1. Written Response to RAI-GEN 1 and RAI-GEN 2 (without Appendix A)
2. AO Environmental Report-2017
3. USAR with Non-Proprietary Chapter 15
4. Technical Specifications (TS)
5. Proprietary RAI-GEN 2 Appendix A
6. Proprietary USAR Chapter 15
7. DM Slaughter Affidavit
8. Electronic PDF copies of enclosures 1, 2, 3, and 4

AEROTEST OPERATIONS, INC.

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Aerotest Radiography and Research Reactor
Docket No. 50-228
Enclosure 1.

February 28, 2005 ARRR License renewal application (ML13120A434) was submitted. Two RAIs were received from the NRC on October 24th 2017 requesting to update the R-98 application. Aerotest Operation's response follows:

RAI GEN-1 Please provide updates and/or supplements, as appropriate, such that the ARRR License renewal application, as supplemented, will be complete and accurate in all materials respect or, state that no updates and/or supplements to the application be required.

The Environment Report attached (Enclosure 2) supports the ARRR relicense application. Briefly the report summarizes the salient features of the reactor facility, supporting auxiliary systems, and relevant requirements and practices to ensure safe facility operation. The current facility maintains natural and engineered safety and control systems, and effectively monitors regulated effluents including radiological releases to minimize environmental impact. In addition, the existing personnel radiation safety program is more than adequate to ensure the health and well-being of Aerotest workforce as well as the general public.

As provided in USAR-TS (Enclosures 3, and 4), the Aerotest ARRR (R-98) reactor has operated 45 years with little change in circumstances before shutting down in 2010 over concerns of foreign ownership. The core structural members and plates are the same without modification since ARRR went critical in 1965 and reflects the fuel-element layout of other TRIGA reactors with the fuel distribution within 6 concentric rings surrounding a central port. The irradiators (thermal-column, radiography, etc) have operated in place for more than 30 years. The ARRR uses natural circulation convective cooling with no recent (>30 years) core modifications that reduced its effectiveness. ARRR has strategically placed safety and shim rods in the C-ring and a regulating rod in the E (or D) ring as standard practice for a three control rod system.

The ARRR TRIGA design operates with a standard TRIGA 8.5 and 12 percent by weight of 20 percent enriched in U-235. All the TRIGA fuel elements used in ARRR were manufactured with the same the chemical and mechanical processes along with extensive US government oversight; this includes the newer stainless-steel clad, 12 percent uranium weight and enrichment specifications that were manufactured by TRIGA International partner, CERCA. Given no changes to the core structure, irradiators, and fuel, the nucleonic and thermal hydraulic behavior

and operating characteristics associated with this facility will not change when adhering to the safety and operating limits detailed in the ARRR Technical Specifications. This is clear from the extensive experimental studies of fuel and core characteristics during operation conducted by General Atomics (fuel-and core-designer), performance studies of the integrated reactor system conducted by TRIGA facility owners, and operational experience and data compiled by Aerotest with the ARRR. Recent modeling of nucleonics and thermal hydraulics of similar core design and fuel for licensed power of 100 KW, 250 KW, and 1 MW, support the findings produced from the earlier experimental studies. However at this time a significant uncertainty exists in the modeling results due to the simplifying assumptions applied in the models and inexact mathematical representative of simplified processes found in the algorithms.

ARRR maintains an up-to-date regularly practiced emergency (October 1982) and security (July 1991) plans and procedures to ensure effective safe and secure operation for personnel and community alike. Small changes to the current emergency and security plans will be submitted under separate cover through a 10 CFR 50.54p/q request. Most of the changes are administrative in nature and no single or combination of changes will reduce the effectiveness of either plan; thus submission of these plans under 10 CFR 50.54q/p is appropriate.

RAI Gen-2 Aerotest currently holds a class 104c license for the ARRR. Please justify that Facility Operating License No. R-98 for the ARRR should be renewed as a class 104c license. Alternatively, clarify that Facility Operating License No. R-98 for the ARRR should be renewed as a different class of license; or, justify why no additional information is required.

Given the reactor design and operating characteristics, its use, and compliance with regulatory statute, the 104c is the appropriate NRC license (R-98) for this TRIGA 250 KW nuclear reactor that went critical in 1965, and is under timely renewal now in 2017.

The conclusion is based on a number of relevant documents including salient portions of 10 CFR 50.21, 50.22, 10 CFR 170.3, Atomic Energy Act-1954 as amended, ARRR Facility history, costs of activities associated with Aerotest Operations, Inc. and the current business plan of the new President (and owner).

AEA-54 and 10 CFR

Section 10 CFR 170.3 defines a research reactor as "a reactor licensed by the commission under the authority of subsection 104c of the Act and pursuant to the provisions of 10 CFR 50.21(c) of this chapter for operation at a thermal power level 10 Megawatts or less, and which is not a testing facility…."

10 CFR 50.21 104 (c) states "A production or utilization facility, which is useful in the conduct of research and development activities of the types specified in section 31 of the Act, and which is not a facility of the type specified in paragraph (b) of this section or in 10 CFR 50.22." In part 10 CFR 50.22 adds the language, *"Provided, however,* That in the case of a production or utilization facility which is useful in the conduct of research and development activities of the

types specified in section 31 of the Act, such facility is deemed to be for industrial or commercial purposes if the facility is to be used so that more than 50 percent of the annual cost of owning and operating the facility is devoted to the production of materials, products, or energy for sale or commercial distribution, or to the sale of services, other than research and development or education or training. "

Also, the language provided in Sec. 106 of the AEA suggests that the size, design and operating characteristics also influence the type of license assigned. "The Commission may a. group the facilities licensed either under section 103 or under section 104 into classes which may include either production or utilization facilities or both, upon the basis of the similarity of operating and technical characteristics of the facilities; b. define the various activities to be carried on at each such class of facility; and......."

The phrase "Annual cost of owning and operating the facility that is devoted...." is not clearly defined and therefore must be better understood. By using the AEA1-1954 as amended as a guide "facility" may be taken as a "Production Facility" or "Utilization Facility". Since ARRR (R-98) is clearly not a production facility, therefore for this discussion it is assumed to be an utilization facility. The term "utilization facility" means (1) any equipment or device, except an atomic weapon, determined by rule of the Commission to be capable of making use of special nuclear material in such quantity as to be of significance to the common defense and security, or in such manner as to affect the health and safety of the public, or peculiarly adapted for making use of atomic energy in such quantity as to be of significance to the common defense and security, or in such manner as to affect the health and safety of the public; or (2) any important component part especially designed for such equipment or device as determined by the Commission. (*In the 10 CFR 50 Utilization facility* is defined as: "(1) Any nuclear reactor other than one designed or used primarily for the formation of plutonium or U–233; or....")

While it is not defined in detail, anywhere in the AEA or 10 CFR, the "facility" may include the reactor fuel rods, core structure, control rods and mechanisms, and instrumentation (detector and controls). The "annual cost" most likely includes the operating costs associated with auxiliary equipment/systems i.e., tank (or primary containment) water heat-exchanger and air control, etc., that are required for safe operation based on its design. It is not clear if fixtures, and their maintenance (permanent irradiators and/or temporary experiments attached to the utilization facility should be considered. (The design, fabrication, and certification of the fixtures for an activity clearly are not; those are considered a development cost.)

Section 10 CFR 50.22 identifies a class of activities performed on the utilization facility that " is deemed to be for industrial or commercial purposes if the facility is to be used so that more than 50 percent of the annual cost of owning and operating the facility is devoted to the production of materials, products, or energy for sale or commercial distribution, or to the sale of services, other than research and development or education or training."

It is interesting to note that nowhere in the AEA 1954 (as amended) that this statement appears wholly or in part. This language will be referred to as the "Metric" for simplicity. The language contained in the Metric does not provide any special consideration to ownership (public, federal, state, or private), organizational structure (non-profit, educational or industrial institutions), etc.,

but to the activity performed on the utilization or production facility. It's sole focus is on the "annual cost of owning and operating the facility" doing activities "devoted to the production of materials, products, or energy for sale or commercial distribution, or to the sale of services". Simply, there is no guidance to define acceptable annual costs or a clear understanding of the discrimination between activities except in very general terms.

The lack of detail specificity in the Metric suggest that the authors of the Metric were placing a barrier to exclude large nuclear reactors that generate electrical power or large quantities of radioisotopes to claim relief from the more significant regulatory obligations experienced as a 103 licensee than those regulatory rules enforced under the 104 license. The significant cost of owning and operating these large scale utilization reactors for commercial commodities (power and radioisotope production) clearly satisfy the Metric without the need for detail definitions and accounting practices to be assigned a 103 license under 10 CFR 50.22. The location of the Metric in the text supports this conclusion; the statement is found in 10 CFR 50.22 and only briefly cited in 10 CFR 50.21. The mid 1970s introduction of the Metric's inclusion in the regulation further reinforces the supposition that the Metric was introduced to exclude power utilities from applying and holding the 104c license.

ARRR Historical Activities and Future Plans

Aerotest Operations ARRR (R-98) was constructed in 1964 and went critical in 1965 under Aeroject-General Nucleonics and was used as a research test bed similar to other low-power TRIGA research reactors. Research activities included but not limited to, neutron flux analysis, thermal/hydraulics, energy distribution, foil activation, fission-track analysis NAA, neutron radiography, radioisotope/tracer research etc. In the 1970s the initial activities began to focus on technologies related to neutron interrogation of materials using neutron radiography. This included the design and installation of an experimental 5-aperture column/duct. Improvements and modification to this irradiator were ongoing for the next 15 years.

In the late 1980s, the type of activities performed on the reactor were now focused on the art and science of material interrogation using neutron imaging. Included in these activities were Gd-doping, Gd-tagging studies of manufactured parts and developing high-resolution conversion/scintillator screens by vapor deposited Gd, Li-6, and/or B-10 to increase the effective resolution of the image in the radiographs. This change in emphasis coincides with the installment of Hugo Simens as President. Ray Tsukimura became President of Aerotest Operation in 1994; under his leadership the technical activities remained focused, but he expanded on the education/training, research and development and investigative (forensic-research) services associated with this topic. Collaborative research, development and investigative activities with Universities, National Laboratories and international equivalents flourished. Most of the work activities applied new science, engineering and/or intellectual knowledge. Only a small portion was classified as strictly production of a high quality neutron radiograph. This work used existing technologies and knowledge without using interruptive or diagnostic skills to understand the object's condition or circumstance.

Dario Brisighella (from Autoliv) became President of Aerotest Operations after the 2008 retirement of Ray Tsukimura. The same research, development and investigative activities with Federal research, educational and international collaborations continued, but to a lesser extent. Early under his leadership Aerotest Operations' direction was being reevaluated as the parent company, Autoliv sought a buyer for the company. A number of meetings were held at his Ogden location where he evaluated the current research activities. The outcome of those meeting included the continuance of the University of Utah MOU and the development of a MOU with UC-Berkeley. ARRR was shut down due to the foreign ownership issue in 2010.

David M. Slaughter, Ph.D. was made President of Aerotest Operations, Inc.in 2017. His background focus is on education and research, nuclear forensics, and nuclear reactor operations and that focus will be the center piece of his tenure. Once the TRIGA 250 KW reactor is operating, neutron detection /interrogation and Mo-99 target research will be added to Aerotest Operations already substantial list of research activities. The nature of these activities are compliant with the AEA section 31 requirements.

Cost of Owning and Operating ARRR

For this analysis, cost was assigned due to function and activity (see Appendix A (Proprietary), Enclosure 4). Based on the Metric definition, the only cost that fits the definition is the annual percentage of direct costs associated with reactor operation for Professional Sales (of neutron radiographs) and that has been estimated at 15% annually between 2000-2010 (Item 5). If 3% (1/5 of item 2 is also included, the result would be 18%; this outcome is lower than the Metric defines. (Item 1 was funded entirely by the parent company Autoliv).

1. 15% Indirect Costs ANI (and other) Insurance, fuel purchase and reactor upgrades, NRC, regulatory Costs (annual fee, inspections, PM, etc.), and Decommissioning & fuel-storage annual contribution.
2. 15% Indirect Costs associated with maintaining building, its infrastructure, and Fed/State/Local regulatory compliance other than reactor operation.
3. 30% Direct Cost associated with nonreactor operations activities
4. 15% Direct Costs Reactor Operation for Research/Development and Investigative (forensic) services (RDI).
5. 15% Direct Costs for Reactor Operation for Professional Sales (PS).
6. 10% Direct Costs for Education/Training activities and Community participation

The Total Annual Cost associated with the Metric from 2010-2017 clearly plummeted from 18% to 0 %.

However, the parent company has recently changed from Autoliv to Nuclear Labyrinth LLC, a small business. If it is assumed that the corporate structure was that of a small business and the activities remain the same in the reported proportions, the relevant costs include direct, 15% (Item 5), indirect, 7.5% (1/2 of 15% , Item 1), and 3% (1/5 of 15%, Item 2). The result would be 25.5%; this outcome would still be lower than the Metric defines. In the future, activities will include research in advance neutron detection/interrogation and Mo-99 target development.

These reactor activities will further lower the proportion of the reactor operating costs associated with N-radiograph sales and thus increasing the gap between the 50% Metric and the lower estimated reactor operating costs due to commercial activities.

It is interesting to note that since the ARRR shutdown in 2010 that a significant amount of activities that were assigned to item 5 has migrated to university owned reactors. The cost of "owning and operating" these facilities are significantly lower in part due to the subsidizes through DOE fuel ownership, DOE's federally funded reactor upgrade programs and cost exceptions from NRC for annual fees and regulatory compliance.

University administrations over the years have reduced the direct and indirect funding associated with education. Lower utilization for research and teaching forced many of these programs to seek alternative funding to cover the "cost of operation." Many of these services are not classified as research, development or investigative in nature. See the definition of professional sales given in the attachment. If I use the same assumptions put forward in my analysis, it is my belief, that many of research reactors above 1 MW power at Universities will fail the 104c license Metric. The University of Missouri program serves as a predominate example. Its cost of "owning and operating" is associated with "commercial" radioisotopes. Also in its recent past it also produced large quantities of irradiated gemstones for an international and national market. However, because the Metric lacks detail specificity, there are a number of allowable assumptions that can gain a favorable outcome. As stated earlier I don't believe it was the original intent and use of the Metric provided in 10CFR 50.22 (authored in the70s) to be applied to test and research reactors when they clearly meet the requirements detailed in section 31 of the AEA (1954 as amended). The 104c designation is consistent with the intent of section 106 AEA as well.

Independent of the intent of the Metric, ARRR 250 KW TRIGA reactor should be relicensed as a 104c because it complies with 10 CFR.50.21 provisions and operates at a thermal power level of 10 Megawatts or less (10 CFR 170.3).

AEROTEST OPERATIONS, INC.

Aerotest Radiography and Research Reactor (ARRR)

Environmental Report

December 2017

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

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1.0 Introduction

1.1 Purpose of the Environmental Report

This Environmental Report supports an application by Aerotest Operations, Inc. to the U. S. Nuclear Regulatory Commission (NRC) to amend Facility Operating License R-98 (Docket No. 50-228) to renew the license for the Aerotest Radiography and Research Reactor (ARRR), currently in timely renewal since 2005. The ARRR is a steady state 250 kW TRIGA-type nuclear reactor that was originally licensed in 1965 and had been operated continuously until 2010.

As required by the National Environmental Policy Act of 1969 (NEPA), the NRC must consider environmental aspects of proposed Commission actions. As required by Part 50.30, Title 10 of the Code of Federal Regulations (CFR), this Environmental Report has been prepared in accordance with 10 CFR Part 51 to aid the Commission in complying with NEPA in connection with the proposed amendment to renew Facility License No. R-98 for the ARRR. This Environmental Report considers the probable environmental effects that can be attributed to the continued operation of the ARRR.

1.2 Need for Amendment to Renew Facility Operating License R-98

The Amendment to renew the license of the ARRR is needed to allow continued operation of the ARRR. The ARRR is one of a limited number of nuclear reactors in the United States capable of providing a neutron source for research and development and services such as neutron radiology. The ARRR is used mainly for neutron radiology. Neutron radiology is used for non-destructive testing and failure analysis of components used by the aerospace, military, and industrial communities. Specific uses of neutron radiography performed at the ARRR include the following:

1. Verification of the presence, absence, or correct placement of explosives, adhesives, o-rings, plastic components, and similar materials;
2. Detection of ceramic residue in the core of cast turbine blades prior to assembly in aircraft engines.
3. Verification of the integrity of welds in propellant tanks used in space flight;
4. Inspection of sustained-release drug delivery systems prior to use in patients with cancer and various neurological and skeletal disorders.

In addition to neutron radiology, the ARRR is used in other research, development and investigative services as a neutron source for activation analysis, irradiations, radiation damage studies, etc. 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 performed at the ARRR 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 ARRR is also used for research and development for new neutron radiography equipment and image quality indicators including conversion screens, neutron detectors, and beam definition devices.

2.0 Facility and Site Characteristics

2.1 Location

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 which uses the property for testing and research.

The ARRR site was originally selected to be above the potential flooding level and at a location where no earth fractures were known to exist. Although the ARRR was originally in an area of sparse population, this is no longer applicable. 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 residential population within the three mile radius of ARRR facility is conservatively bounded at 34,859 which is the US Census 2010 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. As described in the ARRR Updated Safety Analysis Report (USAR), locating TRIGA reactors in highly populated areas has been evaluated as acceptable because 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.

2.2 Reactor

The ARRR is a TRIGA Mark I type reactor that was designed and constructed by the Nuclear Division of Aerojet-General in 1964. It is an open pool type reactor with the pool (i.e., reactor tank) located below ground level. The reactor achieved initial criticality on July 9, 1965 with a licensed steady-state thermal power limit of 250 kW. 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. There is no pulsing capability which is sometimes a design feature for similar reactors.

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. As described in the ARRR Updated Final Safety Analysis Report (USAR), the ARRR relies entirely on the design of the TRIGA fuel elements for the prevention and mitigation of any credible accidents. The design of the TRIGA fuel and Technical Specification imposed operating limits allow the ARRR to respond to any credible event with no hazard to the public without reliance on any engineered safety features. No specific structures, systems, or components are assumed to be operable for the prevention or mitigation of any accident or the protection of the public health and safety.

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 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 3 other TRIGA reactors approved for steady state operation at 250 kW and an additional 2 that can be pulsed. Of the former, these are located at Argonne National Laboratory - West (constructed in 1977), University of Maryland (1974), and Reed College (1968). There are 5 TRIGA reactors approved for steady state operation between 100 and 300 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 potential for and consequences of an accident at the ARRR are no greater than those of other similar reactors using the same fuel systems.

As described in the ARRR Updated Safety Analysis Report (USAR), the largest ground motion expected from a reasonably expectable maximum earthquake on the fault closest to the ARRR facility can be adequately modeled within guidelines in NRC Regulatory Guide 1.60 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.

2.3 Reactor Building

The ARRR reactor building is made of steel with internal rooms built of fire resistant framing and sheetrock covering. The reactor control room and certain offices are housed in a single building. The reactor tank is embedded in the floor, extending 22 feet below and one foot above the floor surface. A [redacted] inch thick by [redacted] inch high block wall made of normal density concrete encloses the reactor area above the floor level. The top of this shield is covered with an [redacted] inch thick wooden shield. An automatic sprinkler system covers the entire building.

A steel, locked perimeter fence surrounds the ARRR facility to form the exclusion area required by and defined in 10 CFR 100 and the restricted area, as defined in 10 CFR 20.

2.4 Cooling, Makeup Water and Cleanup Systems

The ARRR cooling system is comprised of five basic parts: the reactor tank; the primary cooling loop; the secondary cooling loop; the demineralizer system; and, the reactor tank water make-up system.

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. The reactor tank is an aluminum cylinder 10 feet in diameter and 23 feet deep. The tank is open on the top with no openings below the water surface. When filled to its normal operating level of 22 feet (i.e., 17 feet above the top of active fuel), the reactor tank contains about 13,000 gallons of water.

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. 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.

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.

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

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

2.5 Ventilation 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. The ARRR accident analyses, described in the ARRR USAR, do not assume that the ARRR building acts as either a containment or confinement to mitigate the release of radioactivity following a reactor accident.

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. Confinement of airborne radioactive material to the reactor area is enhanced by a ventilation system that does not provide outside air directly into the reactor area. Areas adjacent to the reactor area are supplied with outside air and are maintained at a pressure that is slightly positive relative to the reactor area.

During any radiological event that could spread contamination or airborne radiation within the building or release it to the environment, the reactor 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 area at a pressure slightly higher than the reactor area are not shut off. Any inleakage into the reactor area is directed out of the building through three gravity ventilators in the roof over the reactor area. This prevents air from entering the reactor building through the reactor area and minimizes the potential for spreading airborne radiation or contamination from the reactor area to other parts of the building.

2.6 Liquid Waste Storage

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. This liquid waste storage system ensures that potentially contaminated liquid waste is properly collected, segregated, stored, and disposed.

The liquid waste storage system uses a sump tank to collect liquids from the trenches around the reactor, the heat exchanger building, the demineralizer building, and one of the chemical laboratory sinks. In order to minimize waste volume due to a large leak, 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.

Once collected in the sump tank, potentially contaminated liquids are completely isolated from the sanitary sewer system. An automatic sump pump moves the waste to the primary waste storage tank, one of two above ground storage tanks that are used for storage of liquid wastes which could be contaminated. The primary waste storage tank can be monitored by a GM counter connected to a ratemeter in the control room. A high liquid level indicator annunciates on the console when the level reaches about two feet from the top of the tank. When the primary tank is nearly full (which occurs infrequently), samples are taken for laboratory analysis of the level and type of radioactivity and a special discharge permit is obtained from the Central Costa County Sanitary District prior to discharge. A second above ground tank is available to store waste while the primary tank liquid is being analyzed for radioactivity prior to release.

3.0 Environmental Effects of ARRR Facility Operation

Construction of the ARRR facility was completed in 1964. The last significant construction activity at the ARRR facility was an addition to the reactor building that was completed in 1981. Since 1964, when the ARRR facility was built, there has been no noticeable effect on the terrain, vegetation, nearby waters, or aquatic life. The societal, economic, and esthetic impacts of the construction of the facility have been no greater than that associated with the construction of any other small industrial facility in the area.

No additional construction is anticipated that would be expected to have any significant effect on the terrain, vegetation, wildlife, or nearby waters or aquatic life.

3.1 Nuclear Fuel Cycle

At initial criticality, the ARRR core included 63 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 2017, the ARRR contains 55 aluminum clad elements and 27 stainless steel clad. Aluminum clad elements contain approximately [redacted] grams of U-235. Stainless steel clad elements contain approximately [redacted] grams of U-235. The addition of new fuel elements is limited by the ARRR license which currently allows possession of no more than 5.0 kilograms of U-235.

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 until 2010. 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. Between 1992 and 2003, average power generation at the ARRR averaged 333,300 kW-hours per year. Between 2000 and 2010, average power generation at the ARRR averaged 264,370 kW-hours per year.

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) 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.

3.2 Radiation Exposure of Personnel

The ARRR 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. 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] inch thick high density concrete that surrounds the reactor pool and the active radiography area which includes [redacted] inch thick wooden 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 arm's 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.

Individual radiation exposure is minimized by rotating the assignments of neutron radiography technicians. Employee doses for those performing radiography averaged 2.07 rem per year from 2000 to 2004. 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 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.

Although individual exposures have trended higher 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 large increase in the number of radiographs made in the last several years.

Radiation exposure for all ARRR personnel are maintained as low as reasonably achievable and well within the limits established in 10 CFR 20; therefore, radiation exposure from continued operation of the ARRR does not pose a significant risk to operating personnel or the general public.

3.3 Radioactive Gaseous Effluents

The primary sources of airborne radiation under normal operating conditions is the production of Nitrogen-16 (^{16}N) in the reactor water tank and the production of Argon-41 (^{41}Ar) from the 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. Personnel exposure to ^{16}N gamma is further reduced by operating the primary cooling loop which increases the transit time for ^{16}N to reach the water surface. 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, the environmental effects of the release of ^{16}N to the environment are negligible.

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 involving irradiation. Technical Specifications require that experiments with the potential for activation of significant quantities of air are purged with CO_2 to minimize the formation of ^{41}Ar and that ^{41}Ar presence is monitored during these experiments.

Aerotest Operations, Inc. uses the Environmental Protection Agency's COMPLY program for calculating emissions of ^{41}Ar for annual reporting and verification of compliance with the requirements of 40 CFR 61, National Emission Standards for Hazardous Air Pollutants, for radionuclides. The EPA COMPLY program determinations of annual ^{41}Ar production and the resulting effective dose equivalent 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. Between 1993 and 1997, the ^{41}Ar release rates were consistent and averaged 16.7 curies/year resulting in an effective does equivalent of 0.5 mrem/year. Beginning in 1998, the ARRR has performed fewer irradiation experiments and more neutron radiography resulting in a significant reduction in the amount of ^{41}Ar produced. Between 1998 and 2010 (the last year of regular operation), the ^{41}Ar release rate was consistent and averaged 0.051 curies/year resulting in an effective does equivalent of 0.0016 mrem/year.

The very low levels of ^{16}N and ^{41}Ar released to the environment as a result of the operation of the ARRR result in an effective dose equivalent to the public that is very small fraction of applicable limits established in 10 CFR 20, 10 CFR 50 Appendix I, and 40 CFR 61. Therefore, radiation exposure resulting from gaseous radioactive effluents resulting from the continued operation of the ARRR does not pose a significant risk to operating personnel or the general public.

3.4 Radioactive Liquid Wastes

Small quantities of liquid radioactive waste are generated by regeneration of the reactor water demineralizer and, infrequently, from liquids irradiated as part of sample irradiation. Otherwise, all the low level liquid waste generated at the ARRR 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 and activity levels are verified to be low enough for discharge to the sewer system. 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. Any contaminated liquid waste that does not meet criteria for disposal into the sanitary sewer system is stored for ultimate disposal by a licensed waste disposal company.

It should be noted that the ARRR generates very little liquid radioactive waste. Since 1992, there have been three discharges from the liquid waste storage system to the sanitary sewer system. Approximately 500 gallons was discharged in August of 2004, approximately 400 gallons was discharged in September of 2000, and approximately 900 gallons was discharged in April of 1992. Each discharge was performed in accordance with a Special Discharge Permit issued by the Central Conta Costa Sanitary District. Based on samples performed by a certified laboratory, the amount of radioactivity released during each discharge was a very small fraction of the annual limit imposed by applicable regulations. Future discharges are expected to occur very infrequently based on Aerotest Operations, Inc. efforts to minimize generation of radioactive liquid waste.

Based on the very low volume and the very low activity in radioactive liquid waste expected to be generated from the continued operation of the ARRR, radiation exposure resulting from radioactive liquid effluents does not pose a significant risk to operating personnel or the general public.

3.5 Radioactive Solid Wastes

Annual generation of solid radioactive waste at the ARRR facility is minimal and consists of neutron activated experiments, wastes from experiment handling (e.g., gloves, holders, plastic sheeting, tape, swipe samples and filter paper, etc.), check and calibration sources, and waste from the reactor water demineralizer (e.g., filters, ion exchange resin, etc.). 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.

Other than reactor fuel elements and reactor components, the ARRR generates essentially no high-level waste. Disposal of any high-level radioactive waste and spent nuclear fuel will be conducted in coordination with the Department of Energy, under the agreement described in Section 3.1 of this report.

3.6 Thermal Effluents

The ARRR is licensed to operate at a thermal power level of 250 kW; however, the ARRR has been operating at 180 kW or below since 1992. 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.

Thermal energy produced by the ARRR heats the reactor tank water. The primary cooling loop removes heat from the reactor tank water via a heat exchanger to the secondary cooling loop. The secondary cooling loop, when required for reactor water tank temperature control, removes heat from the primary loop heat exchanger to the environment via an evaporative cooling tower.

The environmental effects of thermal effluents from the evaporative cooling tower needed to cool the ARRR are negligible. Therefore, thermal effluents resulting from the continued operation of the ARRR is not expected to have any significant effect on the terrain, vegetation, wildlife, or nearby waters or aquatic life.

3.7 Hazardous and Chemical Waste

Annual generation of hazardous or chemical waste at the ARRR facility is very small. The ARRR's principal activity, neutron radiography, requires that the industrial X-Ray film, containing small amounts of silver, is processed with automatic film processors. The primary waste stream from this process, containing small amounts of silver, is collected and stored in 55 gallon, non-corrosive drums which are removed by a hazardous waste broker within the 90 day period. The drums are stored in a non-corrosive spill tray capable of containing at least 85 gallons of spillage/leakage. The secondary waste stream, a combination of the developer/rinse water from the automatic

film processors, is collected and pumped into metal replacement "Super Cans." The effluent from the "Super Cans" discharges into the sanitary sewer system in the line that includes a sampling location.

The only other potential waste stream is generated in the Chemistry Laboratory which is not used on a regular basis. The regular sink drains into the sanitary sewer while the radioactive waste is directed into the liquid waste storage system described in Sections 2.6 and 3.4 of this report.

The ARRR facility waste water is discharged into underground pipelines and is collected and cleaned by the Contra Costa County 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 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.

Three solvents (acetone, methanol, and ethanol) are used in small amounts to remove adhesive from the aluminum trays used for the neutron radiography process. None of these solvents is used in quantities large enough to cause runoff into the sanitary sewer system. Solvent use is regulated under permit by the Bay Area Air Quality Management District.

Very small quantities of industrial waste (e.g., fluorescent light tubes, batteries, etc) are disposed of in accordance with a program administered by the CCCSD.

The generation of hazardous or chemical waste at the ARRR facility is very small and the material that is generated is handled in accordance with all federal, state, and local regulations, the environmental effects from hazardous or chemical waste generated as the result of the continued operation of the ARRR facility is negligible.

3.8 Environmental Monitoring

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. The radiation monitoring systems associated with reactor operations at the ARRR facility are provided and maintained as a means of ensuring compliance with radiation limits established under 10 CFR Part 20. Monitoring is performed at the ultimate source (i.e., the areas just above the reactor water tank) by the building gaseous effluent monitor and the building particulate sampler as described in the Updated Final Safety Analysis Report. 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, 17 locations are monitored.

3.9 Environmental Consequences of Potential Accidents

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, 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. NUREG/CR 2387 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. Furthermore, NUREG/CR 2387 states that 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. Therefore, 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).

Based on the discussion in the ARRR Updated Safety Analysis Report, any credible accident at the ARRR ranging from failure of experiments to the largest core damage and fission product release would result in doses of only a small fraction of 10 CFR Part 20 guidelines and are considered negligible with respect to the environment. Considering the low probability of any accident at the ARRR and the low consequences of an accident, continued operation of the ARRR does not pose a significant risk to operating personnel or the general public.

3.10 Long Term Effects of Facility Operation

As described in the report and the ARRR Updated Final Safety Analysis Report, operation of the ARRR facility does not have any significant effect on the terrain, vegetation, wildlife, or nearby waters or aquatic life. Operation of the ARRR does not result in any significant release of gaseous or liquid radioactive effluents and the generation of solid radioactive waste or chemical or hazardous waste is minimal. Therefore, continued operation of the ARRR facility has a minimal impact on the environment and does not pose a significant risk to operating personnel or the general public.

4.0 Cost and Benefits of Facility and Alternatives

The ARRR is one of a limited number of nuclear reactors in the United States capable of providing a neutron source for research and development and services such as neutron radiology. The unique services provided by the ARRR facility are listed in Section 1.2 of this report. Therefore, continued operation of the ARRR facility is considered beneficial.

5.0 Conclusion

Based on the discussion in the ARRR Updated Safety Analysis Report and this report, Aerotest Operations, Inc. believes that there will be no adverse impact on the environment as the result of the continued operation of the ARRR facility.

AEROTEST OPERATIONS, INC.

Aerotest Radiography and Research Reactor (ARRR)

UPDATED SAFETY ANALYSIS REPORT (USAR)

DECEMBER 2017
REVISION 1

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

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

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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 Aerotest Operations, Inc, a wholly owned subsidiary of Nuclear Labyrinth LLC. 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 on 0.9 acre 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 true. However, 31 research reactors are in operation in the United States, 16 of which are TRIGA reactors. 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.

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 3 ton capacity bridge crane that can cover the entire area. The reactor tank is embedded in the floor, extending 22 feet below and one foot above the floor surface. A [redacted] inch thick by [redacted] inch high block wall made of normal density concrete encloses the reactor area above the floor level. The top of this shield is covered with an [redacted] inch thick wooden shield.

A steel, locked 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 original 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. Later fuel elements were purchased from CERCA. 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. Currently, twenty-seven 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] inches diameter (nominal) by [redacted] inches high for aluminum clad elements and [redacted] inches 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% $\Delta k/k$. 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 to as low as reasonably achievable (ALARA). Under no circumstances will normal 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.

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1.4 COMPARISON WITH SIMILAR FACILITIES

The design of the ARRR fuel is similar to those of 15 TRIGA fueled reactors currently operating in the United States. Many 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 3 TRIGA reactors approved for steady state operation at 250 kW and an additional 2 that can be pulsed. Of the former, these are located at Argonne National Laboratory - West (constructed in 1977), University of Maryland (1974), and Reed College (1968). There are 5 TRIGA reactors approved for steady state operation between 100 and 300 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	86	61	93	60
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)	10 (diameter) x 23 (depth)	6.5 (diameter) x 12.5 (depth)	7 (diameter) x 21 (depth)	15 (l) x10 (w) x 25 (depth)
Rod speed (inches/minute)	12/20*	19/25	19	19.8

* 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.

The ARRR is normally operated for an eight hour shift five days a week. 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, amended Sept. 10, 2010) (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.

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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] inch active fuel elements into the core in addition to the [redacted] inch 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.
July 17, 2017	**License Amendment No. 5 and Tech Spec Change No. 9:** Transfers license to Nuclear Labyrinth, LLC.

Figure 1-1
ARRR Facility Layout

[redacted]

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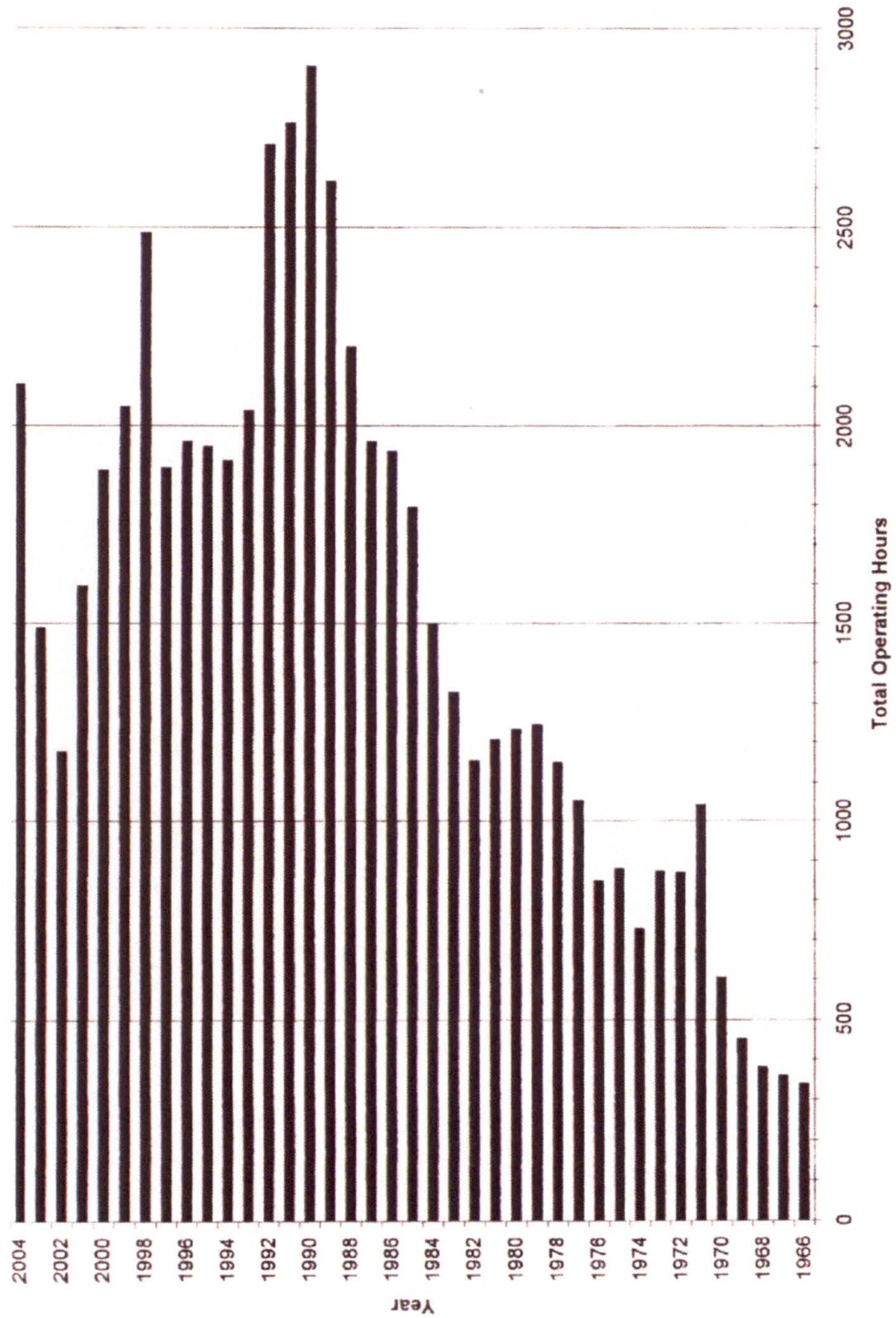
Figure 1-2
ARRR Annual Operating Hours
(1966 – 2004)



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.

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2.0 SITE CHARACTERISTICS

2.1 GEOGRAPHY AND DEMOGRAPHY

The ARRR facility is located inside a steel perimeter fence with a locked 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 50 feet. [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 35,272 which is the US Census 2010 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

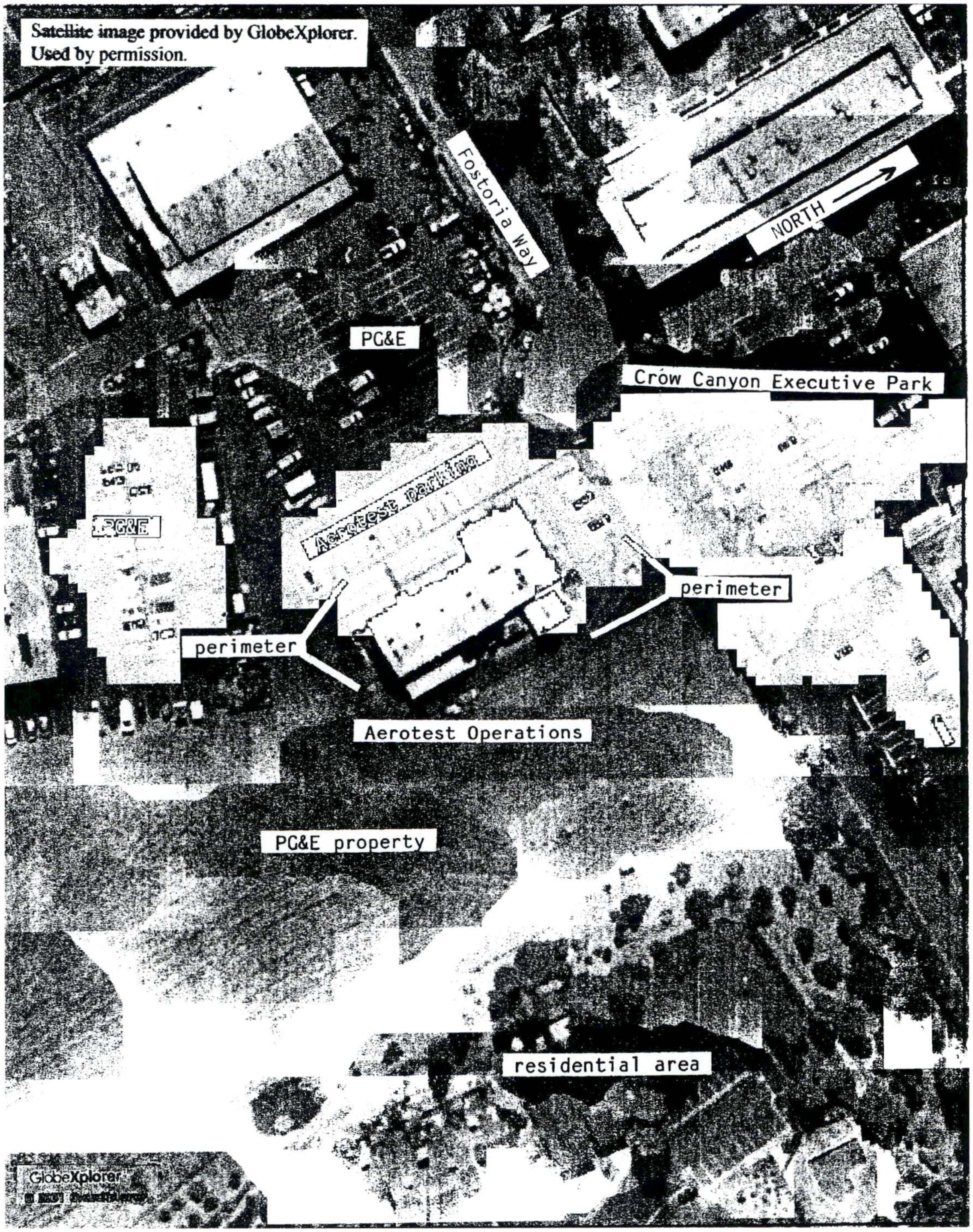


Figure 2.2
Street Map of the Area Surrounding the ARRR Facility

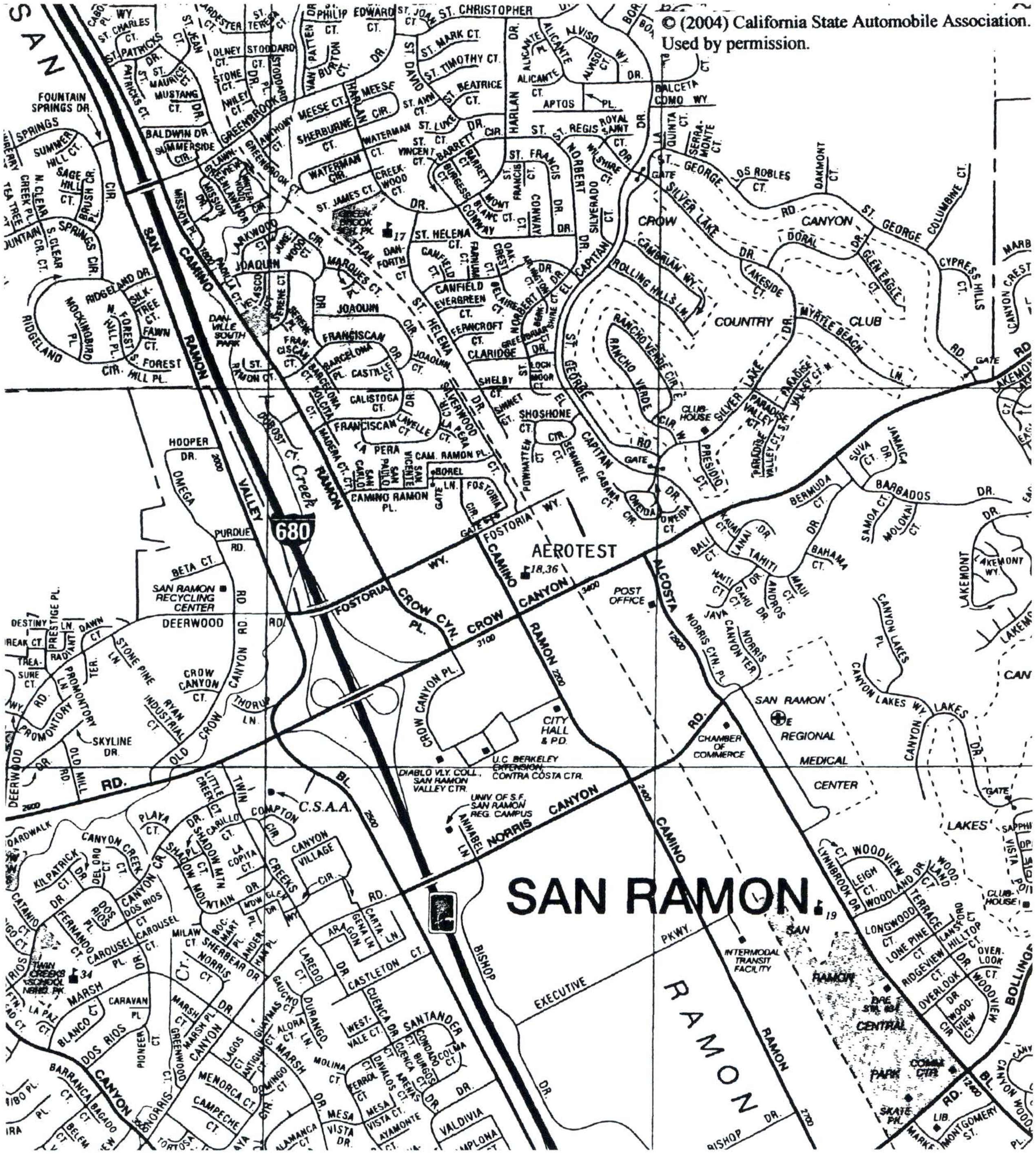
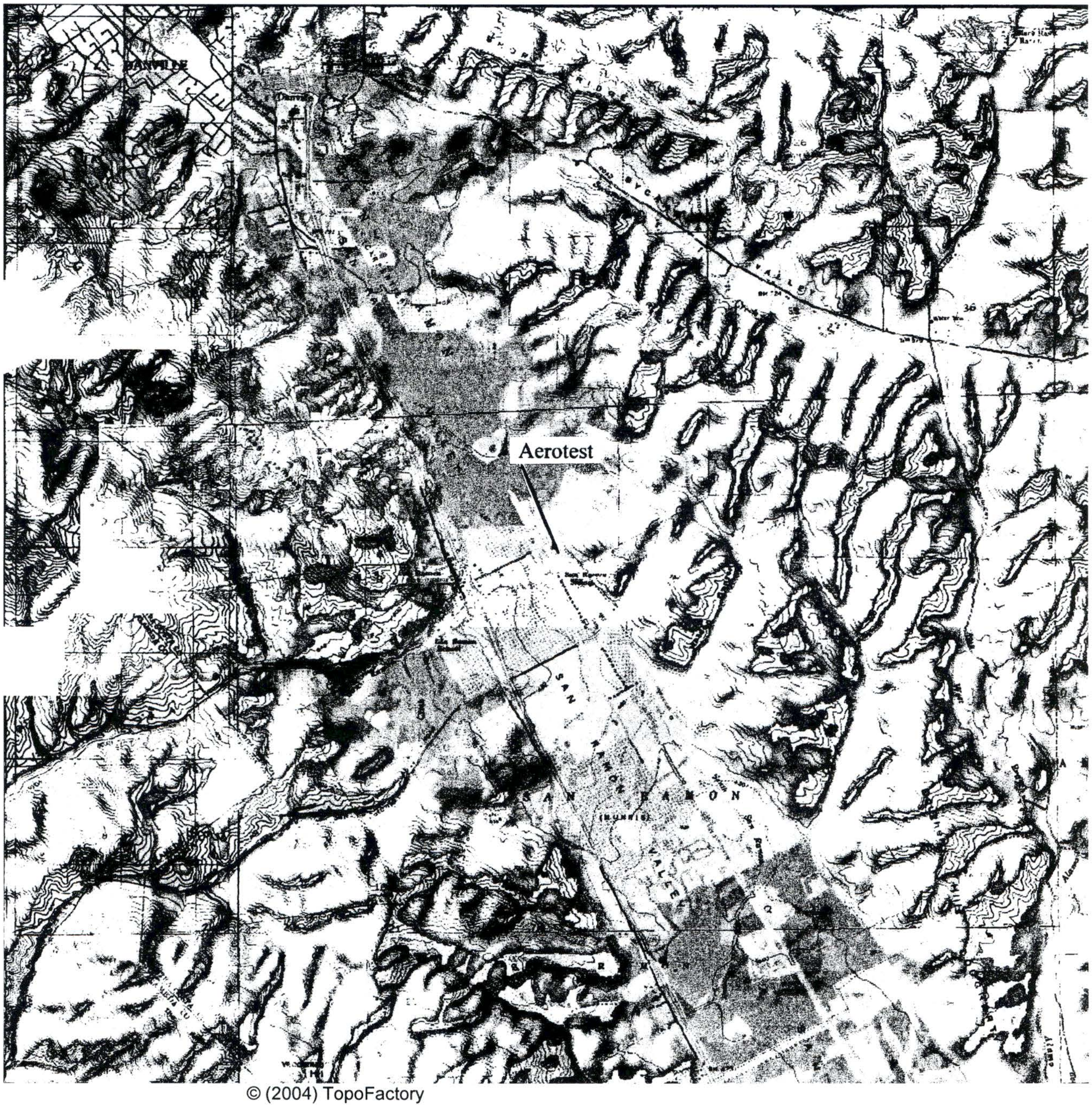


Figure 2.3
Topographic Map of the Area Surrounding the ARRR Facility



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 Crow Canyon Medical Center, Danville Materials, and various sub-rentals in those buildings. 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 2000 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 sprinkler system covers the entire building and portable fire extinguishers are strategically located throughout the building. An alarm system is installed to detect unauthorized entry into the building.

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 22 ft below the surface and one foot above the floor. A [redacted] inch thick by [redacted] inch 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.

Six fuel storage pits, each [redacted] feet deep and [redacted] inches in diameter, are located in the floor in the [redacted] of the reactor room. The fuel storage pits can be individually flooded and used for storage of fuel elements, if necessary. The fuel storage pits can also be used for the storage of any highly radioactive materials. The fuel storage pits are closed with [redacted] foot thick concrete shield plugs.

A 3-ton 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]

REDACTED FOR PUBLIC RELEASE

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.

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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) In-core Irradiation Capsules;

(8) Large Component Irradiation Box;

(9) Pneumatic Transfer Facility; and

(10) Beam Port (Not Installed).

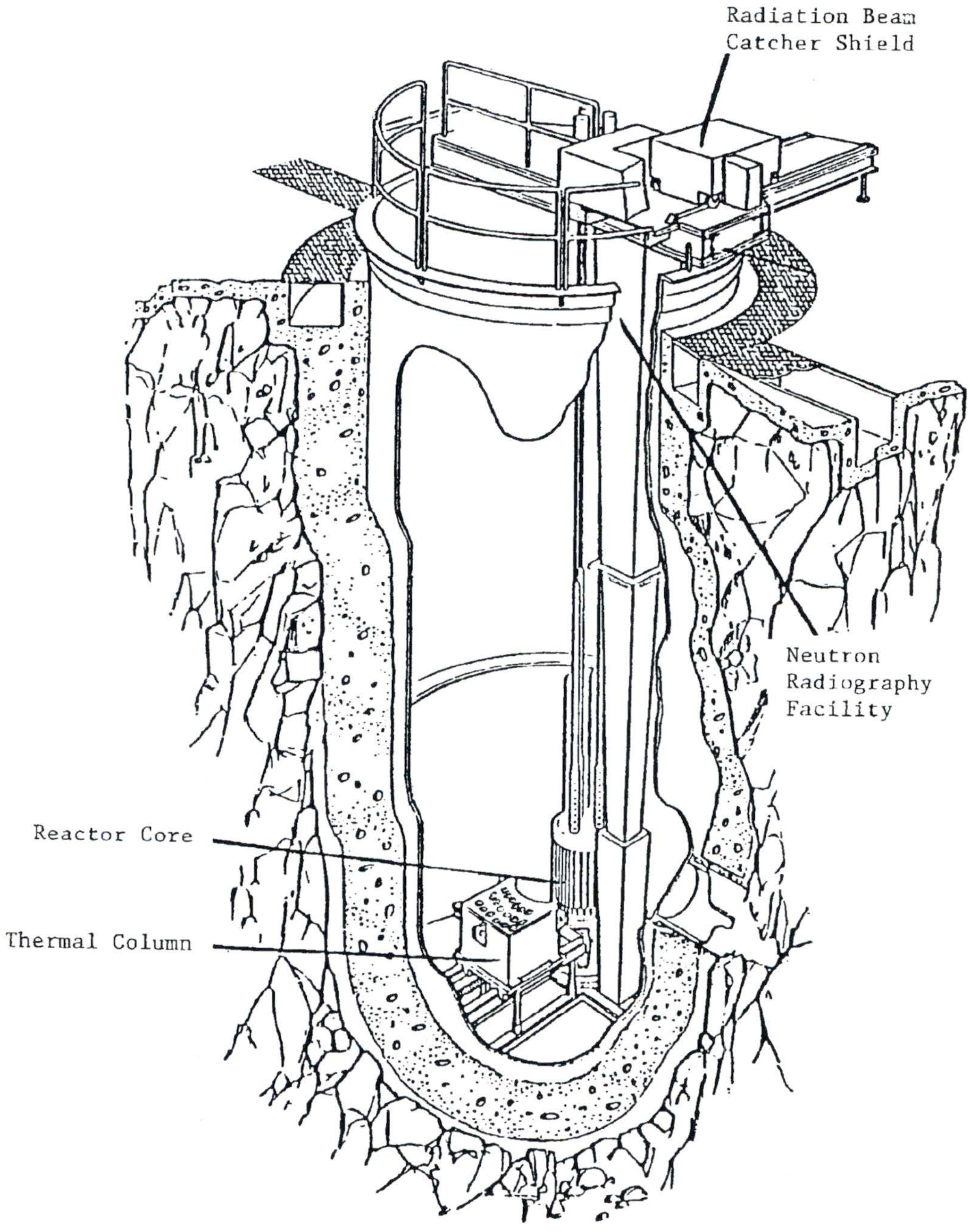
4.1.2 TRIGA Reactors - Intrinsic Safety Features

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.

Figure 4.1-1
Cutaway Drawing of the ARRR Reactor



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 127 grid positions (126 in six 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 63 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. The ARRR currently consists of 55 aluminum clad, 27 stainless steel clad and 40 graphite elements.

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: 0.030 inch thick Stainless steel elements: 0.020 inch 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] grams Stainless steel elements: [redacted] grams
Active fuel length	Aluminum elements: [redacted] inches Stainless steel elements: [redacted] inches
Moderator:	Zirconium hydride and water
Reflector:	Demineralized water: (∞) Graphite : sides 1.5 inches top & bottom: 4 inches (aluminum) top & bottom: 3.5 inches (stainless)
Core (active) Dimensions:	Diameter: [redacted] inches Height: [redacted] inches (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 a pintle 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 5 feet high and has an inside diameter of about 22 inches. The base of the structure is bolted to T-beams emplaced on the bottom of the reactor tank. The center of the [redacted] inch active fuel region is approximately 3 ft above the bottom of the reactor pool. The region below the core is accessible through large openings in the lower portion of the shroud.

The elements are supported and spaced by upper and lower grid plates made of 6061 aluminum. The upper grid plate is fixed to the shroud by a combination of bolts and clamps. The lower grid plate is bolted to the shroud. As stated above, the upper and lower grid have a total of 127 grid positions (126 in six 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] inches in diameter in the aluminum upper grid plate. The upper grid plate has a thickness of 5/8 inches. 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.

The upper grid plate includes a hexagonal section in the center that can be removed for the insertion of specimens about 4.4 in. in diameter into the region of highest flux. This requires prior relocation of the central fuel element and the six elements from the B-ring to the outer portion of the core. Two triangular sections, each of which encompasses one D-ring and two E-ring holes, are cut out of the upper grid plate. When fuel elements are placed in these locations, their lateral support is provided by special aluminum pieces. With the fuel elements removed, there is room for inserting specimens up to 2.35 inches in diameter.

The bottom grid plate holes receive the lower-end fixtures of the fuel elements. The bottom grid plate has a thickness of 3/4 inches. 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 x 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 outermost positions in the grid array. 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 vertical center 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.

The neutron source holder can be withdrawn from the core for tests or storage by means of a stainless steel cable attached to the holder and connected to a structural member at the top of the reactor tank.

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 20 inches (51 cm) long with outside diameters of 7/8 inches (2.22 cm) for the regulating rod and 1.25 inches (3.2 cm) 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 15 inches.

Each of the three control rods can be inserted into any of the 127 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.

Typical 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 127 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 1.495 inches (3.80 cm). 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 2 inches 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 and Interlocks

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. Control rod interlocks provide controls as required by Technical Specifications.

(i) Am interlock prevents 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.

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

(iii) An interlock prevents simultaneous withdrawal of the shim rod and the regulating rod.

(iv) Limit switches 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

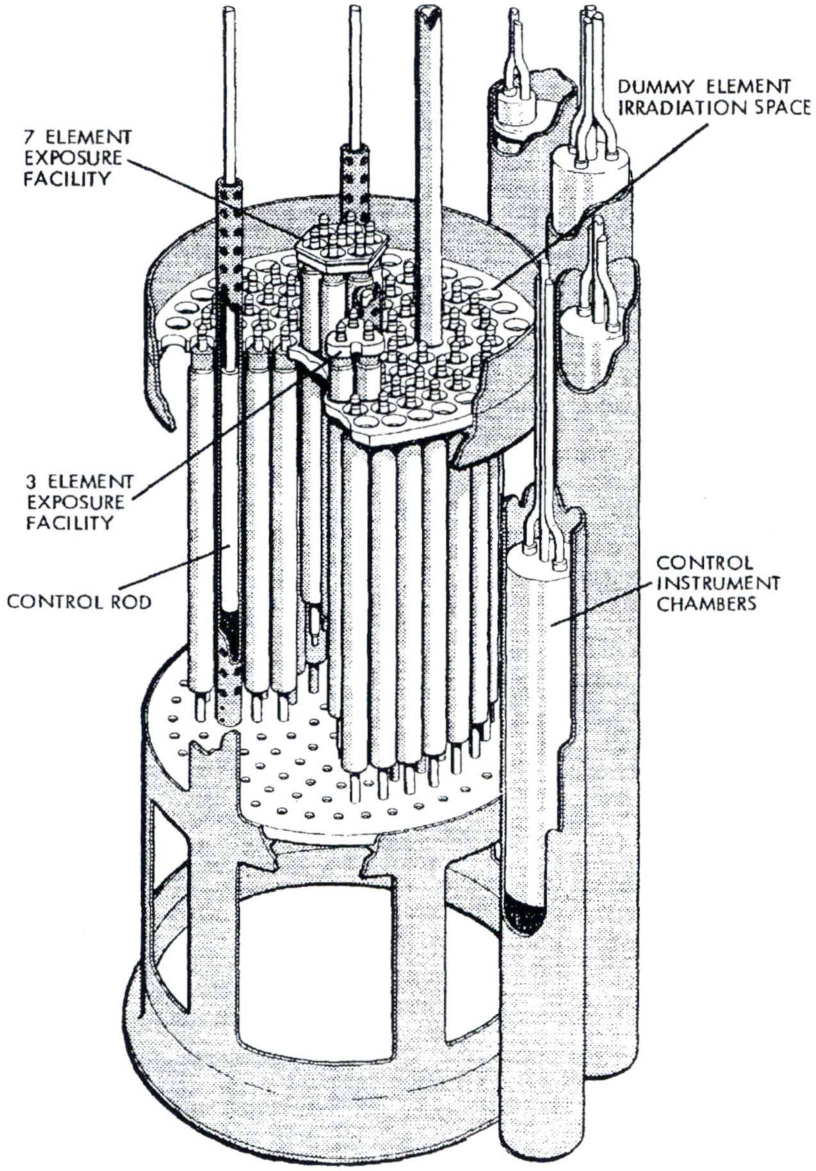


Figure 4.2-2
Typical ARRR Core Arrangement

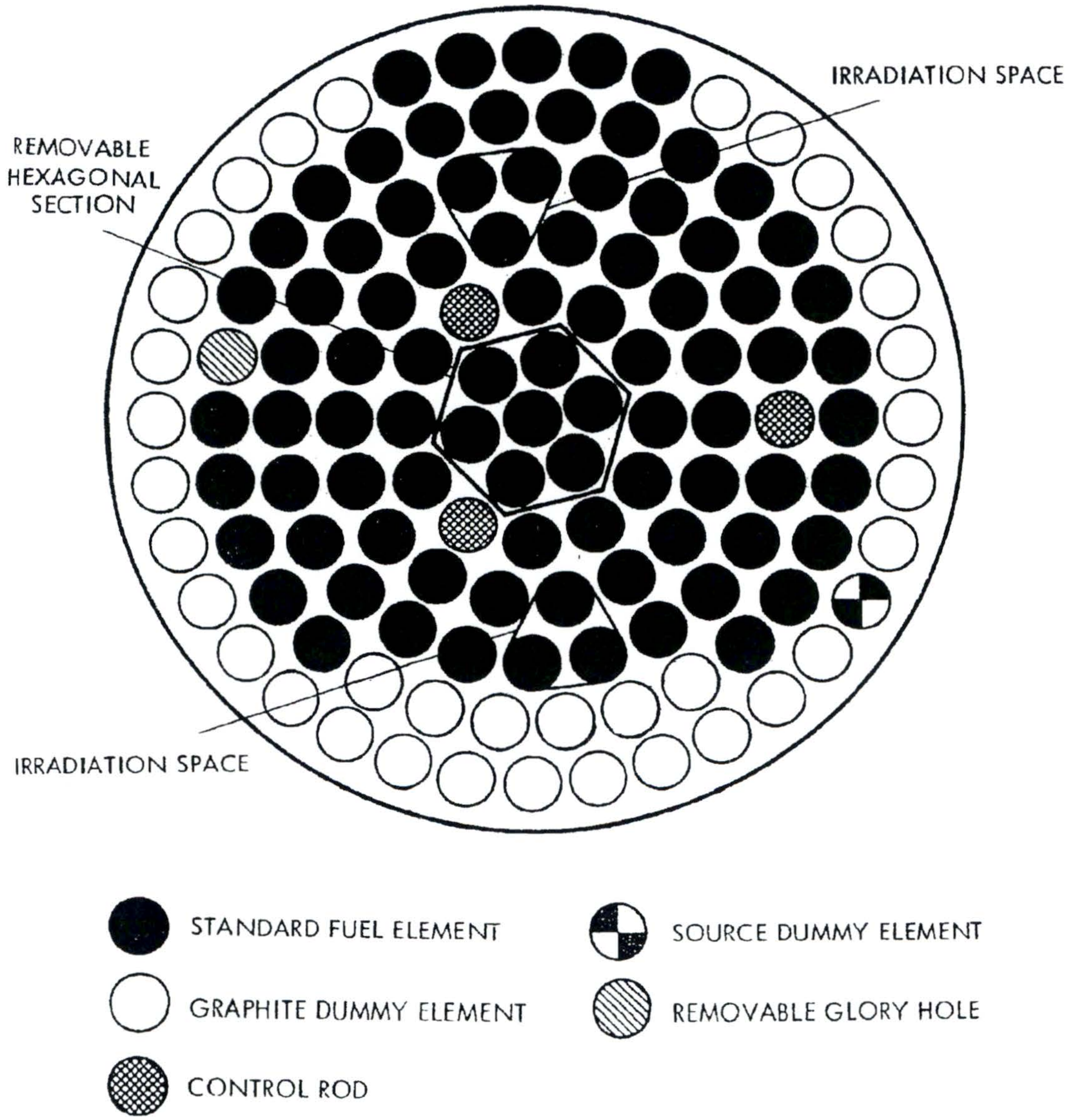


Figure 4.2-3
ARRR Aluminum Clad Fuel Element

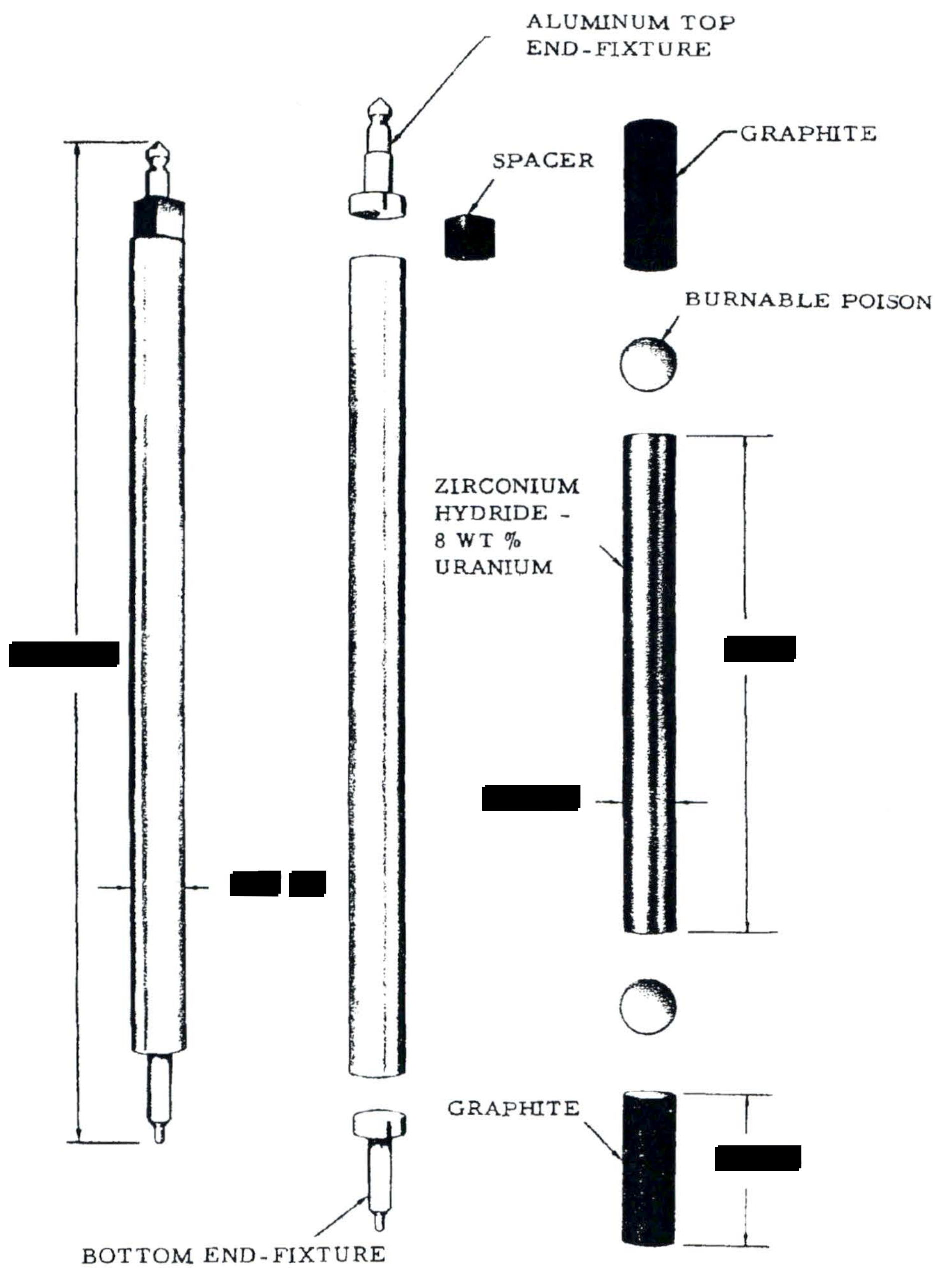


Figure 4.2-4
ARRR Stainless Steel Clad Fuel Element

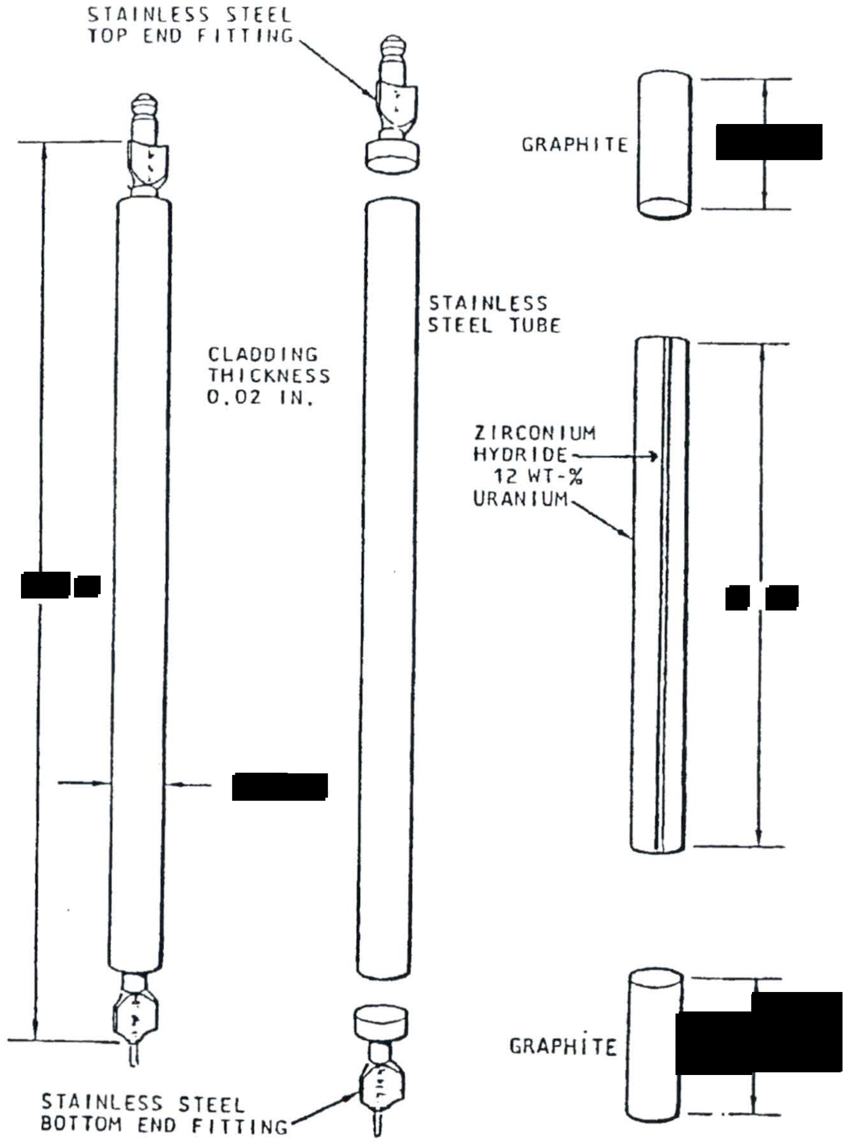


Figure 4.2-5
ARRR Core and Support Structure

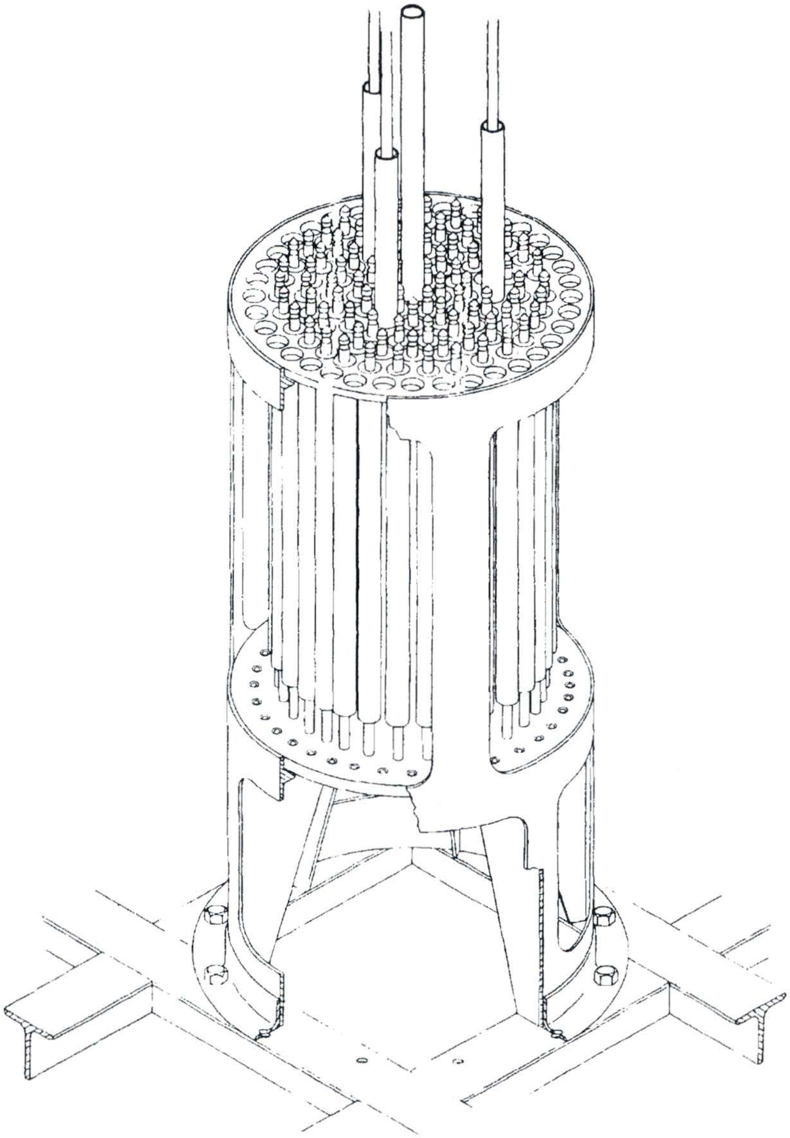


Figure 4.2-6
ARRR Neutron Source Holder

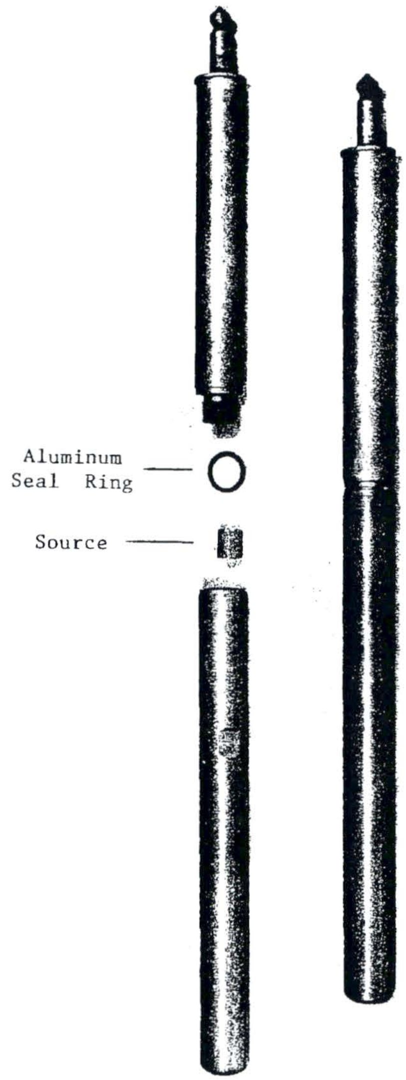


Figure 4.2-7
ARRR Control Rod Drive Connecting Rod and Barrel Assembly

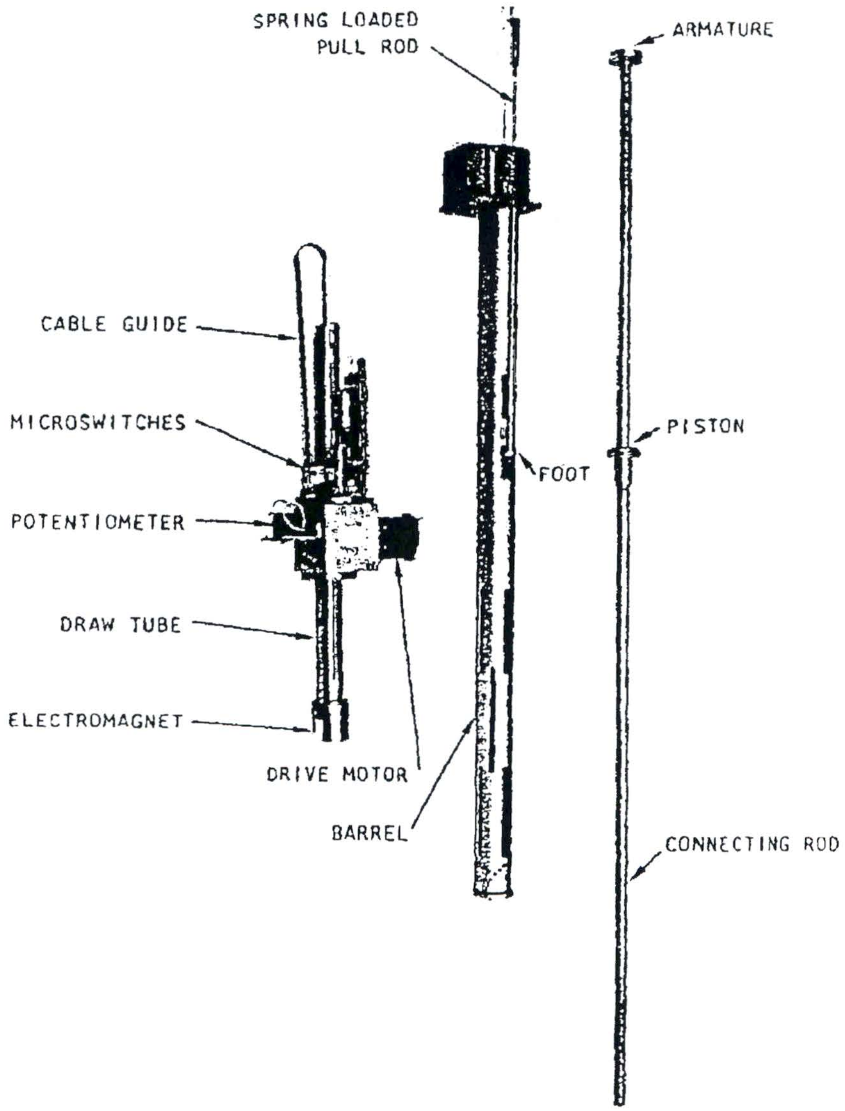
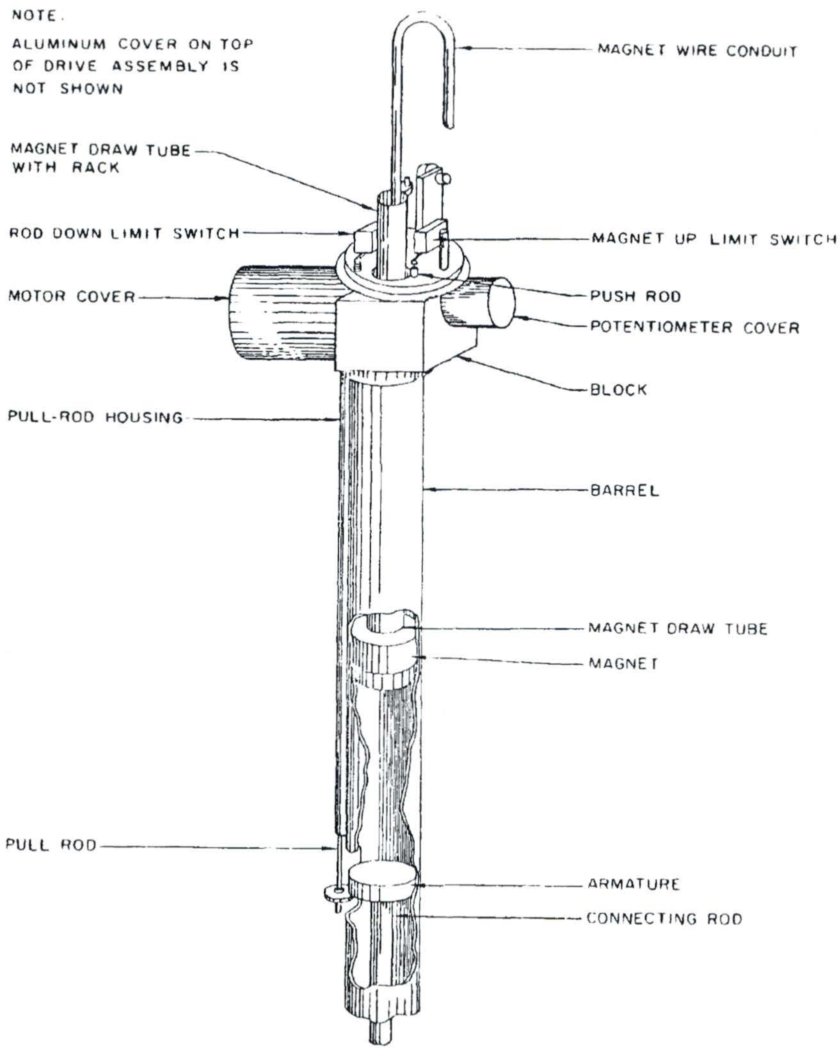


Figure 4.2-8
ARRR Control Rod Drive Assembly



4.3 REACTOR TANK

4.3.1 Reactor Tank Design

The reactor tank is an aluminum cylinder 10 feet in diameter and 23 feet deep that extends 22 feet 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 24 inch outside diameter pipe about 13 ft long 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 2 feet wide by 2 feet 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 biological shield, and for reactor core cooling. Technical Specifications require that tank level is >16 feet above the top of the active core. 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 17 ft 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 in the wall cabinet east of the reactor shielding.

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 located in the reactor tank 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 which gives access to the control rods and instrument tubes for maintenance. 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 10 inch structural steel I-beams. Two of these beams support the 3/4 inch-thick aluminum control rod drive mounting plate. This plate locates the control rods and glory hole. The reactor bridge is secured at on[redacted] 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 400 lb, in addition to the normal weight of the control rods and equipment, will result in a maximum I-beam deflection of 0.0025 inches. 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] inches thick and wooden (fir) beams [redacted] inches 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 63 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 2010, the ARRR consisted of 55 aluminum clad elements and 27 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:

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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.

REDACTED FOR PUBLIC RELEASE

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 10 feet in diameter and 23 feet 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 22 feet (i.e., 17 feet above the top of active fuel), the reactor tank contains about 13,000 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. 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 just one foot 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 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.

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 that states that “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 Water Temperature High signal. The 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 22 feet), 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 three inch 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. Additionally, the sump is supplied with bleach 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 <u>temperature switch</u> 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 <u>radiation monitor</u> 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 heat exchanger building 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 10 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. The rotary vane flow switch described above 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, ± 3 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 6 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)

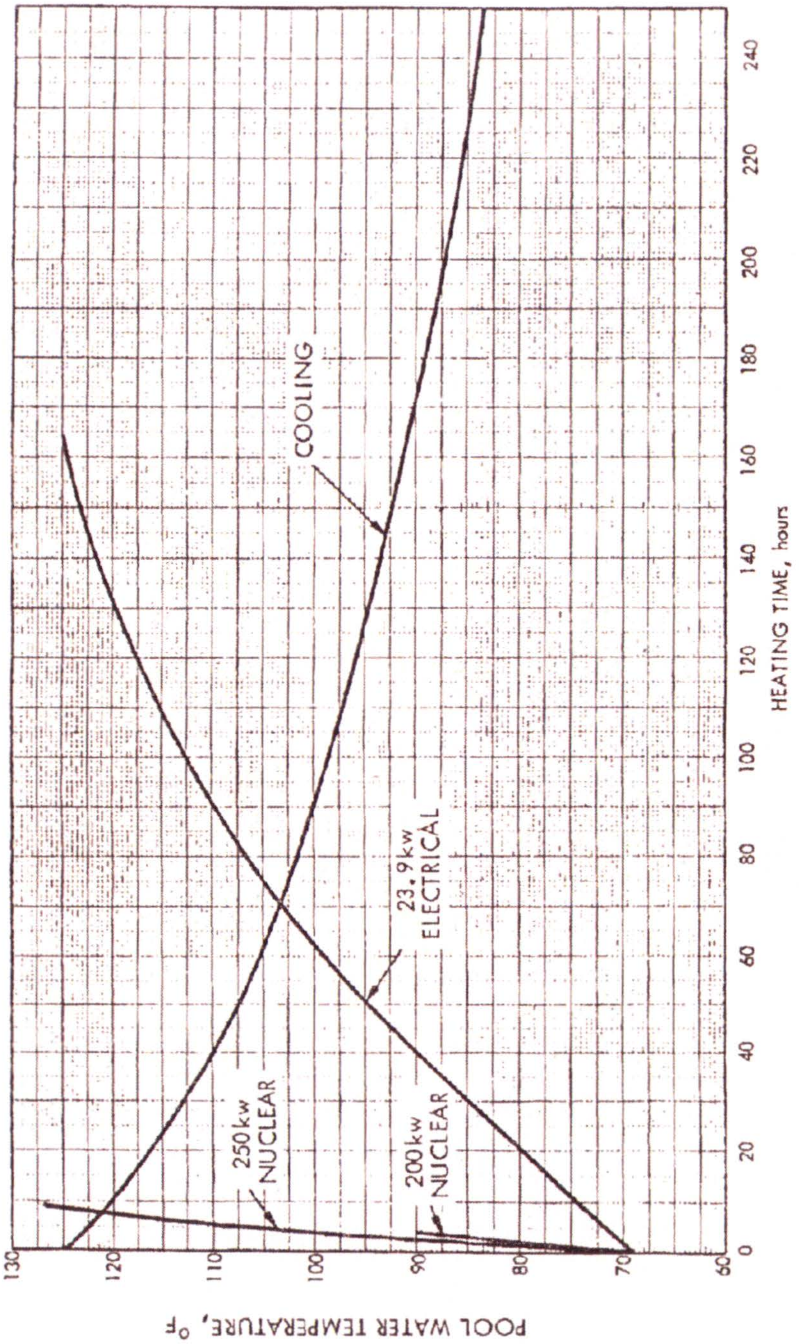


Figure 5-2
Primary and Secondary Cooling Loops

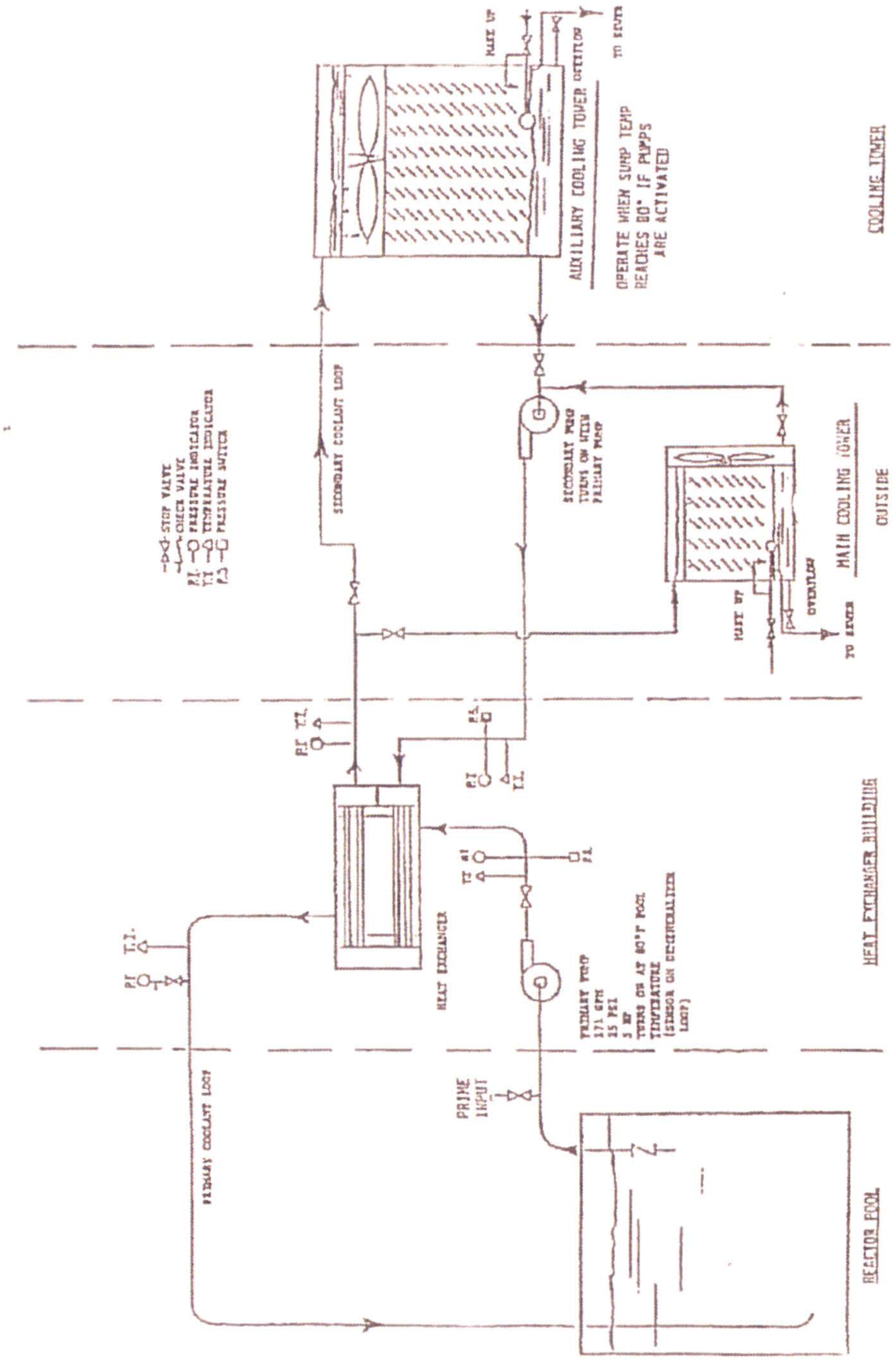
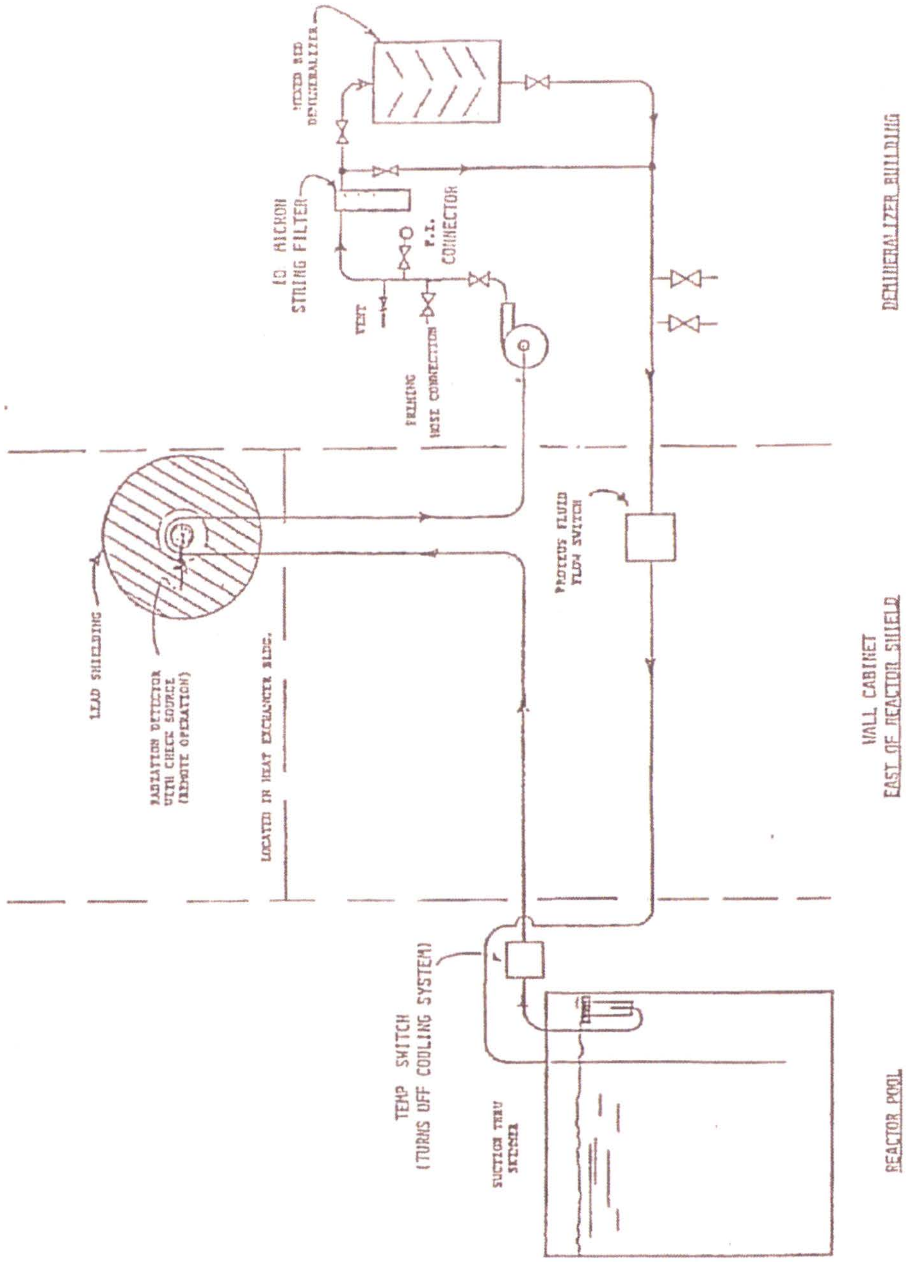


Figure 5-3
Demineralizer System



5.7 REFERENCES

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

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

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 leakage 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 manually at one location 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 13,000 gallons 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 REACTOR CONTROL AND SAFETY 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:

<u>Rod Control System</u> 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.

<u>Reactor Safety Channels (RSC)</u> is used to monitor important reactor operating conditions (e.g., reactor power level, reactor period) and initiate automatic reactor shutdown via the rod control system if pre-set limits are exceeded or required conditions are not met.

<u>Reactor Auxiliary Safety Systems</u> (RAS) are used to monitor primary coolant conductivity and reactor water radioactivity and provide audible and visual indication to the operator if pre-set limits are exceeded.

<u>Criticality Alarm, Radiation and Radioactive Gaseous Effluent Monitoring Systems</u> 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 RSC 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 RSC and RAS. Channel 1 is a solid state system. The instruments normally monitored by an operator or that may require operator action are located in a low-silhouette, three-panel

desk console in the reactor control room. Additional power supplies and process instrumentation chassis are located in a separate instrumentation rack placed near the control console in the control room.

Technical Specification Table 3.2-1, Reactor Safety Channels, Table 3.2-2, Reactor Auxiliary Safety Systems and Table 3.5-1, Criticality Alarm, and Area and Effluent Radiation Monitoring Instrumentation 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, single 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.

Six lighted pushbuttons and three lighted double pushbuttons on the operator console allow the operator to control the CRDs and provide indication of the status of the CRD and associated control rod and allow the reactor operator to scram the reactor. Limit switches (micro-switches), mounted on the CRD assembly, 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. The nine lighted pushbuttons are arranged in three vertical rows, one row for each rod drive. 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 double pushbuttons indicate current to the CRD magnet and that the magnet is in contact with the rod armature. When the double pushbuttons 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

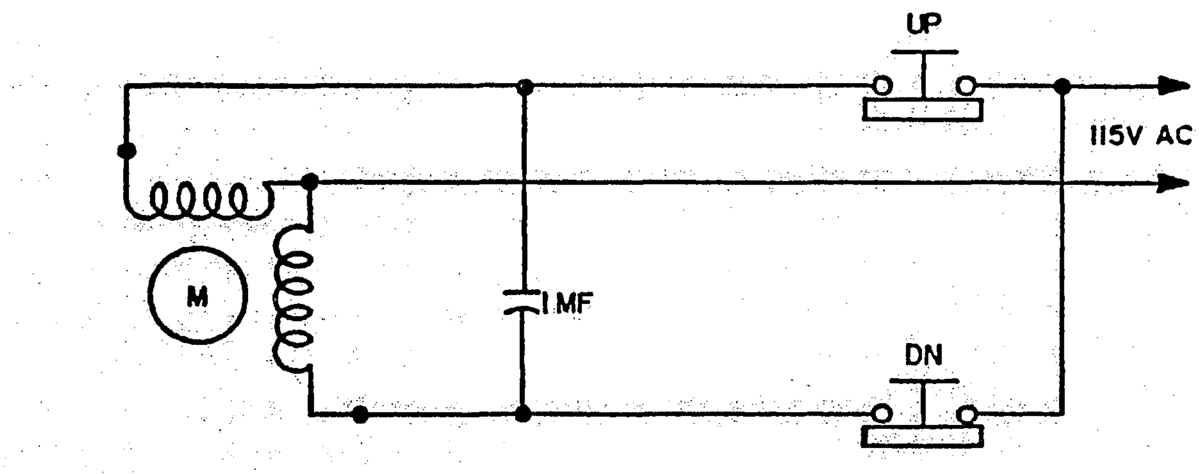
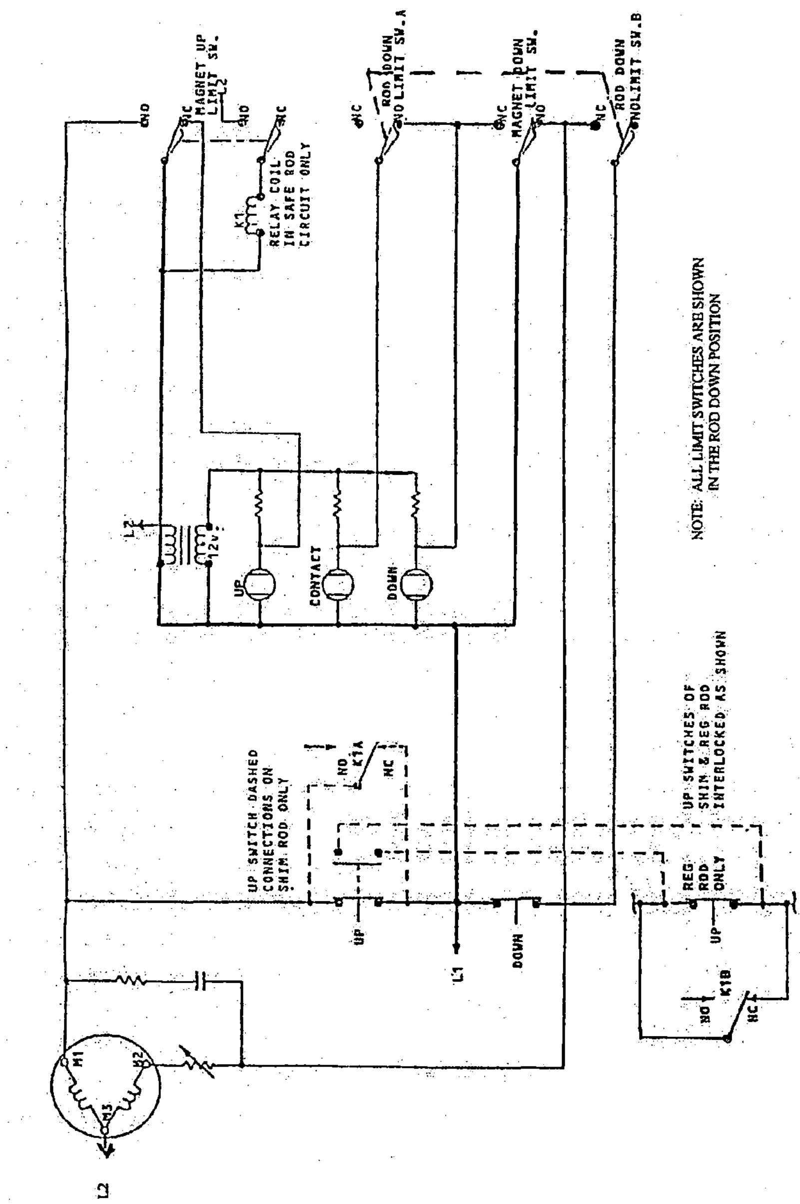


Figure 7.1-2
Control Rod Drive Motor Control Circuit
with the Drive Limit and Indicating Circuits



7.2 REACTOR SAFETY CHANNELS (RSC)

The purpose of the RSC 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 RSC 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 switch 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.

Most of the reactor safety channels 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 automatically or manually reset after the scram condition has been cleared.

All annunciators and alarms are combined within two chassis. The indicators, scram reset switches, lamp test, and horn-acknowledge switch are on the front panel for convenient operator control. The bi-stable trip cards and associated relays are mounted at the rear of the console in a closed drawer.

Requirements for periodic channel checks, channel tests, and channel calibrations for each of the reactor safety channels are specified in the Technical Specifications.

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

Channel 1 Source Range Counts.
Channel 2 logarithmic Power
Channel 3 Linear Power
Channel 4 Linear Power
Low Reactor Tank Water Level.
Seismic Disturbance.
High Reactor Tank Water Temperature.
Manual Scram Bar.
Master Key Switch.

7.2.1 Source Counts -Channel 1:

Source Counts- Channel 1, is used to measure neutron levels and provides indication of both number of neutron counts and count rate (reactor period).

Channel 1 detector 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.

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 reactor safety channel scram functions associated with channel 1: short period; and, low source level.

Short Period 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.

Low Source Level 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 reactor safety 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 conservatively 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 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 and low source level, 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 1×10^{-10} amps. On decreasing power, channel 1 is automatically restored to service when channel 2 indicates approximately 9×10^{-11} amps to ensure channel 1 is operating before channel 2 reaches the lower end of its range.

7.2.2 Logarithmic Power- Channel 2:

Logarithmic Power- 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. A channel 2 bi-stable trip card is used to automatically disable the scrams for short period and low source level in channel 1 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 provided by channels 3 and 4. This feature ensures that low neutron flux levels are monitored on channel 2 until neutron flux is above channel 3 and 4 scram set points.

Channel 2 Detector is a compensated ionization chamber. A compensated ionization chamber is used to measure neutron flux in the low and medium power ranges by discriminating between neutron and gamma radiation and has 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 4×10^{-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 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; loss of instrument power and, Low Detector Voltage.

Channel 2 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.

Channel 2 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).

Channel 2 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 safety channel 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 and Low Detector Voltage 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 more conservative than the LSSS specified in the Technical Specifications.

7.2.3 Linear Power-Channel 3

Linear Power-Channel 3 is used to measure approximately 30 watts to 300 KWs 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 reactor safety scram functions associated with channel 3: High Linear Power; Low Linear Power; and Low Detector Voltage.

Channel 3 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 80% of full scale.

Channel 3 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.

Channel 3 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 reactor safety function ensures that low detector voltage is a fail-safe condition.

Bi-stable trip cards associated with high and low Detector Voltage change the 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 Linear Power- Channel 4

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 levels when operating near the source range.

Channel 4 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 Channel 3.

Channel 4 provides a reactor scram signal before reactor power falls below 5% of full scale on each range. . This reactor scram function operates identically to Channel 3.

Channel 4 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 Channel 3.

7.2.5 Reactor Tank Water Level

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 by directly removing power from the control rod magnets. 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

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 Modified Mercalli scale X.

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 Reactor Tank Water Temperature

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

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 Master Key Switch

The Master Key Switch is a two position, key locked switch that controls the power supply used to energize the scram bus (i.e., the CRD electromagnets). The Master Key Switch must be in the "ON" position to energize the scram bus and control rod electromagnets.

Figure 7.2-1
Block Diagram of the Nuclear Instrumentation

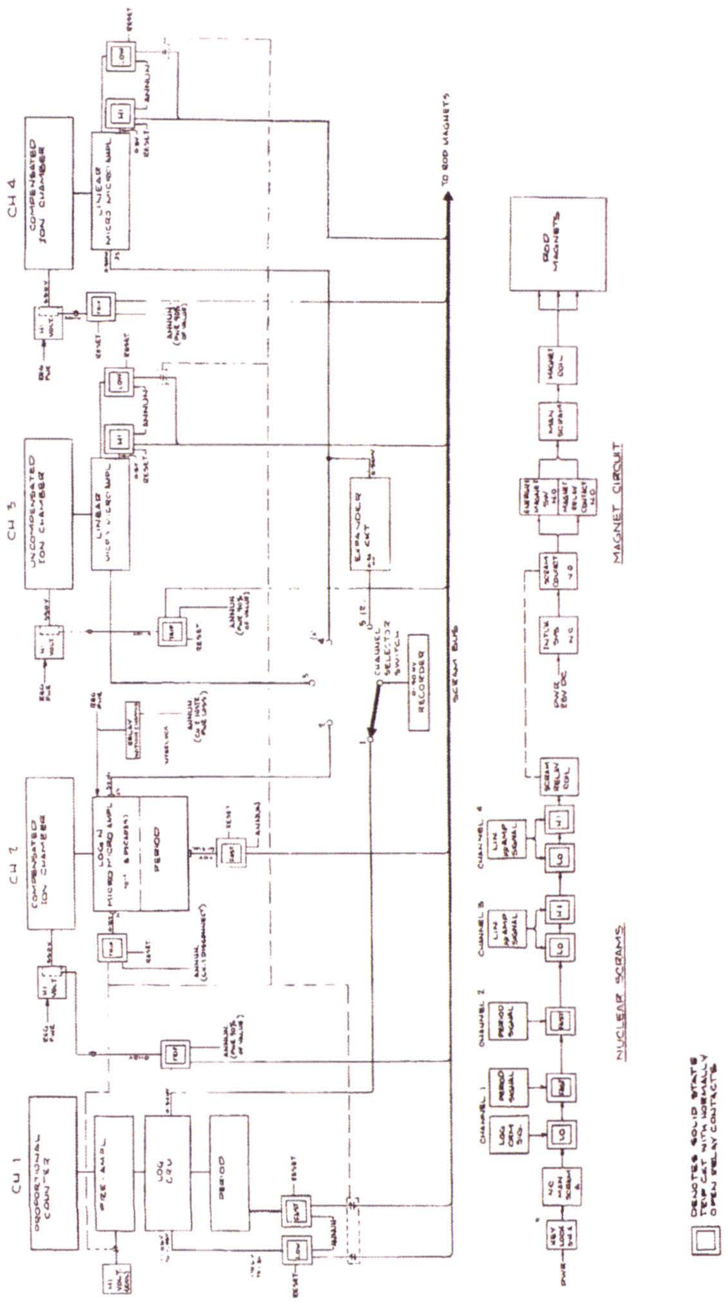


Figure 7.2-2
Logic Diagram of the Reactor Protection System

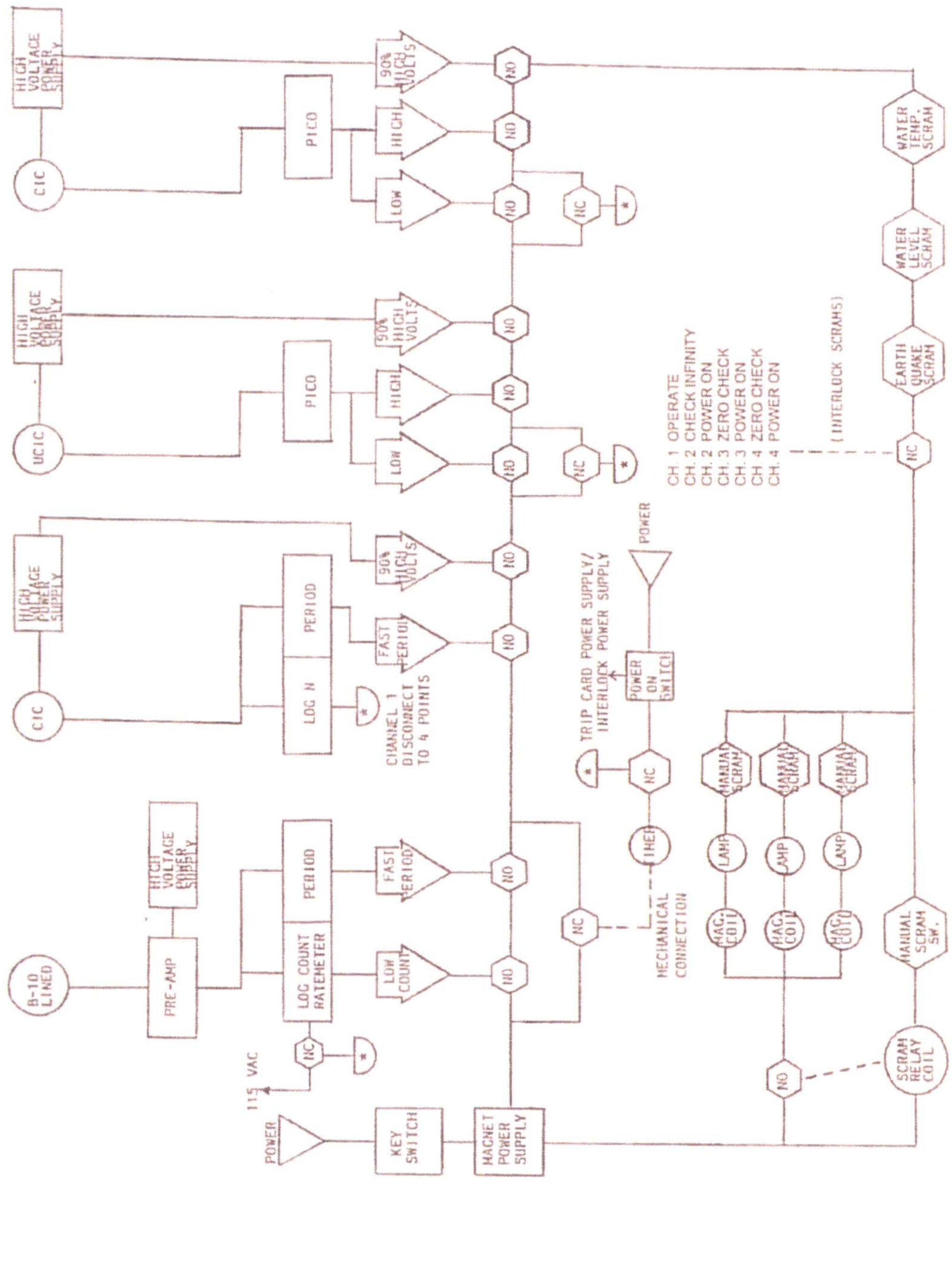


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Figure 7.2-3
Schematic Diagram of the Scram Bus and CRD Electro Magnet Circuit

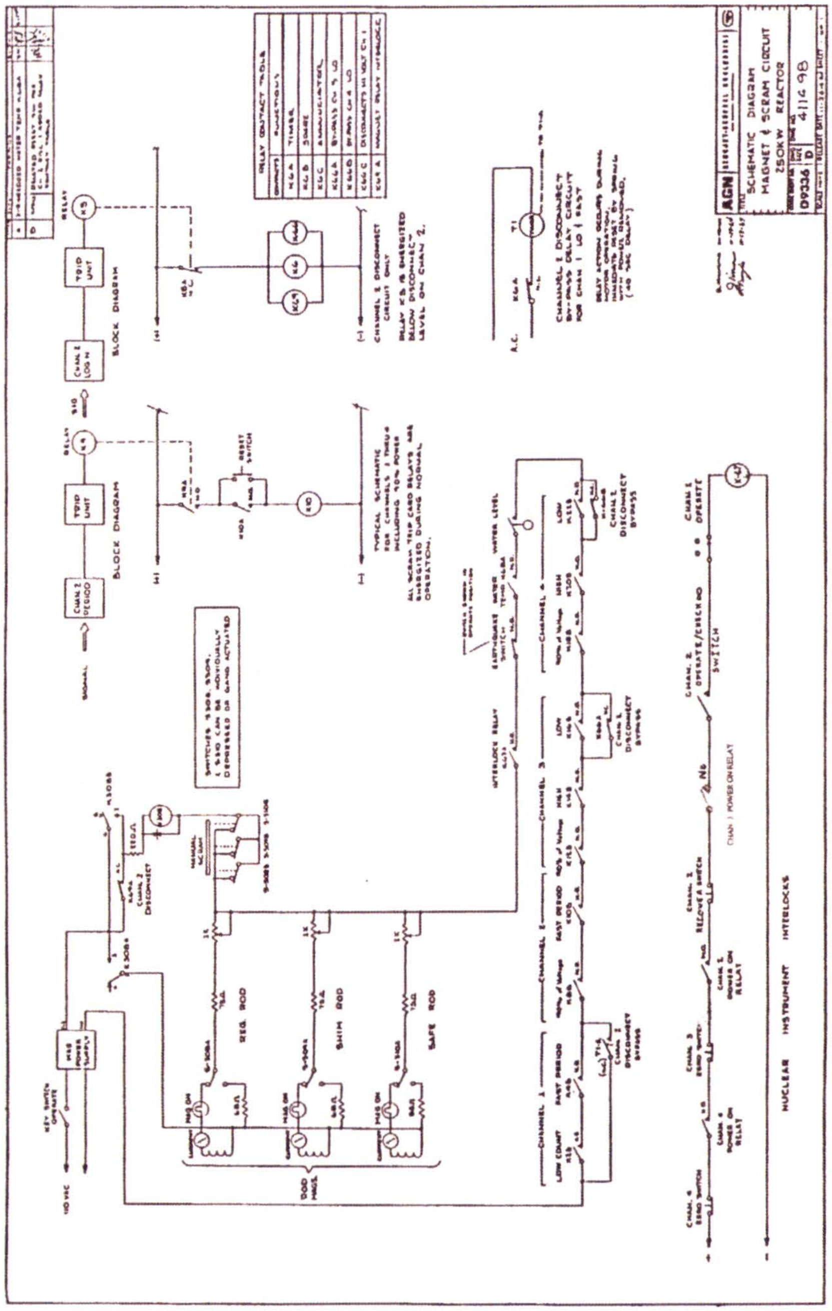
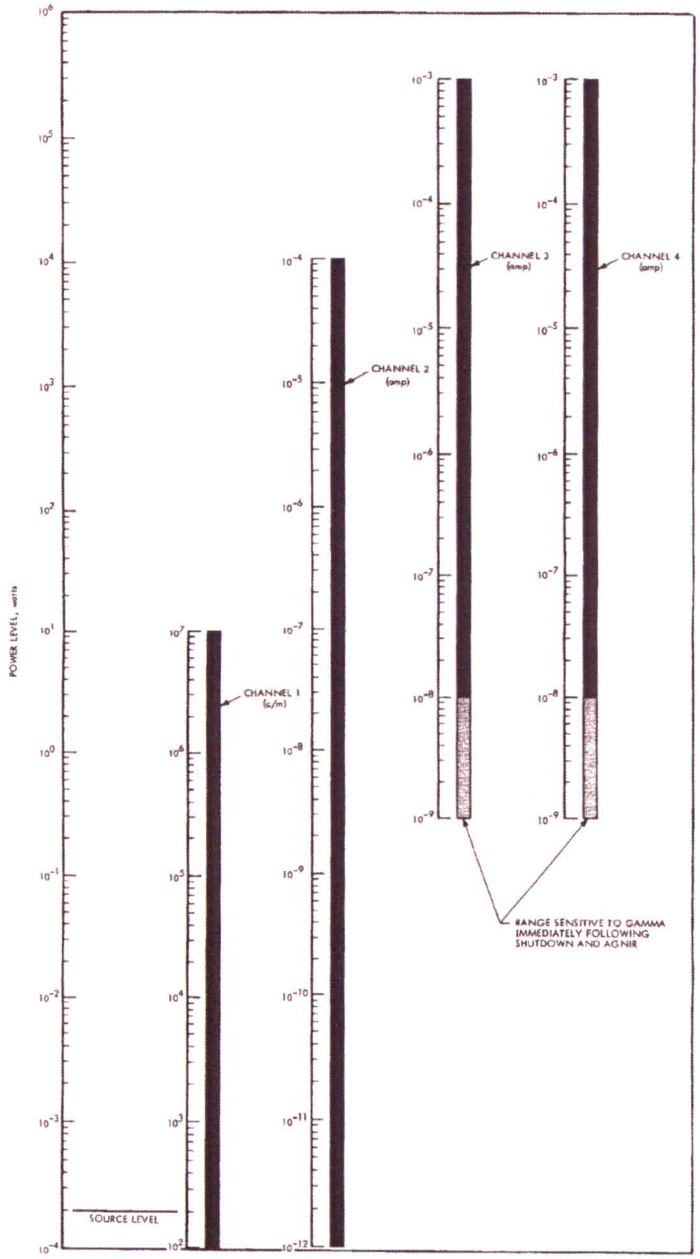


Figure 7.2-4
Normal Operating Range of the 4 Neutron Flux Detectors



7.3 REACTOR AUXILIARY SAFETY SYSTEMS (RAS)

The Reactor Monitoring systems (RMS) consist of sensing devices and associated circuits which (in most cases) 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 RAS functions are specified in the Technical Specifications.

7.3.1 Bridge Crane Location

The bridge crane is normally stored approximately 18 feet north and above the pool tank and is maintained in the storage position when not in use. Bridge crane location is monitored by a position switch that actuates visual and audible alarms on the reactor operator console when the 3-ton bridge crane is moved off the storage position. The bridge crane is designed for lateral accelerations in excess of 1.0 g. 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 bridge 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 by attaching multipliers to the instrumentation. 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. In addition to the local alarm, this system annunciates at a commercial monitoring company. 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.

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 13,000 gallon 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. In addition to the local alarm, this system annunciates at a commercial monitoring company. 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.

The Criticality and Area Radiation Monitoring alarm system uses 5 "D" NiCd cells for auxiliary power. This auxiliary power supply is held at full charge by a low voltage rectifier circuit. The auxiliary power supply capacity is 10 hours under high limit alarm conditions and 90 hours 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. This site security alarm [redacted]. The battery recharges continuously when electrical power is available.

The ARRR reactor building is also equipped with 14 battery-powered lanterns that activate when the building power fails to allow safe egress from any room in 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: <u>confinement</u> of airborne radioactive material to the reactor high bay area; and, <u>exclusion</u> of airborne radioactive material from inhabited areas, especially the control room.

<u>Confinement</u> 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: four doors with automatic hydraulic closers; one normally closed spring-loaded door; one normally open door that leads to the setup room; and, one normally closed sliding window to the control room. By procedure, the window and all doors, except for the double doors to the setup room, are required to remain closed during reactor operation except as needed for momentary passage. In the event of a high radiation alarm, the double doors to the setup room are manually closed.

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. 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. This feature is tested quarterly.

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 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. 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 and 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]

REDACTED FOR PUBLIC RELEASE

9.2 FUEL STORAGE

9.2.1 Fuel Storage in the Reactor Tank

A fuel element storage rack with positions for 21 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

Six 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] inches in diameter and [redacted] ft 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 500 curie source located 2 ft above the bottom of the hole will not exceed 2 mrem/hour at the floor surface.

Technical Specifications allow a maximum of 19 fuel elements (700 grams of U-235) in each of the six 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 must be secured with a lock and chain except during fuel transfer operations involving the associated pit. No fuel elements are currently being stored in the fuel storage pits.

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.

9.2.4 Fuel Storage Safe

New (unirradiated) fuel elements are stored in a combination lock Mosler safe.

Figure 9.2-1
Typical Fuel Element Storage Pit

[redacted]

9.3 FUEL HANDLING TOOLS

Although a large number of special tools are required for specific operations in the reactor core, two types of fuel handling tools and the control rod guide locking tool are considered vital for ARRR operation.

9.3.1 Socket Wrench Tool (Control Rod Guide Tubes)

[redacted]

9.3.2 Fuel-Handling Tools: Pneumatic and Mechanical

[redacted]

[redacted]

[redacted]

9.3.3 Fuel-Element Characterization Rig,

The apparatus shown in Figure 9.3-3 is the Fuel Element Characterization Rig. It has a central support fixture and rotator cup that will hold the element for visual inspection. The fuel element is rotated 360 degrees to inspect its entire physical condition. A metric-scale next to it allows for physical measurement of the element's length. This is useful for determining the elongation of the element. The tube (far left) is the go/no-go for the sagitta test or element bow. The far right tube is used to assist in encapsulating a breached elements in a stainless steel can. The rotator cup can be exchanged for a fixture that emulates the hole-size found in the lower core plate. This allows for a seat test to be performed on the lower pin of the element. The rig also contains a tube close to the element for insertion of a radiation detector.

Figure 9.3-1
Socket Wrench for Installation and Removal
of Control Rod Guide Tubes

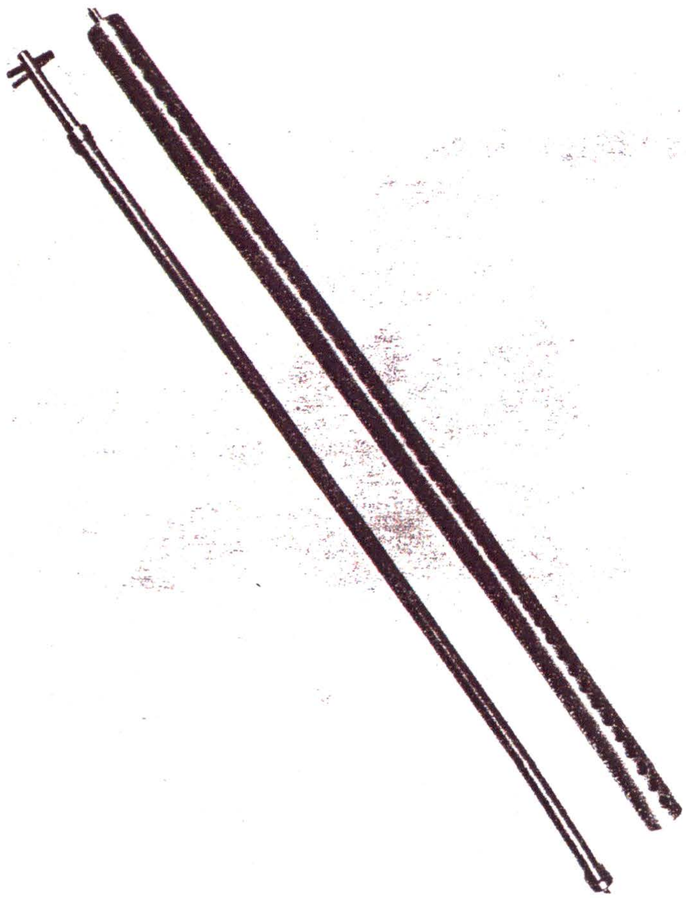


Figure 9.3-2
Pneumatic Fuel-Handling Tool

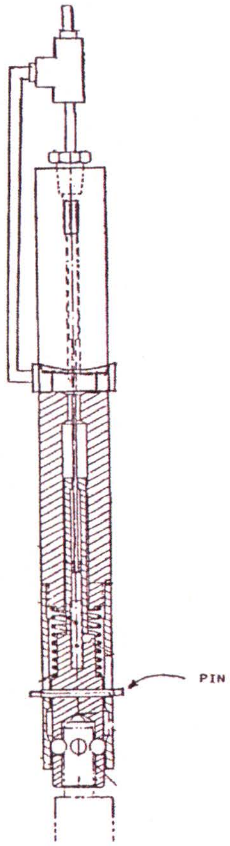
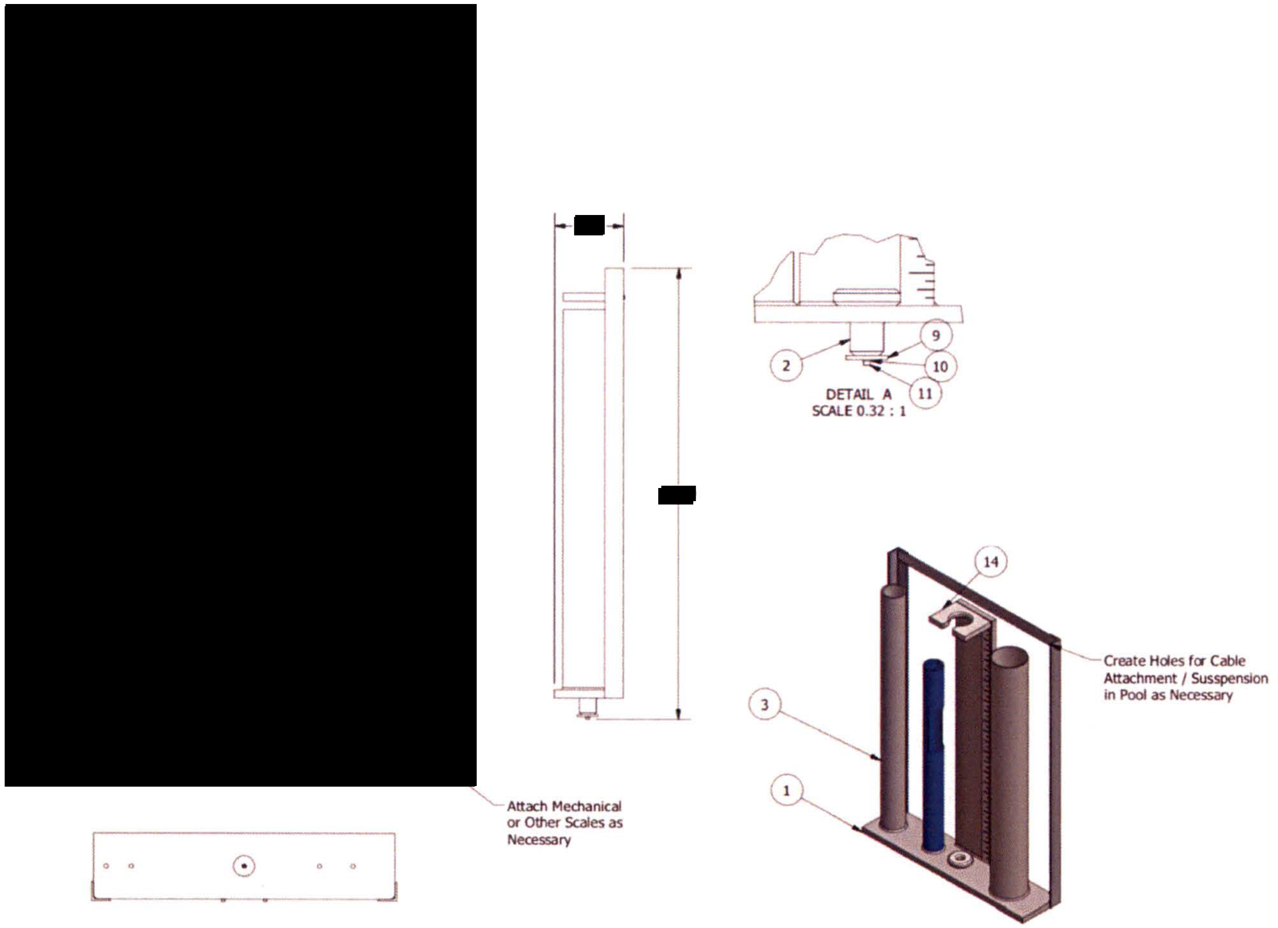


Figure 9.3-3
Fuel-Element Characterization Rig



PARTS LIST			
ITEM	QTY	PART NUMBER	DESCRIPTION
1	1	Base	1/4" x 4" ASTM B221 AL 6061
2	1	Rotator Cup	2" AL Round 6061
3	1	Stand	AL Tubing 2" O.D. .035" Wall
4	1	Back	1/4" x 4" ASTM B221 AL 6061
7	1	Rad Chamber	AL Tubing 1.63" O.D. .0.058" Wall
8	1	Scale	As Required
9	1	Bottom Retainer	1/8 AL 6061
10	3	ASME B18.21.1 - No.5	Heavy Helical Spring Lock Washers(Inch Series)
11	1	ANSI B18.3 - No. 5 - 40 UNC - 3/8 HS HCS	Hexagon Socket Head Cap Screw
12	1	Canister Hold Pipe	3" AL Pipe .125 Wall
14	1	Element Support	1/2" x 4" ASTM B221 AL 6061
15	2	ANSI B18.3 - No. 5 - 40 UNC - 3/4 HS HCS	Hexagon Socket Head Cap Screw
16	52.000 in	AISC - L 1.13 x 1.13 x 1/8 - 26	Angle Steel
17	1	Cross Bar	1/8" x 1" AL 6061

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 10 inch city water main located west of the Reactor Building. The fire main includes a standard fire department pumper connection located outside the facility security fence. An outside Post Indicator Control Valve, located away from the building near the security fence, 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 overhead sprinklers with fusible links. 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 a contractor.

A fire alarm pull box is located adjacent to the control room exit door. When the sprinklers are activated or the fire alarm pull box is used, a local alarm warns personnel to evacuate the building. In addition to the local alarm, the alarm system annunciates at a commercial monitoring company. The monitoring company contacts the SRVFPD and, if the alarm occurs during non-working hours, the on-call representative of Aerotest Operations, Inc. The pull box is tested every six months.

Fire extinguishers (Type ABC, Halon, Halotron and Type D) 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 a contractor Annual training of Aerotest Operations personnel in use of fire extinguishers is provided by a contractor.

Aerotest Operations, Inc. reviews emergency interface requirements with the SRVFPD annually to ensure that the SRVFPD will respond and assist ARRR personnel when required. The SRVFPD HAZMAT training includes assisting ARRR personnel in emergencies that may involve contamination with radioactivity and they are equipped with portable radiation survey instruments. SRVFPD personnel tour the ARRR facility on an annual to biennial basis for familiarization.

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 on the wall connecting the control room and the reactor room. This system detects the presence and indicates the intensity of gamma radiation and serves as both an area radiation monitor and a criticality alarm. In addition to the local alarm, this system annunciates at a commercial monitoring company. 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 SRVFPD. If the alarm occurs during non-working hours, the monitoring company contacts the on-call representative of Aerotest Operations, Inc. and the SRVFPD.

An evacuation alarm (designated as "Radiation") can be actuated from a switch located next to the ventilation shut off switch (designated as "Blower") adjacent to the control room exit door. Pressing the Radiation button will actuate the building radiation alarm. No signal is sent to the automatic alarm monitoring company. A reset button is located under each button.

The telephone system provides the primary communication mode. If company telephones are disrupted, personal cellular phones may be used. The alarm system is connected to the alarm company via multiple radio and relay stations. Continuity of the system is verified periodically by a polling computer. Automatic backup is a part of the system. A call list for staff and outside organizations is located at the reactor console.

An intercom and public address system is incorporated into all 17 telephones located throughout the ARRR building. Public address speakers are also located throughout the building.

Video surveillance cameras are installed at various indoor and outdoor locations and are monitored in the control room.

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 Specifications 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
- Pneumatic Transfer Facility
- 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 east 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 23 feet. The top of the beam tube terminates at the bottom of the reactor bridge structure. The external dimensions of the beam tube are about 8 inches by 10 inches near the base and expands to 22 inches by 34 inches at the top. The top of the vertical beam tube is supported laterally at the top of the pool.

The lower 48 inches 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 84 inches of the vertical beam tube is covered with lead for gamma shielding. This lead shield is 3 inches thick on the reactor side and 1 inch thick on the other three sides. The lead is protected from the pool water by welded sheets of aluminum. All components of the neutron radiography facility contacting the pool water are fabricated from aluminum or stainless steel.

The vertical beam tube includes an electronically 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] inch thick concrete block shielding stacked to a nominal height of [redacted] inches above the floor that surrounds the entire top of the reactor water tank. The neutron radiography facility is supported by [redacted] inch 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] inch thick wood (fir) beams that cover the reactor enclosure. This shielding provides operating personnel additional shielding.

This shielding enclosure is penetrated at the north and south ends on the east side of the reactor by the neutron radiography facility. Access to the top of the vertical beam tube is through openings on both the north and south ends with cross sectional dimensions of approximately 37 inches wide by 18 inches high. 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 made from wood and lead that is 48 inches by 47 inches by 16.5 inches and has wheels that roll on tracks attached to the tunnel. The wheels allow the beam catcher shield to be rolled to the south end to radiograph large items on the north end. The beam catcher shield reduces the radiation level due to neutrons and gammas within the reactor room and at the exclusion area fencing.

Rapid flooding of the lower section of the vertical beam tube would 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 contained within a wooden structure. The can has an outer diameter of about 20 inches and a height of about 28 inches. 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 5/8 inches and is securely bolted to the reactor support bridge. 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 19.5 lb 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 northeast side of the concrete shielding that surrounds the reactor water tank. 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 five rows of seven vertical holes through the graphite block. The vertical holes allow specimens to be inserted into the graphite block

for irradiation. The five rows of irradiation holes (A through E) are six inches 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 five rows, were used during startup physics testing (Reference 10.4.2) to measure flux levels at various positions in the thermal column.

There are seven irradiation holes in each row. The irradiation holes are 1.5 inches 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 4 feet along the radial axis of the core and is 2 feet wide and 2 feet deep. It is located on the south side of the reactor and positioned adjacent to the core. The thermal column is positioned using tapered pins and is bolted to the bottom of the reactor pool tank. Installation and removal of the whole assembly is accomplished with the facility crane and remote handling tools.

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 one inch 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% Δk/k (($0.01) with respect to the water it replaces. The small reactivity worth is due to the 2 inch 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. Four slotted beams, two 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, 1.5 inches 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 1.35 inches. 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 seven 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 $\leq 1.46\%\ \Delta k/k$ ($\leq \$2.00$). 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., $2.19\%\ \Delta k/k$ ($\leq \$3.00$)).

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 dry beam tube, up to 6 inches in diamete, 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. 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 16 square inches. The facility will accommodate specimens up to about 4.4 inches in diameter.

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 2.35 inch diameter or triangular experiments to a maximum of 3.0 inches 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 34 cubic inches 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 20 cubic feet 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.

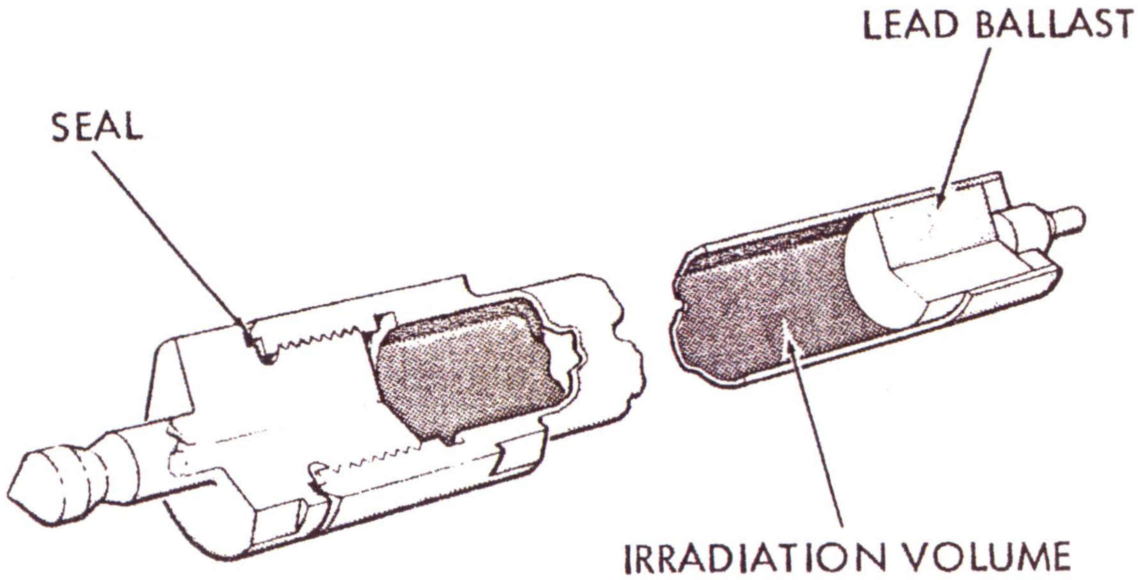
The pneumatic transfer facility may be located in any reactor core position. When installed, the facility is operated with dry CO_2 and exhausted through a filter ventilation system, which is monitored for radioactivity. The in-core portion of the transfer facility has a maximum void volume of 34 cubic inches 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 24 inch outside diameter pipe about 13 ft 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] inch thick concrete block shielding stacked to a nominal height of [redacted] inches above the floor that surrounds the entire top of the reactor water tank. This shielding supports the [redacted] inch thick wood (fir) beams which cover the entire enclosure of the reactor water tank. This 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 5/8 inches that surrounds the CRDs and is securely bolted to the reactor support bridge.

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 1000 grains (2.28 ounces)/unit with up to 1 pound in the radiation field at any one time and with a maximum of 50 pounds in process or in storage. The 50 pound 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 more than 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.

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

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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 2017, no spent fuel elements have been removed from the reactor core or transported from the facility and no spent fuel elements are currently being stored in the six fuel storage pits, located in the floor of the reactor room. Additionally, no highly radioactive waste or highly contaminated component is stored in the six 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.

Radiation Work Permits are required for all maintenance and other support personnel performing jobs in radiation areas to which they are not regularly assigned. The RWP 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 RWP since they are under direct supervision and therefore familiar with the radiation safety requirements of the area. All non-employees in radiation and non-radiation areas receive a radiation orientation.

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] inch thick high density concrete that surrounds the reactor pool and the active radiography area which includes [redacted] inch thick wooden 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. Multiple personnel are trained to perform this task, so teams perform 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 years 2000-2004. 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 <u>Criticality Alarm and Area Radiation Monitor</u>, described in USAR Chapter 7, is a fixed gamma monitor employing Geiger tube detectors that is 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. 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. 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.

The <u>building gaseous effluent monitor</u>, 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 after allowing for radon decay..

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.

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 may be 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 two 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. No 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.

Two above ground storage tanks and one sump tank are used for storage of liquid wastes which could be contaminated. The sump tank is used to collect liquids from the trenches around the reactor, the heat exchanger building, the demineralizer building, and one of the chemical laboratory sinks. Potentially contaminated liquids are completely isolated from the sanitary sewer system. An automatic sump pump moves the waste to the primary waste storage tank. This tank can be monitored by a GM counter connected to a ratemeter in the control room should there be any suspicion that measurable amounts of radioactive material have been released. A high liquid level indicator annunciates on the console when the level reaches about two feet from the top of the tank.

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 those 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.

A second above ground tank is used to temporarily store waste while the primary tank liquid is being analyzed for radioactivity prior to release.

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 DOT specification packaging for radioactive materials provided for this purpose. These 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

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 Reactor Supervisor, Radiation Safety Officer, and the Reactor Safeguards Committee.

The ARRR Reactor Supervisor (RS) is assigned the responsibility for implementing and is directly responsible for enforcing all rules, regulations, and procedures relating to reactor operation, safety and security. The RS must review and approve all procedures and experiments involving the operation of the reactor is responsible for conducting and documenting routine and non-routine activities associated with reactor safety operation and security. The RS must approve all operations involving the reactor and has the

authority to stop any operation which, in his or her opinion, is unsafe. The RS reports directly to the President or his designee and is a member of the facility Reactor Safeguards Committee. The qualifications of the RS must include, at a minimum, a Bachelor's degree in Physical Science, Engineering or 4 years of experience as an SRO at the AO facility.

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 President or his designee and is a member of the facility Reactor Safeguards Committee. The qualifications of the RSO must include, at a minimum, a Bachelor's degree in Biological, or Physical Science, Engineering or 2 years of experience in personnel and environmental radiation monitoring programs at the AO facility.

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, Corporate Representative and at least one other member. Members will come from 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 also verifies that the ARRR is operated within the limits of the license by performing at least one unannounced audit annually.

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.

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 4 voting members, of whom no more than two are members of the operating staff. 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 the Radiological Safety Officer who reports directly to the President, 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 for the ARRR facility.

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 the transient and accident scenarios for the ARRR facility indicate the ARRR facility is very conservatively bounded by generic and actual ARRR historic accident analyses that concluded that none of the credible accidents (actual or postulated) pose an uncontrolled release of fission products.
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 incidents of mishandling/ malfunction of the fuel at ARRR (and from general analysis for TRIGA reactors) there is no credible event at the ARRR facility that could result in the exposure to the general public greater than that allowed by 10 CFR 20, App. B, Table II, for unrestricted areas.

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 63 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 2017 Aerotest Operations possesses 55 irradiated aluminum clad elements, 27 irradiated stainless steel clad elements, 22 canned irradiated aluminum clad elements and 12 unirradiated 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 50 and 65 fuel elements. At initial criticality, the ARRR core consisted of 63 aluminum clad TRIGA fuel elements. Additional fuel elements are added 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] inches 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% Δk/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

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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

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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 as "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

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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 [redacted] 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 [redacted] inch steel I-beam structure described in USAR Chapter 10, Experiments and Explosives. Large concrete blocks [redacted] 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 86 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-98.

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

Section 15 contains confidential information submitted under 10 CFR 9.17(a)(4). This information is Confidential to Aerotest Operations and is identified with brackets as such [ ].

Aerotest Operations, Inc. (AO), a California corporation is the current NRC licensed owner and operator of the ARRR and a wholly owned subsidiary of Nuclear Labyrinth, LLC. In June 2017 after NRC approved an indirect license transfer, all outstanding shares of Aerotest Operations, Inc. were acquired by Nuclear Labyrinth, LLC, a Utah company. The Board of Directors of AO, and Nuclear Labyrinth, LLC are composed entirely of United States citizens.

15.1 Financial Ability to Operate a Non-Power Reactor

Aerotest Operations, Inc. had annual total revenues of approximately [ ] and net income of [ ] during the years 2000-2010. During that time, activities were performed in areas of research, development, investigative (forensic research). AO also participated in improvement of neutron radiology processes and development of effective curriculum to teach the skill. It is anticipated that these areas will play an ever more expansive role in the future services offered. This financial picture is expected to stabilize after the first year leaving the next 4 years as shown in the revenue/cost projections with increases in income and larger expenditures available for upgrading the ARRR and improving AO's research and development capabilities (See Table 15.1, Proprietary). AO has approximately [ ] in cash reserves to cover the initial year for restart activities if needed. In addition, the Facility is self-insured at the sum of [ ] which is available in a financial protection standby trust for the life of the reactor.

15.2 Financial Ability to Decommission the Facility

The financial resources to decommission the facility per 10 CFR 50.33(k) and 50.75(e) will be readily available from funds placed in trust by Nuclear Labyrinth, LLC for Aerotest Operations, Inc. Decommissioning Trust provisions include:

1. The sum of [ ] in a decommissioning trust fund for the Facility; the onsite estimated cost was generated by Energy Solutions Inc. for the Facility.
2. The sum of [ ] in a segregated account in the decommissioning trust fund for the disposal of the Facility's nuclear fuel elements pursuant to U.S. Department of Energy ("DOE") Contract DE-CR01-83NE4484, as amended;
3. The sum of [ ] in a segregated account in the decommissioning trust fund intended to fund the acquisition of fuel element storage casks;
4. The sum of [ ] in a segregated account in the decommissioning trust fund that manages the nuclear fuel elements after permanent cessation of the Facility's operations and before acceptance of the fuel by the DOE;

To ensure an unlikely event of a deficit in decommissioning funds, a sum of [ ] is available in an Irrevocable Letter of Credit.

15.3 Compliant 104c License

Given the reactor design and operating characteristics, its use (Atomic Energy Act-1954 as amended), compliant with 10 CFR.50.21 provisions, and operation at a thermal power level of 10 Megawatts or less (10 CFR 170.3), the 104c is the appropriate NRC license (R-98) for this TRIGA 250 KW nuclear reactor. A detail analysis is provided in a response to RAI-GEN-2 (October 24, 2017 letter -MI17277B261)

The costs assigned due to function and activity are shown below. The annual percentage of direct costs associated with reactor operations (that are not associated with Research, Development and Investigative research, or education/training activities) came from Professional N-Radiograph Sales and that portion has been estimated at 15% annually between the years 2000-2010 (Item 5). If 3% indirect costs (1/5 of item) is also included, the result would be 18%; this outcome is lower than the 50 % Metric defines. (Item 1 was funded entirely by the parent company Autoliv).

1. 15% Indirect Costs ANI (and other) Insurance, fuel purchase and reactor upgrades, NRC, regulatory Costs (annual fee, inspections, PM, etc.), and Decommissioning & fuel-storage annual contribution.
2. 15% Indirect Costs associated with maintaining building, its infrastructure, and Fed/State/Local regulatory compliance other than reactor operation.
3. 30% Direct Cost associated with nonreactor operations activities
4. 15% Direct Costs Reactor Operation for Research/Development and Investigative (forensic) services (RDI).
5. 15% Direct Costs for Reactor Operation for Professional N-Ray Sales (PS).
6. 10% Direct Costs for Education/Training activities and Community participation

The Total Annual Cost associated with the Metric from 2010-2017 clearly plummeted from 18% to 0 %.

However, the parent company has recently changed from Autoliv to Nuclear Labyrinth LLC, a small business. If it is assumed that the corporate structure was that of a small business and the activities remain the same in the reported proportions, the relevant costs include direct, 15% (Item 5), indirect, 7.5% (1/2 of 15% , Item 1), and 3% (1/5 of 15%, Item 2). The result would be 25.5%; this outcome would still be lower than the 50% Metric defines. In the future, activities will include research in advance neutron detection/interrogation and Mo-99 target development. These reactor activities will further lower the proportion of the reactor operating costs associated with N-radiograph sales and thus increasing the gap between the 50% Metric and the lower estimated reactor operating costs due to commercial activities.

AEROTEST OPERATIONS, INC.

Aerotest Radiography and Research Reactor (ARRR)

APPENDIX A:

TECHNICAL SPECIFICATIONS

Version 1-2017

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

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

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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_{eff},
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 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 \$1.00when 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 \$1.00when 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.
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

Applicability: This specification applies to the temperature of the reactor fuel.

Objective: The objective is to define the maximum fuel element temperature that can be permitted, with confidence that no damage to the fuel element cladding will result.

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

Basis:

The most important safety limit for a TRIGA reactor is the fuel element temperature. This parameter is well suited as a single specification because it has been measured directly with embedded thermocouples and the measured results studied extensively for the TRIGA core designs, ZrH-U fuel type and composition, with fuel meat cladded with stainless or aluminum. The temperature safety limit for the TRIGA fuel elements is based on data which indicates that the internal stresses within the fuel element, due to hydrogen pressure from the dissociation of the zirconium hydride, will not result in compromise of the fuel element cladding if the fuel temperature is not allowed to exceed 535 °C. A loss of integrity of the fuel element cladding could arise from a buildup of excessive pressure if the fuel element temperature exceeds the temperature safety limit. The fuel element temperature and the ratio of hydrogen to zirconium in the fuel- moderator material determine the magnitude of the pressure buildup. The mechanism for the pressure buildup is the dissociation of hydrogen from the zirconium hydride moderator that has been blended with uranium to form the fuel mixture encased within the fuel element cladding.

Because the ARRR uses both aluminum clad and stainless steel clad fuel elements, this SL is based on the aluminum clad fuel which is more limiting. The 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 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.

The highest fuel temperature located in the center position (A1) is estimated to be less than 300 °C during steady state operation at 250 KW 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

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):

Applicability: This specification applies to the energy generated in the reactor during operations, at all times.

Objective: The objective is to ensure that the fuel temperature safety limit will not be exceeded.

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

Basis:

The 250 kW limit for rated thermal power 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 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.

3.1.2 Excess Reactivity:

Applicability: This specification applies to the reactivity condition of the reactor and the reactivity worths of control rods and experiments, and applies for all modes of operation at all times.

Objective: The objective is to ensure that the reactor can be shutdown at all times.

Specification: The maximum excess reactivity with the reactor in reference core condition, with and without experiments in place, shall be $3.00.

Basis:

The nominal total rod worth is $8.07 Subtracting the shutdown margin ($0.50) and the nominal rod worth of the most reactive rod ($3.71) provides the shutdown reactivity of $3.86, well above the excessive reactivity limit of $3.00.

The specification of $3.00 helps ensure that the ARRR's operational power densities, fuel temperatures, and temperature peaks are maintained within the evaluated safety limits. The specified excessive reactivity allows for power coefficients of reactivity, xenon poisoning, most experiments, and operational flexibility.

3.1.3 Shutdown Margin (SDM):

Applicability: This specification applies to the reactivity condition of the reactor and the reactivity worth of the control rods and experiments. It applies for all modes of operation at all times.

Objective: The objective is to ensure that the reactor can be shutdown at all times and to ensure that the fuel temperature safety limit will not be exceeded.

Specification: The Shutdown Margin, with and without experiments in place, shall be ≥ $0.50.

Basis:

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.

3.1.4 Core Configuration:

Applicability: This specification applies to the configuration of fuel elements and in-core experiments.

Objective: The objective is to ensure the provisions are made to restrict the arrangement of fuel elements and experiments so as to provide assurance that excessive power densities will not be produced.

Specifications:

1. The reactor core shall be an arrangement of TRIGA LEU cylindrical stainless-steel clad, and aluminum clad, low hydride fuel-moderator elements with neutron reflection provided by graphite elements in aluminum cladding.
2. The reflector, excluding experiments and experimental facilities, shall be a combination of water, graphite, and heavy water.
3. Fuel shall not be removed from or inserted into the core unless the reactor is subcritical by more than the calculated worth of the most reactive fuel element.
4. Control rods shall not be removed manually from the core unless the core has been shown to be subcritical, and shutdown margin requirements met, with those control reds removed.

Basis:

The ARRR utilizes solid fuel elements, developed by General Atomics (GA), in which the zirconium-hydride moderator is homogeneously combined with enriched uranium. The unique feature of these fuel-moderator elements is the prompt temperature coefficient of reactivity, which gives the TRIGA reactor its built-in safety by automatically limiting the reactor power to a safe level in the event of a power excursion. The ARRR reactor core consists of a lattice of cylindrical stainless steel clad, high hydride ($U\text{-}ZrH_{1.6}$) fuel-moderator elements, and aluminum clad, low hydride ($U\text{-}ZrH_{1.0}$) fuel-moderator elements. Neutron reflection in the radial direction is provided by graphite and heavy water elements in an aluminum cladding. Also the core is immersed in a water tank, which acts as a thermal shield and a moderator. The core components are contained between top and bottom aluminum grid plates. The top grid plate has 126 positions for fuel elements and control rods arranges in 6 concentric rings around a central port.

The core will be assembled in the reactor grid plate located at the bottom of tank filled with light water, in combination with graphite and heavy water elements for neutron economy and to enhance requirements for experimental facilities.

3.1.5 Fuel Condition

Applicability: This Specification applies to all fuel elements

Objective: The objective is to maintain integrity of the fuel cladding.

Specifications: The reactor shall not be operated with damaged fuel elements, except for the purpose of locating damaged elements. A fuel element shall be considered damaged and must be removed from the core if:

a. Cladding defect exists as indicated by release of fission products;

b. Visual inspection identifies bulges, gross pitting or significant corrosion;

c. The sagitta transverse bend exceeds 0.0625 inches over the length of the cladding;

d. The length exceeds its original length by 0.125 inches; or

e. The burn up in the fuel matrix exceeds 50 percent of the initial concentration

Basis:

Catastrophic failure or the obvious visual identification of bulges gross pitting or extensive corrosion of the fuel is sufficient to warrant declaration of the fuel as damaged. (NUREG-1537)

3.2 REACTOR CONTROL AND SAFETY SYSTEMS

3.2.1 Control Rods:

Applicability: This specification applies to the reactor control and safety systems, whenever the reactor is in the reactor operating condition.

Objective: The objective is to achieve reactivity control and prompt shutdown of the reactor to prevent fuel damage.

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

Basis:

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 a 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.

3.2.2 Scram Time:

Applicability: This specification applies to the control rod scram times, whenever the reactor is in the reactor operating condition.

Objective: The objective is to achieve prompt shutdown of the reactor to prevent fuel damage.

Specification: 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.

Basis:

This specification ensures that the reactor will be promptly shutdown when a scram signal is initiated. Experience and analysis have indicated that for the range of transients anticipated for a TRIGA reactor, the specified scram time is adequate to ensure the safety of the reactor.

This LCO requires that control rods be fully inserted 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.

3.2.3 Reactivity Insertion Rate:

Applicability: This specification shall apply whenever the reactor is in the reactor operating condition.

Objective: The objective is to ensure that the reactor operator can easily control reactor power.

Specification: The maximum rate of reactivity addition by the control rods shall be +$0.11/second.

Basis:

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.

3.2.4 Reactor Safety Interlocks:

Applicability: This specification applies to the reactor safety system channels and interlocks, whenever the reactor is in the reactor operating condition or reactor shutdown condition.

Objective: The objective is to specify the minimum number of reactor safety systems and interlocks that must be operable for safe operation.

Specification:

(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 withdrawal 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.

Basis:

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 $1.00 and is not at significant risk from an inadvertent addition of reactivity.

3.2.5 Reactor Safety Channels:

Applicability: This specification applies whenever the reactor is in the reactor operating condition.

Objective: The objective is to ensure accurate monitoring and safe operation.

Specification: Reactor Safety Channels:

(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.

Basis:

LCO 3.2.5 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 Safety Channels, provides a detailed description of each of the channel functions.

Although the reactor operator is expected to act conservatively in response to the inoperability of multiple channels, allowing one hour to attempt restoration of multiple inoperable channels is acceptable because none of the 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.

3.2.6 Reactor Auxiliary Safety Systems:

Applicability: This specification applies whenever the reactor is in the reactor operating condition.

Objective: The objective is to ensure safe operation of the reactor.

Specification: Reactor Auxiliary Systems:

(1) Each Reactor Auxiliary Systems, 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.

Basis:

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 Auxiliary Safety Systems, provides a detailed description of each function.

Required actions and completion times for inoperable channels are provided in Table 3.2-2, footnotes (a), (b) and (c).

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.

Table 3.2-1
Reactor Safety Channels

Channel	Minimum No. of Required Channels (c)(d)	Required Function(s)	Limiting Safety System Setting	Annunciator and Alarm Set Point
1 Counts Proportional Counter	1(a)	Reactor Trip: Short Period Low Source Level	 ≥ 3 seconds ≥ 120 cpm	 ≥ 3 seconds ≥ 120 cpm
2 Log Power Compensated Ion Chamber	1	Reactor Trip: Short Period Loss of Inst Power Low Detector Voltage	 ≥ 3 seconds X ≥ 500 volts	 ≥ 3 seconds X ≥ 500 volts
3 Linear Power Un-Compensated Ion Chamber	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
4 Linear Power Compensated Ion Chamber	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 Auxiliary Safety Systems

Channel	Minimum No. of Required Channels	Required Function(s)	Annunciator and Alarm Set Point
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 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:

Applicability: This specification applies to the reactor tank water level, at all times, whenever the reactor is in the reactor operating condition or reactor shutdown condition.

Objective: The objective is to ensure adequate circulation for natural convection cooling, prevent a loss-of-coolant accident, and to provide adequate shielding for personnel working around the reactor pool.

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

Basis:

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.

3.3.2 Primary Coolant Water Temperature:

Applicability: This specification applies to the reactor pool and the primary cooling systems whenever the reactor is in the reactor operating condition or reactor shutdown condition.

Objective: The objective is to ensure that the prompt negative temperature coefficient is maintained in the fuel, and that the resin in the demineralizer system is not damaged by high temperatures.

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

Basis:

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.

3.3.3 Primary Coolant Water Quality:

Applicability: This specification applies to the reactor pool and the primary cooling systems, at all times.

Objective: The objective is to minimize corrosion in the reactor pool, and minimize activated corrosion products in the water.

Specification: 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.

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.

Basis:

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.

3.4 VENTILATION SYSTEMS

3.4.1 Control Room:

Applicability: This specification applies to the air pressure in the control room whenever the reactor is in the reactor operating condition or reactor shutdown condition and during movement of irradiated fuel.

Objective: The objective is to prevent airborne radioactive material from entering the control room during an emergency.

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

Basis:

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.

3.4.2 Reactor Building Circulation Fans:

Applicability: The specification applies to all ventilation systems open to the high bay area whenever the reactor is in the reactor operating condition or reactor shutdown condition and during movement of irradiated fuel.

Objective: The objective is to prevent airborne radioactive material from entering the control room during an emergency.

Specification: 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.

Basis:

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.

3.5 CRITICALITY ALARM, RADIATION AND RADIOACTIVE EFFLUENT MONITORING

3.5.1 Criticality Alarm and Area Radiation Monitoring:

Applicability: This specification applies to the radiation monitoring information that must be available to the reactor operator during reactor operations, any fuel movement, or control rod work, or any handling of radioactive material with the potential for airborne release.

Objective: The objective is to ensure that sufficient radiation monitoring information is available to the operator to ensure safe operation of the facility.

Specification:

(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.

Basis: The radiation monitors provide information to operating personnel of any impending or existing danger from radiation so that there will be sufficient time to evacuate the facility and take the necessary steps to prevent the spread of radioactivity to the environment.

3.5.2 Gaseous Effluent Monitoring:

Applicability: This specification applies whenever the reactor is in the reactor operating condition or reactor shutdown condition and during movement of irradiated fuel.

Objective: The objective to ensure that the health and safety of the public is not endangered by the discharge of any radioactive gaseous effluent from the ARRR.

Specification: 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.

Basis: 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.

3.5.3 Particulate Effluent Monitoring:

Applicability: This specification applies to the air sampling system, whenever the reactor is in the reactor operating condition or reactor shutdown condition and during movement of irradiated fuel.

Objective: The objective is to ensure that the staff is aware of any excessive airborne radioactivity.

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

Basis: 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.

3.5.4 Post-Accident Radiation Monitoring:

Applicability: This specification applies to the so-called "disaster dosimeters" which shall be present in the facility.

Objective: The objective is to provide accurate post-accident analysis of radiation levels in the facility.

Specification: 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.

Basis: In the event of a criticality accident, it is desirable to know the neutron energies and dose that occurred. There is one disaster dosimeter box in the control room, and another mounted on the wall in the setup room to provide information for the two most likely places that personnel would be when an event started.

3.5.5 Portable Radiation Monitoring Instruments:

Applicability: This specification applies to portable radiation monitoring equipment, whenever the reactor is not shutdown, or if there is any work involving any possible radioactivity or in any possible radiation field, and at all other times except when instruments are out of the plant for independent calibration.

Objective: The objective is to ensure that portable instruments are always available when personnel are in the plant.

Specification: 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.

Basis:

SR 4.5.5 requires quarterly calibrations of each required portable radiation monitoring instrument. The radiation monitors provide information to personnel of any impending or existing danger from radiation so as to prevent unnecessary exposure and prevent the spread of radioactivity.

3.5.6 Radioactive Effluent Limits:

Applicability: The specification applies to any effluents from the ARRR whether gas or liquid, at all times.

Objective: The objective is to ensure that the health and safety of the public is not endangered by the discharge of any radioactive effluent from the ARRR.

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

Basis:

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.

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 a 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:

Applicability: The specification applies to all experiments performed with the reactor, prior to insertion of the experiment into the reactor enclosure.

Objective: The objective is to prevent damage to the reactor or excessive release of radioactive materials in case of a failure of an experiment.

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

Basis:

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. 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.

3.6.2 Reactivity Limits during Experiments:

Applicability: The specification applies to the reactivity limits on experiments installed in the reactor and its experimental facilities, prior to initiation of the experiment.

Objective: The objective is to prevent damage to the reactor or excessive release of radioactive materials in case of a failure of an experiment.

Specification: 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 ≤ $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 ≤ $2.00.

(2) The reactivity worth of any single independent experiment not rigidly fixed in place shall be ≤ $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 ≤ $1.00.

(3) Experiments having moving parts shall be designed to have reactivity insertion rates <$0.10/second, except that moving parts worth <$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.

Basis:

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 maximum worth of a single experiment is limited so that its removal from the reactor in reference core condition will not result in the reactor achieving a power level high enough to exceed the fuel element temperature safety limit. 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., $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. The limit poses a restriction on the total absolute value of reactivity of experiments being run at any given time to prevent excessive positive and negative reactivity effects from experiments.

3.6.3 Special Nuclear Material (SNM) included in Experiments:

Applicability: This specification applies to any experiment that contains Special Nuclear Material that is inserted into the reactor, at all times.

Objective: The objective is to prevent excessive reactivity changes or damage to the reactor by limiting SNM quantities in experiments inserted into the reactor.

Specification: 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.

Basis: 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.

3.6.4 Materials Used during Experiments:

Applicability: This specification applies to experiments installed in the reactor and its experimental facilities.

Objective: The objective is to prevent damage to the reactor components or excessive release of radioactivity in case of failure of an experiment.

Specification: 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.

Basis: 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.

3.6.5 Explosive Materials in Experiments:

Applicability: This specification applies to experiments installed in the reactor and its experimental facilities.

Objective: The objective is to prevent damage to the reactor or excessive release of radiation in case of a failure of an experiment.

Specification: 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.

<u>Basis</u>: 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 5×10^2 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.

3.6.6 Configuration and Potential Failure Mechanisms:

Applicability: This specification applies to any experiments around the reactor, at all times.

Objective: The objective is to prevent damage to the reactor, excessive release of radiation in case of a failure of an experiment, or operation outside of licensed limits.

Specification:

(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.

Basis: These issues are considered in the review and approval of experiments required by LCO 3.6.1.

3.6.7 Glory Hole Facility:

Applicability: This specification applies to any a glory hole facility located in any reactor core position and whenever the reactor is in the reactor operating condition.

Objective: The objective is to prevent excessive generation of Argon-41.

Specification:

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.

Basis:

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 1.35 inches. 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.

3.6.8 Vertical Tubes:

Applicability: This specification applies to any Vertical Tubes attached to the thermal column, whenever the reactor is in the reactor operating condition.

Objective: The objective is to prevent excessive production and release of Argon-41.

Specification: 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.

Basis:

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.

3.6.9 Other Irradiation Facilities:

Applicability: This specification applies to the central irradiation facility and the two triangular facilities in the reactor core, and the Dummy Element Irradiation Capsules (DEIC).

Objective: The objective is to provide limits on core use for items other than fuel elements.

Specification:

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.

Basis:

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.

3.7 FUEL STORAGE AND TRANSFER

3.7.1 Fuel Handling Tools:

Applicability: This specification applies to the fuel handling tools, at all times except when the tools are in use.

Objective: The objective is to ensure that the fuel handling tools cannot be accessed by unauthorized users.

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

Basis: This requirement prevents unauthorized movement of fuel and minimizes the potential for theft or diversion of a fuel element.

3.7.2 Fuel Storage in the Reactor Tank:

Applicability: This specification applies to any reactor fuel element, at all times.

Objective: The objective is to prevent accidental criticality in fuel storage areas.

Specification: 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.

Basis: This limit is acceptable because it provides a very conservative margin for the prevention of an unplanned criticality in the storage racks.

3.7.3 Fuel Storage in the Fuel Storage Pits:

Applicability: This specification applies to any reactor fuel elements, at all times.

Objective: The objective is to maintain safe subcriticality and to prevent unauthorized removal.

Specification: 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.

Basis:

These limits eliminate the potential for an unplanned criticality in the fuel storage pits and ensure that decay heat levels will be minimal. It also 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.

3.7.4 Fuel not in the Reactor and not in Storage:

Applicability: This specification applies to any reactor fuel elements, at all times.

Objective: The objective is to prevent theft or diversion of fuel elements.

Specification: No more than one fuel element shall be allowed in the facility which is not in storage or in the core lattice.

Basis: 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.

3.7.5 Fuel in Shipping Containers:

Applicability: This specification applies to reactor fuel elements, whenever a fuel shipping container is in use.

Objective: The objective is to allow for the transfer of fuel elements from a shipping container into storage.

Specification: Fuel may be contained in an approved fuel shipping container within the limitations of LCO 3.7.4. When an approved fuel shipping container is used, then the licensed limitations on k_{eff} for the container shall apply.

Basis: 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.

3.7.6 Fuel Transfer in the Reactor Tank:

Applicability: This specification applies to the movement of fuel elements within the reactor tank.

Objective: The objective is to ensure safe movement of elements without accidental criticality.

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

Basis: 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.

3.7.7 Fuel Transfer Outside the Reactor Tank:

Applicability: This specification applies during transfers of fuel outside the reactor tank.

Objective: The objective is to ensure safe movement of elements without accidental criticality.

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

Basis: 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

3.8.1 Reactor Building Intrusion Alarm:

Applicability: This specification applies to the reactor building burglar alarm system, panic and duress alarms, and motion detection alarms, at all times.

Objective: The objective is to ensure the ability to alert local law enforcement to the presence of intruders.

Specification: The reactor building alarm shall be monitored continuously.

Basis: LCO 3.8.1 requires that the reactor building alarm be monitored at all times to detect unauthorized entry into the ARRR building.

The system should be operable at all times except in the rare case of component failure, when the vendor is expected to dispatch a repair person immediately. This site security alarm has a minimum 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.

3.8.2 Explosive Material at the Reactor Facility:

Applicability: This specification applies to all explosive materials in the building, at all times.

Objective: The objective is to minimize potential injury.

Specification: 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 when explosives are present on site (which includes standards containing explosive materials) 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).

Basis: 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.

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):

Applicability: This specification applies to the surveillance requirement for power level monitoring channels.

Objective: The objective is to verify that the maximum power level of the reactor meets the license requirements.

Specification: A reactor thermal power calibration shall be performed annually. Linear Power Channel 3 and Channel 4 shall reflect the results of this thermal power calibration.

Basis: The power level channel calibration will ensure that the reactor will be operated at the proper power level.

4.1.2 Excess Reactivity:

Applicability: This is specification applies to the surveillance requirements of the reactor excess reactivity.

Objective: The objective is to verify that requirements on excess reactivity are met for operational cores.

Specification: 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.

Basis: The excess reactivity of the core is measured to ensure that during all states of operation, criticality can be maintained for licensed operation limits. With the accumulation of fission product poison build up and the fissile material burn up, excess reactivity must be available for power transients and maintaining criticality.

4.1.3 Shutdown Margin (SDM):

Applicability: This specification applies to the surveillance requirement of shutdown margin.

Objective: The objective is to verify that the requirement for the shutdown margin is met for the operational core.

Specification: SDM shall be verified to be within the limits of LCO 3.1.3 ($0.50) annually and following any significant change to the core or any control rod.

Basis: The reactivity worth of the control rods is measured to ensure that the required shutdown margin is available, and to provide an accurate means for determining the reactivity worth of experiments inserted in the core. Experience with TRIGA reactors gives assurance that the measurement of the reactivity worth on an annual basis is adequate to ensure no significant changes in the shutdown margin.

4.1.4 Core Configuration Limitation:

Applicability: This specification applies to the surveillance requirement for core configuration.

Objective: The objective is to verify the core is in a safe reviewed and approved configuration.

Specification: Each core configuration change shall be determined to meet the requirements of LOC 3.1.4 prior to the core loading.

Basis: The requirement of LCO 3.1.4 assures acceptable safety analysis is complete for a core configuration, as well as prevent accidental fuel damage, fuel addition, or criticality events.

4.1.5 Fuel Condition:

Applicability: This specification applies to the surveillance requirements for determining Fuel cladding condition.

Objective: The objective is to verify the integrity of the cladding of fuel elements.

Specification:
A minimum of 20% of the fuel elements in the reactor core lattice shall be examined by visual/tested inspection annually.

Each fuel element in the core shall have been examined by visual inspection, performed a go/no-go Sagitta (transverse bend) test, and when appropriate a bottom pin seat test within the previous five years.

Basis: The requirements of 3.1.5 ensure acceptable safety analysis is to prevent accidental fuel damage. Because fuel in storage experiences negligible wear except due to handling, it is not subject to the regularly scheduled visual/test inspection. Inspections may be required before an element is moved from storage into the core in order to insure it meets the fuel condition parameters before insertion in the core.

4.2 REACTOR CONTROL AND SAFETY SYSTEMS

4.2.1 Operable Control Rods:

Applicability: This specification applies to the surveillance requirements for measurements, tests, and calibrations of the control and safety systems.

Objective: The objective is to verify the performance and operability of the systems and components that are directly related to reactor safety.

Specification: Free movement of each control rod for insertion is verified by the performance of SR 4.2.2.

Basis: Control rod scram tests will ensure that the control rod drives are operable prior to an extended run.

4.2.2 Scram Time:

Applicability: This specification applies to the surveillance requirement for control rod drop times.

Objective: The objective is to ensure that there are no obstructions to prevent the rods from dropping into the core as designed.

Specification: The control rod scram times shall be measured and verified to be within the limits of LCO 3.2.2 (≤ 600 milliseconds) semiannually and following any significant change to the core or any significant work on the control rods or the control rod drive system

Basis: Control rod drop times will ensure that there has been no introduction of corrosion or any other source of friction into the guide tubes.

4.2.3 Reactivity Insertion Rate:

Applicability: This specification applies to the surveillance requirement for the reactivity insertion rate.

Objective: The objective is to ensure that the reactivity insertion rate does not exceed the LCO 3.2.3 requirement.

Specification: The reactivity worth of each control rod shall be measured annually and following significant changes to the core or any control rod. 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 (11 cents per second).

Basis: The annual frequency for verification of control rod reactivity worth is consistent with the expected slow change in reactivity worth over time. 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 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.

4.2.4 Reactivity Insertion Monitoring:

Applicability: This specification applies to the surveillance requirement for reactivity insertion monitoring.

Objective: The objective is to ensure that any reactivity increase during introduction of an experiment to the reactor core does not exceed limitations.

Specification: 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.

Basis: Any introduction of an experiment which can affect reactivity in the core requires careful monitoring.

4.2.5 Reactor Safety Interlocks:

Applicability: This specification applies to the surveillance requirements for measurements, tests, and calibrations of the control and safety systems.

Objective: The objective is to verify the performance and operability of the systems and components that are directly related to reactor safety.

Specification:

(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.

Basis: Tests will ensure that the safety system channels are operable on a daily basis or prior to a run.

4.2.6 Reactor Safety Channels:

Applicability: This specification applies to the surveillance requirements for reactor protection systems.

Objective: The objective is to verify the condition and operability of system components affecting safe and proper control of the reactor.

Specification:

(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.

Basis: Channel tests will ensure that the safety system channels are operable on a daily basis.

4.2.7 Reactor Ancillary Safety System:

Applicability: This specification applies to the surveillance requirements for the Reactor Monitoring System instrumentation.

Objective: The objective is to verify the condition and operability of system components directly related to the channels that monitor reactor room systems.

Specification:

(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.

Basis: Channel tests will ensure that the monitoring channels are operable on a daily basis.

4.3 COOLANT SYSTEMS

4.3.1 Reactor Tank Water Level:

Periodic verification (quarterly) 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 (quarterly) of primary coolant water scram 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:

Applicability: This specification applies to the surveillance requirements for coolant quality.

Objective: The objective is to ensure the water quality and radioactivity of the reactor coolant remains within defined limits.

Specification:

(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.

Basis: A small rate of corrosion continuously occurs in any water-metal system. In order to limit this rate, and thereby extend the longevity and integrity of the fuel cladding, a water cleanup system is required. Experience with water quality control at many reactor facilities has shown that maintenance within the specified limits provides acceptable control.

4.4 VENTILATION SYSTEMS

4.4.1 Control Room:

Applicability: This specification applies to the ventilation system in the high bay area, office area, and the control room.

Objective: The objective is to ensure the proper operation of the ventilation system to prevent uncontrolled releases of particulate radioactive material into the evacuation area.

Specification: The control room shall be verified to be at a positive air pressure with respect to the reactor room quarterly.

Basis: Experience accumulated over several years of operation has demonstrated that the tests of the ventilation system on a quarterly basis are sufficient to ensure the proper operation of the system and control the release of particulate matter.

4.4.2 Reactor Building Circulation Fans:

Applicability: This specification applies to the reactor building circulation fans.

Objective: The objective is to minimize the movement of particulate radiation in the event of a radiological emergency.

Specification: 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.

Basis: Experience accumulated over several years of operation has demonstrated that the tests of the ventilation system on a quarterly basis are sufficient to ensure the proper operation of the system and control the release of particulate matter.

4.5 CRITICALITY ALARM, RADIATION AND RADIOACTIVE EFFLUENT MONITORING

Applicability: This specification applies to the surveillance requirements for the criticality alarm, area radiation monitoring equipment, and to effluents.

Objective: The objective is to ensure that the radiation monitoring equipment is operating with appropriate alarm settings and to ensure a gaseous and liquid affluence are in accordance with 10 CFR 20.

Specification:

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.

(i) A channel calibration of the channel listed in Table 3.5-1 shall be performed biennially.

(2) 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.

Basis: Experience has shown that periodic verification of area radiation and air monitoring system operations in conjunction with annual calibration is adequate to correct for any variation in the system due to a change of operating characteristics over a long times they. Monitoring and calculating the amount of gaseous and liquid affluence will allow assurance that they are in accordance with 10 CFR 20.

4.6 EXPERIMENTS

Applicability: This specification applies to the surveillance requirements for experiments installed in the reactor and its experimental facilities, and for irradiations performed in the irradiation facilities.

Objective: The objective is to prevent the conduct of experiments or irradiations that may damage the reactor or release excessive amounts of radioactive materials as a result of failure.

Specification

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.

Basis:

It has been demonstrated over a number of years that experiments and irradiations reviewed by the Reactor Staff and the Reactor Safeguards Committee can be conducted without endangering the safety of the reactor or exceeding the limits and the Technical Specifications.

4.7 FUEL STORAGE AND TRANSFER

Applicability: This specification applies to the surveillance requirements for fuel storage and transfer.

Objective: The objective is to verify safe and proper handling of the fuel elements, and to ensure that no fuel is diverted.

Specification:

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 quarterly.

Basis: This requirement prevents unauthorized movement of fuel and minimizes the potential for theft or diversion of a fuel element.

SR 4.7.1 requires quarterly 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.

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 k_{eff} 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.

Basis: 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, which 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.

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.

Basis: 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.

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.

Basis: 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.

4.7.5 Fuel in Shipping Containers:

(1) Verification that an outgoing fuel shipping container is approved shall be made prior to use.

Basis: 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.

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:

Applicability: This specification applies to the surveillance requirements for testing the reactor building intrusion alarms.

Objective: The objective is to ensure that the alarm system is always operable.

Specification:

(1) The Reactor Building Alarm system shall be tested monthly.

Basis: Experience has shown that the intrusion system works well and a monthly check should be more than adequate to detect the rare failure.

4.8.2 Explosive Material at the Reactor Facility:

Applicability: This specification applies to any explosive materials that may be brought into the facility for the purpose of being neutron radiographed.

Objective: The objective is to maintain safe handling of explosive materials at all times and to minimize any harm in the event of an unplanned function.

Specification:

(1) Administrative controls that implement the requirements of LCO 3.8.2 shall be reviewed annually.

Basis: 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 destination.

5.0 DESIGN FEATURES

5.0 SITE AND FACILITY DESCRIPTION

5.0.1 A steel, locked perimeter fence surrounds 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.

5.0.2 The minimum distance from the center of the reactor pool to the boundary of the exclusion area fencing is 50 feet.

5.0.3 The principal activities carried on within the exclusion area are associated with the operation of the reactor.

5.0.4 The reactor is housed in a steel building.

5.0.5 Reactor building ventilation is achieved by gravity ventilators located on the roof of the building.

5.0.6 An alarm system detects unauthorized entry into the reactor building.

5.1 REACTOR COOLANT SYSTEM

5.1.1 The reactor is cooled by the pool water which circulates by natural convection.

5.1.2 The pool water is cooled by a pumped cooling system consisting of a primary and secondary loop.

5.1.3 The water purity is maintained by a mixed bed demineralizer.

5.2 REACTOR CORE AND FUEL

5.2.1 Reactor Core:

(1) The reactor core shall consist of standard TRIGA fuel elements, 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 are the same as the fuel elements.

(4) Core and fuel design ensures 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.

5.2.2 Reactor Fuel:

The TRIGA fuel elements used in the ARRR have the following nominal characteristics at fabrication:

Fuel alloy:	uranium-zirconium hydride
Enrichment:	≤ 20 wt % U-235
Cladding:	Aluminum or Stainless steel
Fuel matrix:	8 wt. % (aluminum), or ≤ 20 wt.% (stainless steel)

5.2.3 Control Rods:

(1) The reactor has three control rods: safety, shim, and regulating.

(2) Each control rod contains boron carbide as a neutron poison, sealed in an aluminum or stainless steel tube.

5.3 FISSIONABLE MATERIAL STORAGE

5.3.1 The fuel storage pits located in the floor of the reactor room 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.

5.3.2 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 is stored in a geometric array where k_{eff} is ≤ 0.8 for all conditions of moderation and reflection using light water.

5.3.3 Fuel may be contained in an approved fuel shipping container within the limitations provided in LCO 3.7.4 and LCO 3.7.5.

5.3.4 A fuel handling tool is used for transferring fuel elements of low radioactivity between the storage pits and the reactor. A shielded fuel transfer cask is used for the transfer of highly radioactive fuel elements between the storage pits and the reactor.

5.4 EXPERIMENTAL FACILITIES

5.4.1 Neutron Radiography Facility:

(1) The beam tube consists of a two-section tapered tube having a rectangular cross section.

(2) The upper and lower sections of the beam tube are equipped with a fill and drain line.

(3) All components contacting the pool water are fabricated from aluminum or stainless steel.

(4) The beam catcher shield consists of a movable radiation shield.

5.4.2 Thermal Column:

(1) The thermal column is 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 is positioned remotely on steel locating pins immediately adjacent to the reactor core.

(3) Slotted beams are 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.

5.4.3 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 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 if the licensed operator is not a SRO. 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).

(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 ≥ $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, Sections 4-6.

6.2 REVIEW AND AUDIT

6.2.1 Reactor Safeguards Committee (RSC)

The Reactor Safeguards Committee shall be comprised of least 4 but not to exceed 5 voting members knowledgeable in fields which relate to nuclear safety. The members are appointed by the President of Aerotest of Operations, Inc. to serve one year terms. One of these voting members will be from the Board of Directors of Aerotest Operations, Inc. and will serve as the Chair. If the Chair is unable to attend one or a number of committee meetings, then the Chair may designate a committee member as Chair pro tem that is not part of operating staff. A corporate officer (President or Vice-President), Radiological Safety Officer, and Reactor Supervisor shall be ex-officio members. Of these ex-officio members, the Reactor Supervisor and Radiation Safety Officer or their designee have voting privileges. At least one appointed voting member should be from outside Aerotest Operations Inc. (and not a member of operating staff).

6.2.2 RSC Charter and Rules

The operations of the RSC shall be in accordance with this charter, which include provisions for:

(1) Meeting Frequency: The RSC shall meet annually at intervals not to exceed 15 months. (Note: The facility license requires a meeting at least once per year and as frequently as circumstances warrant consistent with effective monitoring of facility activities);

(2) Quorum: A quorum is comprised of not less than 4 of the voting membership or their designee where the operating staff does not constitute a majority;

(3) Voting Rules: On matters requiring a vote, if only a quorum is present a unanimous vote of the quorum (4 voting members) is required; otherwise a majority vote is required;

(4) Subcommittees: The Chair may appoint subcommittees comprised of members of the RSC including ex-officio members to perform certain tasks. Subcommittees or members of the RSC may be authorized to act for the committee; and

(5) Meeting Minutes: The Reactor Supervisor will act as recording secretary. It will be the responsibility of the Reactor Supervisor to prepare the minutes which will be distributed to the RSC including the President of Aerotest Operations, Inc. within three months. The RSC will review and approve the minutes of the previous meetings. A complete file of the meeting minutes will be maintained by the Reactor Supervisor.

6.2.3 RSC Review Function

The review responsibilities of the Committee or a designated subcommittee shall include, but are not limited to the following:

(1) Review and evaluation of determinations of whether proposed changes to equipment, systems, tests, experiments, or procedures can be made under 10 CFR 50.59 or would require a change in Technical Specifications or license conditions;

(2) Review of new procedures, major revisions of procedures, and proposed changes in reactor facility equipment or systems which have significant safety impact to reactor operations;

(3) Review of new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity;

(4) Review of proposed changes to the Technical Specifications and U.S. NRC issued license;

(5) Review of the Aerotest Operation radiation protection program;

(6) Review of violations of Technical Specifications, U.S. NRC issued license, and violations of internal procedures or instructions having safety significance;

(7) Review of operating abnormalities having safety significance;

(8) Review of reportable occurrences listed in Section 6.6.1 and 6.6.2 of these TS; and

(9) Review of audit reports.

6.2.4 RSC Audit Function

The audit function shall include selective (but comprehensive) examination of operating records, logs, and other documents. Discussions with cognizant personnel and observation of operations should be used also as appropriate. In no case shall the individual immediately responsible for an area perform an audit in that area. Audits shall include but are not limited to the following:

(1) Facility operations, including radiation protection, for conformance to the Technical Specifications, applicable license conditions, and standard operating procedures: at least once per calendar year (interval between audits not to exceed 15 months);

(2) The results of action 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);

(3) The retraining and requalification program for the operating staff: at least once every other calendar year (interval between audits not to exceed 30 months);

(4) The reactor facility emergency plan and implementing procedures: at least once every other calendar year (interval between audits not to exceed 30 months); and

(5) The reactor facility security plan and implementing procedures: at least once every other calendar year (interval between audits not to exceed 30 months).

Deficiencies uncovered that affect reactor safety shall immediately be reported to the RSC Chair and President of Aerotest Operations, Inc. and. A written report of the findings of the audit shall be submitted to the RSC within 3 months after the audit has been completed.

6.2.5 Audit of ALARA Program

The Chair of the RSC or designated alternate (excluding anyone whose normal job function is associated with ARRR(98) operation and radiation safety) shall conduct an audit of the reactor facility ALARA program annually. The auditor shall transmit the results of the audit to the RSC at the next scheduled meeting for its review and approval.

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 designated corporate officer, 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.

(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:

Written procedures shall be approved by Reactor Supervisor or his alternate and reviewed by RSC 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 Reactor Supervisor and RSC.

(2) Substantive changes to the previous procedures shall be made effective only after documented review and approval by the Reactor Supervisor or his alternate and review by the RSC.

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 alternate.

6.5 EXPERIMENTS

6.5.1 Experiment Review and Approval:

(1) All new experiments or class of experiments shall be approved in writing by the Reactor Supervisor or his designated alternate and reviewed by the RSC 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 Reactor Supervisor and other 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 approved in writing by the Reactor Supervisor or designated alternate and review by the RSC.

(2) Minor changes that do not significantly alter the experiment may be approved by the Reactor Supervisor or designated alternate.

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 his designated alternate, Corporate Officer, and RSC Chair

(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 and Corporate Officer 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.

(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 ≥ $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.

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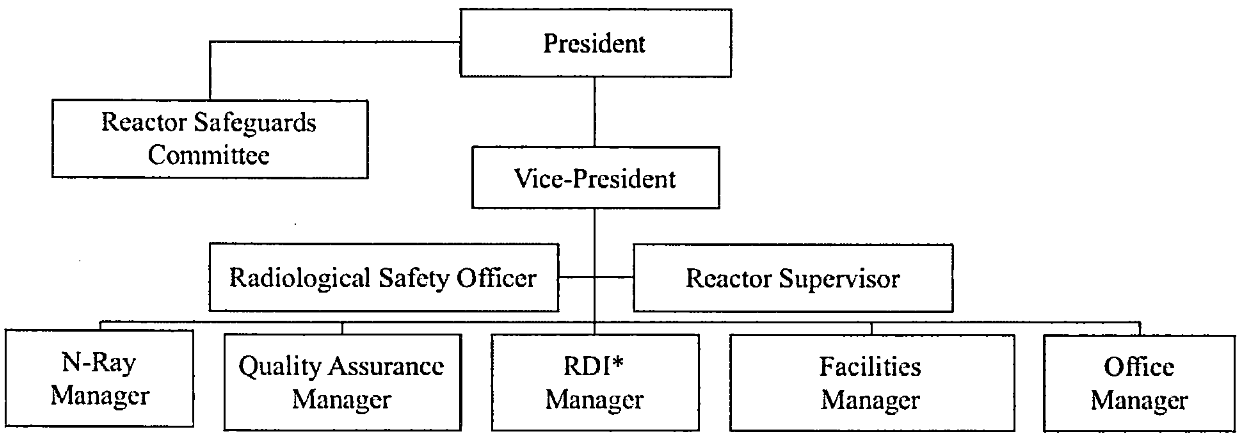
Figure 6.1
Aerotest Operations, Inc.
Organization Chart



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