Review and Approval:

(1) All new experiments or class of experiments shall be reviewed by the RSC and approved in writing by the Reactor Supervisor or designated alternates prior to initiation. CTS 12.1, 3.2

The documentation of experiments shall include the following: CTS 9.2

(a) The purpose of the experiment;

(b) A description of the experiment; and

(c) An analysis of the possible hazards associated with the performance of the experiment.

(2) The description, review, and approval shall be documented as shown in the Aerotest Experiment Type Review (AETR) form. Supporting material may be attached to that form as required. DOC 6.5.1

(3) The safety review for each experiment type shall include but not be limited to those items delineated on the AETR form and any other items which could credibly reduce reactor safety or subject personnel to unacceptable hazards. DOC 6.5.1

(4) The safety review for each experiment type shall evaluate the experiment in the most reactive condition. CTS 9.1

6.5.2 Experiment Performance: DOC 6.5.2

(1) It shall be the responsibility of the Reactor Supervisor to delegate responsibilities as required to ensure that experiments fall within approved experiment types and restrictions.

(2) It shall be the responsibility of all persons to immediately notify appropriate personnel if an experiment does not fall within established limits.

(3) The console operator shall review all information and shall exercise final control over that particular experiment with respect to reactor operation.

(4) The Reactor Supervisor shall be notified immediately if a deviation occurs or a problem arises.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

6.5.3 Changes to Experiments:

(1) Substantive changes to previously approved experiments shall be made only after review by the RSC and approved in writing by the Reactor Supervisor or designated alternates. CTS 12.1.3.2

(2) Minor changes that do not significantly alter the experiment may be approved by the Reactor Supervisor or designated alternates. DOC 6.5.3

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

6.6 REQUIRED ACTIONS

6.6.1 Action to be Taken in Case of Safety Limit (SL) Violation: CTS 10.3 DOC 6.6.1

(1) The reactor shall be placed in the reactor secured condition, and reactor operations shall not be resumed until authorized by the Nuclear Regulatory Commission (NRC).

(2) The SL violation shall be promptly reported to the Reactor Supervisor or designated alternates.

(3) The SL violation shall be reported to the NRC.

(4) A SL violation report shall be prepared. The report shall describe the following:

(a) Applicable circumstances leading to the violation including, when known, the cause and contributing factors;

(b) Effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public; and,

(c) Corrective action to be taken to prevent recurrence.

(5) The report shall be reviewed by the RSC and any follow-up report shall be submitted to the NRC when authorization is sought to resume operation of the reactor.

6.6.2 Action to be Taken for Reportable Event described 6.7.3: CTS 10.3 DOC 6.6.2

(1) Any event that may be reportable in accordance with Technical Specification 6.7.3 shall be reported to the Reactor Supervisor or designated alternate immediately; and,

(2) Action shall be initiated immediately to place the reactor and facility in a condition where the reportable condition no longer exists or the reactor is in the reactor secured condition. Actions shall continue until the reportable condition is corrected.

(3) If the reactor is placed in the reactor shutdown condition, the reactor shall not be placed in the reactor operating condition until authorized by the Reactor Supervisor or designated alternate.

(4) Reactor Supervisor or designated alternate shall make a determination if the event is reportable in accordance with Technical Specification 6.7.3.

(5) Event shall be reviewed by the RSC at their next scheduled meeting.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

6.7 REPORTS

6.7.1 Operating Reports: DOC 6.7.1

(1) *Routine operating reports covering the operation of the facility during the* previous calendar year shall be submitted to the NRC and shall include the following:

(a) The Annual Summary of Changes, Tests, and Experiments as required by 10 CFR 50.59.

(b) The Annual Occupational Exposure Reports as required by 10 CFR 20.2206.

(c) The Annual Material Status Reports as required by NUREG/BR-0007

(d) The Nuclear Material Transaction Reports as required by NUREG/BR-0006.

6.7.2 Changes to the Facility or Organization: DOC 6.7.2

(1) A written report to the NRC shall be made within 30 days of any of the following:

(a) Permanent changes in the facility organization involving the President, Aerotest Operations, Inc. or the General Manager.

(b) Significant changes in the transient or accident analysis as described in the Safety Analysis Report. License Cond 2.D.3

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

6.7.3 Reportable Events:

(1) There shall be a report no later than the following working day by telephone and confirmed in a written report by FAX or similar conveyance to the NRC that describes the circumstances of the event within 14 days of any of the following:

License Cond 2.D

(a) Violation of a Safety Limit;

(b) Release of radioactivity from the site above allowed limits;

(c) Any of the following:

(i) Operation with actual safety-system settings for required systems less conservative than the Limiting Safety System Settings specified in the Technical Specifications.

(ii) Operation in violation of LCO established in the Technical Specifications unless prompt remedial action is taken.

(iii) A reactor protection system component malfunction which renders or could render the reactor protection system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdowns. (Note: Where components or systems are provided in addition to those required by the Technical Specifications, the failure of the extra components or systems is not considered reportable provided that the minimum number of components or systems specified or required by the Technical Specifications perform their intended reactor safety function.)

(iv) An unanticipated or uncontrolled change in reactivity $\geq 0.73\%\ \Delta k/k$ ($\geq \$1.00$). Reactor trips are excluded.

(v) Abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or containment boundary (excluding minor leaks) where applicable, which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both.

(vi) An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

6.8 RECORDS

6.8.1 Records Format: DOC 6.8.1

Records may be in the form of logs, data sheets, or other suitable forms. The required information may be contained in single or multiple records, or a combination thereof.

6.8.2 Records to be Retained for Five Years: DOC 6.8.2

(1) The following records shall be retained for a period of at least five years or for the life of the component involved if less than five years:

(a) Normal reactor facility operation (but not including supporting documents such as checklists, log sheets, etc., which shall be maintained for a period of at least one year);

(b) Principal maintenance operations;

(c) Reportable occurrences;

(d) Surveillance activities required by the Technical Specifications;

(e) Reactor facility radiation and contamination surveys where required by applicable regulations;

(f) Experiments performed with the reactor;

(g) Fuel inventories, receipts, and shipments;

(h) Approved changes in operating procedures;

(i) Records of meeting and audit reports of the RSC.

6.8.3 Records to be Retained for at Least One Certification Cycle: DOC 6.8.3

Records of retraining and requalification of licensed operators shall be maintained at all times the individual is employed or until the license is renewed.

6.8.4 Records Retained for Life of the Facility: DOC 6.8.4

(1) The following records shall be retained for the life of the reactor facility:

(a) Gaseous and liquid radioactive effluents released to the environs;

(b) Radiation exposure for all personnel monitored;

(c) Drawings of the reactor facility.

(2) Applicable annual reports, if they contain all of the required information, may be used as records in this section.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

6.8.5 Facility Specific Records:

(1) In addition to those records required under the facility license and applicable regulations, the following records shall be kept for the life of the reactor facility: License Cond 2.E

(a) Reactor operating records, including power levels;

(b) Records of in-pile irradiations;

(c) Records showing radioactivity released or discharged into the air or water beyond the effective control of the licensee as measured at the point of such release or discharge; and,

(d) Records of emergency reactor scrams, including the reasons for emergency shutdowns.

(2) The following records shall be kept for 7 years when explosive materials are to be irradiated or radiographed: CTS 12.3

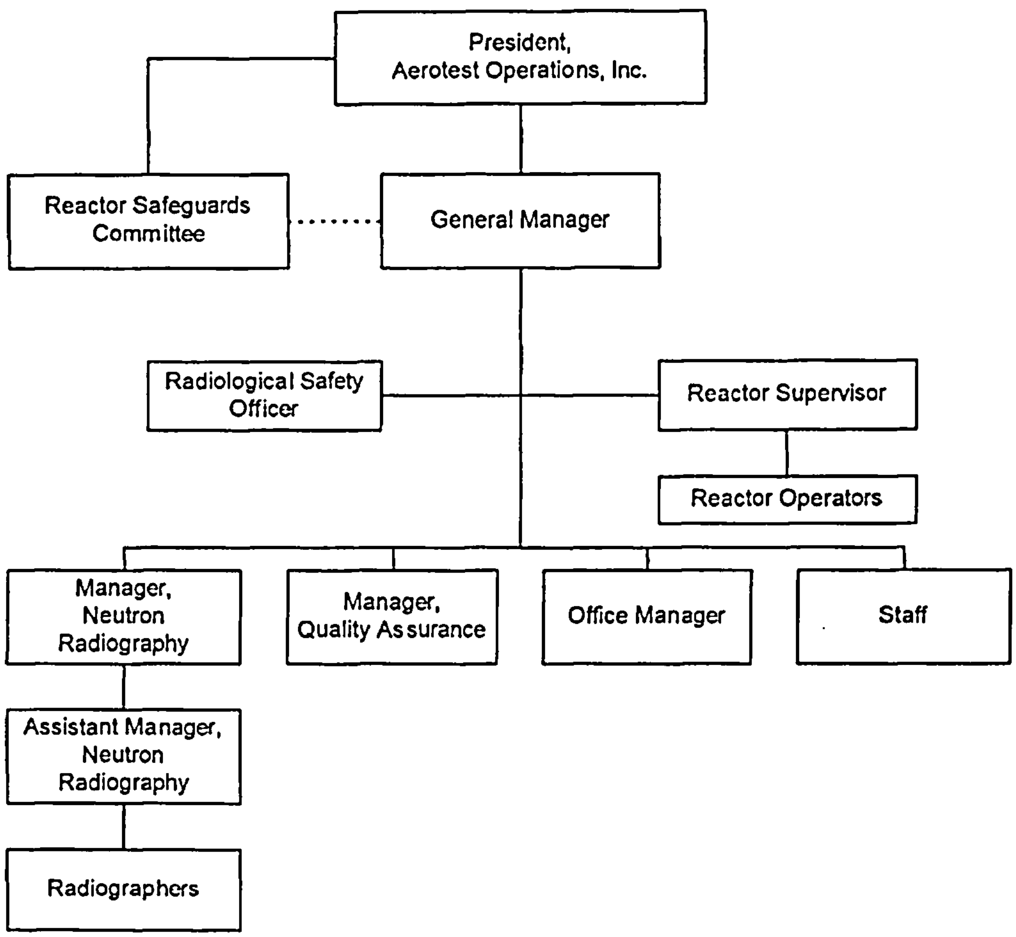
(a) The type and quantity of explosive material;

(b) The date, time of day, and length of exposure;

(c) The total neutron and gamma exposure levels.

Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).

Figure 6.1
Aerotest Operations, Inc.
Organization Chart



Annotated to show differences between Current Technical Specifications (CTS) and proposed Technical Specifications (TS).