Safety Analysis Report
for the

RHODE ISLAND NUCLEAR SCIENCE CENTER

Reactor

Operated by the

Rhode Island Atomic Energy Commission

License No. R-95

Docket No. 50-193

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CHAPTER ONE

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1.0 THE FACILITY

1.1 Introduction

The Rhode Island Nuclear Science Center (RINSC) open pool reactor is owned and operated by the State of Rhode Island. The reactor facility is located in Narragansett, Rhode Island on the University of Rhode Island Narragansett Bay Campus (NBC).

As originally installed, the reactor and support systems built by General Electric Company were adequate for operation at one (1) MW thermal (t) under license number R-95 issued on July 21, 1964. At present, an amendment issued on September 10, 1968 permits operation up to a maximum of two (2) $MW_{(t)}$. The Nuclear Regulatory Commission (NRC) conversion order to switch from high to low enriched uranium fuel was issued on March 17, 1993 following approval of the revised Safety Analysis Report (SAR).

This SAR is submitted pursuant to 10 CFR 50.64 by the Rhode Island Atomic Energy Commission to apply for a twenty-year license renewal. The analyses required for the preparation of this report have been a joint project of the Reduced Enrichment for Research and Test Reactor (RERTR) group at Argonne National Laboratory and the staff at the Rhode Island Nuclear Science Center (RINSC). The Rhode Island Atomic Energy Commission is responsible for the contents of this report.

1.1.1 Location of the Facility

The RINSC is located on a three-acre site situated on the NBC off South Ferry Road, in Narragansett, Rhode Island 02882. This three-acre parcel, listed in the Town of Narragansett tax assessor's office as Map N-C, Lot 7, is leased from the Rhode Island Department of Higher Education. The reactor building and the NBC are located on a portion of a 27-acre former military reservation originally called Fort Kearney.

1.1.2 Purpose of the Facility

The Rhode Island Legislature created the Rhode Island Atomic Energy Commission under the General Laws of Rhode Island, which states (in part), "to contract for, construct and operate a nuclear reactor within the state for the purpose of research, experimentation. . ., to co-operate with and make available, under proper safeguards, the use of said reactor by the colleges, universities and industries of this state."

Today, the Rhode Island Nuclear Science Center (RINSC), after more than 38 years of successful operation, provides research reactor operation services to both the research and industrial communities.

1.2 General Description of the Facility

1.2.1 Introduction

The reactor was initially licensed to operate at one (1) $MW_{(t)}$. The original design of the reactor permits later conversion to five (5) $MW_{(t)}$ operation after specific NRC review and approval. It operates with forced-circulation cooling, with optional natural convection cooling at power levels below 0.1 $MW_{(t)}$. It is presently licensed to operate at two (2) $MW_{(t)}$.

In general, the facility provides six beam tubes for neutron experiments, a thermal column for thermal neutron use, gamma ray experiment facilities and two pneumatic tube systems for activation analysis. Laboratory support services are also made available for researchers.

The reactor facility is composed of five basic systems: (1) the pool and biological shielding; (2) the reactor core, core suspension, drives, and drive shafts; (3) the controls and instrumentation systems; (4) the experiment facilities; and (5) the process and cooling systems.

1.2.2 Reactor

The reactor is a heterogeneous open pool, water-cooled (natural and forced convection mode) type, with capability of being operated at 5 $MW_{(t)}$. The fuel is LEU and is reflected with a combination of graphite and beryllium. The reactor core is located near the bottom of a 32-foot deep, aluminum-lined concrete pool. Normal operation is with the core in the high power section (Figure 1-1).

The core consists of fuel elements in a rectangular array, surrounded on four sides by reflector elements. Four safety control blades and a servo-actuated regulating blade control the reactivity. The control blades move vertically within a pair of shrouds that extend the length of the core. Core elements are contained in a grid box that is enclosed on four sides to confine the flow of cooling water between elements. A suspension frame supports the grid box, core, and the drive mechanisms.

With the award of a Department of Energy (DOE) grant, work began in August 1986 to convert the fuel system from 93% enrichment UAl_x fuel to a less than 20% enrichment U_3Si_2-Al fuel. As part of the conversion, priority was given to designing the core for greater flux at existing power levels. Also, consideration was given to ensuring that the new core would be capable of operating at power levels up to 5 $MW_{(t)}$.

Direct visual and mechanical accesses to the core and mechanical components are available from the top of the pool for inspection, maintenance, and fuel handling. The pool water provides adequate shielding of personnel standing over the pool. The mechanical components of the bridge and frame are of a design that permits the structure to be mechanically moved along a rail system mounted on the top of the pool walls. The mechanical components are described in greater detail in Chapter 4, Reactor Description.

The pool area is lined with an aluminum liner that aids in maintaining water purity and minimizes water leakage into the concrete. Coolant piping protrudes through the concrete shield structure to allow connections to the core. Major experiment facilities converge toward the core, and afford ample opportunity for the simultaneous performance of a number of different experiments.

The heart of the reactor consists of a core supported by a suspension frame, in an open pool of water. The frame is bolted to a bridge, movable from one end of the pool to the other by means of a hand crank that, when turned, moves the wheel-mounted bridge along rails provided at the top of the concrete structure.

The reactor core sits on a 7 x 9 grid plate with the four corner grid positions occupied by the suspension frame support posts. These support posts connect the grid plate/box to the reactor bridge that spans the open pool. The support posts are water-filled, providing a convenient neutron detector location. The grid plate/box is suspended about 8 meters (26.3 feet) below the pool water surface.

This grid plate is installed at the bottom of a grid box whose four sides are enclosed. The top of the grid box is open to the pool and the bottom connects to an enclosed plenum for coolant flow. The grid box also contains two permanently installed shrouds in which four boral control blades move.

The stainless steel regulating blade in the reflector region of the LEU core can be relocated. While some grid positions are shown vacant for clarity in Figure 1-2, during operation each grid position must contain either a fuel element, a reflector, or an irradiation basket. Otherwise, the coolant flow would by-pass the active core through the vacant grid position. A three-inch lead thermal shield, with cooling piping, is positioned in the pool between the grid box and the thermal column.

Surrounding the reactor, at the main floor level, is approximately [redacted] of reinforced concrete. The basic purpose of the massive concrete structure is to provide shielding for personnel working in the reactor building. The massiveness and also the location of the structure on a heavy concrete substructure (a concrete gun mount from the former Fort Kearney), provides excellent protection for the reactor core against natural phenomena.

The reactor control room is located at the pool level. Two outside walls of the control room are the confinement building walls and two interior walls are standard 2 x 4 wood construction. The reactor control console is located in the control room and manages all control blade movements and contains devices that provide interlocks, scrams, and systems indication. It processes and displays information on control blade positions, power level, and coolant system parameters. The reactor instrumentation includes chart recorders to display information graphically and the data can also be saved for future reference.

1.2.3 Reactor Confinement Building

The reactor confinement building is a reinforced concrete building 69 feet long by 62 feet wide by 51 feet high that houses the open pool research reactor as shown in Figure 1-3.

The reactor confinement building exhaust system is designed to provide a slightly negative pressure with respect to surrounding areas to prevent the release of any radioactive gases that are produced through any penetration other than through the 115 foot exhaust stack. The off-gas systems remove radioactive gases from experiment facilities and helps to maintain concentrations of gases below limits of 10 CFR 20 in the reactor confinement building.

The biological shield surrounding the pool is reinforced concrete that varies in thickness from approximately [redacted] at the reactor base, to approximately [redacted] at the pool level. The biological shield is designed to keep personnel exposures as low as reasonably achievable and to protect the reactor from natural phenomena and has been shown to be effective by the results of surveys taken during the operational history of the facility.

The RINSC reactor is housed in a building specifically designed for reactor operation. It includes the many systems needed to support this type of operation. The RINSC facility consists of one building which houses the reactor and support areas. The confinement building (reactor building) is a concrete building normally maintained under a negative pressure. Normal operation results from reactor room exhaust air being removed by the reactor room exhaust blower, through the butterfly valve at the building wall penetration, and into the reactor exhaust stack. Confinement air intake comes through a butterfly valve at the building wall penetration into the building heated air duct. The two butterfly valves close air tight on activation of the emergency exhaust system. Additionally, off-gas is removed from experiment facilities by an off-gas blower and filter system. The blower removes gases from the thermal column, beam port drain and vents, etc. and discharges into the reactor room exhaust piping. A pneumatic system blower discharges its gas removal into the off-gas discharge line. The off-gas and pneumatic systems operation are more fully described in Chapters 9 and 10. A dilution air blower provides additional airflow and discharges into the stack at its base. Normal building differential pressure (Δp) is measured at two locations, (1) in the control room, and (2) in the adjacent lab wing outside of confinement.

The pool is divided into three interconnected sections: (1) the high-power section, (2) the storage section, and (3) the low-power section. A movable gate can be used to separate the sections and allows for individual draining of the high and low power sections. The reactor core is moved to the high-power section and coupled to the coolant lines for operation in power ranges above $0.1MW_{(t)}$ using forced circulation. For operation in power ranges of $0.1MW_{(t)}$ and below, the reactor may be operated in any of the sections using natural convection cooling.

As spent fuel elements are generated, they are placed in the fuel storage racks located in the storage or low-power sections of the pool. The spent elements may be used as a source of gamma radiation.[1]

The facility has a radiation monitoring system consisting of area monitors with audible/visible warnings to prevent personnel from inadvertent exposure to high radiation levels. Beam port override shutter controls and position indication are in the control room. Some beam port experiment areas have individual shutter operating controls and indications. The pneumatic system sample stations have lead receiving boxes and radiation monitors for personnel protection. Manual reactor scram buttons are located in the control room and on the instrument bridge over the pool.

The facility contains proven and reliable electrical, water, makeup water and waste water systems. In addition, the facility has fire detection and suppression systems, intercom system, radiation monitoring systems, security system and an emergency evacuation system. Primary and secondary cooling systems and a primary water cleanup system are used for heat removal and pool water purification.

1.2.4 Shared Facilities

The RINSC facility has a shared heating system with the adjacent University of Rhode Island Center for Atmospheric Chemistry Studies (CACS) building. This building supplies heat (circulated hot water) directly into the existing reactor building heat piping. The hot water heating supply is generated by the CACS dual unit gas-fired boiler system.

The RINSC provides de-mineralized water to the CACS. The makeup de-mineralizer system for the reactor has sufficient capacity to allow a portion for laboratory use in the CACS. The CACS laboratories can store additional de-mineralized water in case the RINSC shuts down its makeup system for resin recharge. The CACS de-mineralized water demand is small and may be isolated, if necessary, from the loop through the piping system.

1.2.5 Laboratories

The facility provides a variety of laboratories operating under an Agreement State license issued by the Rhode Island Department of Health. The facility houses several chemistry laboratories. In addition to standard equipment such as air, gas, vacuum, and water lines, some laboratories are equipped with hoods that exhaust through absolute particulate and charcoal filters.

1.3 Comparison with Similar Facilities

The LEU fuel, described in chapter 4, has been developed and tested and is now being used in the RINSC reactor at its present operating licensed power level of 2 $MW_{(t)}$. The operation and accident conditions of the RINSC are no greater than those of similar reactors using the same fuel systems, and therefore present no undue risk to the health and safety of the public.

The LEU fuel, control-blade drives, control blades, and experimental systems are similar to systems in other reactors (University of Massachusetts at Lowell, etc.). These items have well established operating experience.

1.4 Summary and Conclusions on Principal Safety Considerations

1.4.1 Nuclear

The analyses presented in this report demonstrate that the RINSC reactor has been designed and constructed and can be operated, as described herein, without undue risk to the health and safety of personnel in the facility and the general public.

The approach taken in this document to demonstrate the safety of the RINSC reactor is to show that:

- The RINSC reactor fuel and instrumentation and control systems are of proven design, based on past operating experience of systems with the same or similar designs, which have been approved for operation by U.S. Government agencies;
- The operating and accident conditions of the RINSC reactor are no greater than those of other similar reactors using the same fuel systems, and therefore present no undue risk to the health and safety of the public;
- The RINSC has been maintained and its components and systems have been updated and/or replaced as needed;
- The RINSC has safely operated for more than 38 years.

The RINSC reactor fuel, control-rod drives, control rods, and experiment systems are similar to many other systems used throughout the United States. These items have well-established operating experience. Conversion of the RINSC reactor to LEU fuel was accomplished in 1993. The reactor operates at a nominal steady-state power of 2 $MW_{(t)}$.

Abnormal conditions or postulated accidents discussed in this report (See Chapter 13) include:

- Maximum Hypothetical Accident (MHA);
- Reactivity insertion;
- Loss of coolant;
- Loss of heat-removal system;
- Fuel cladding failure, and;
- Pyrotechnic detonation.

The limiting fault condition (i.e., the Maximum Hypothetical Accident), which assumes failure of fuel cladding and an air release of all the fission products from a single fuel plate, will result in radiation doses to RINSC personnel and the general public for both thyroid and whole body that are acceptable. Chapter 13 contains a detailed discussion of this accident scenario.

Radiation exposures to personnel working in the RINSC from both direct and airborne radiation during normal operation have been analyzed. In addition, actual radiation levels have been measured. This analysis and measurements show that the highest exposures occur when personnel are working on the beam floor when the reactor is operating. Under these conditions, personnel will be subjected to a maximum radiation field of less than 0.5 mrem/hr. Chapter 11 contains the personnel exposure analysis. All personnel entering radiation areas will be closely monitored, exposures kept as low as possible, and in no case will they be allowed to exceed the 10 CFR Part 20 guidelines.

The effects of Ar^{41} and N^{16} concentrations during normal operation of the reactor have also been evaluated for both RINSC personnel and the general public. These isotopes result in exposures of only a few mrem/yr to RINSC personnel. Their release to the atmosphere, via the exhaust stack, results in a maximum down wind concentration below the 10 CFR Part 20 guidelines for unrestricted areas, see Chapter 11 and Appendix A for analysis.

Radiation-monitoring equipment has been installed at key locations to monitor radiation levels and to sound alarms if preset values are exceeded.

1.5 Summary of Operations

The Rhode Island Atomic Energy Commission has been operating the only state-owned research reactor for more than 38 years. As one of the higher flux reactors (when compared with other University Research Reactors [URR]), the RINSC has significant potential to carry out a wide range of research and educational programs. Since the RINSC operates the only nuclear reactor in the State, it is responsible for providing tours, briefings and training to high schools, college classes and the general public from a large geographical area.

The operating schedule makes the reactor available for full power operation during most of the year on a daily (one shift) basis. Reactor utilization by outside users is fostered and encouraged.

1.6 Compliance With the Nuclear Waste Policy Act of 1982

The RINSC has an agreement with EG&G Idaho, Inc. under subcontract No. C88-101958 (DE-AC07-76ER03488) for reactor fuel assistance AR-80-88. The contract is for the disposal of spent fuel. A copy of the contract is enclosed herein and satisfies the requirements of the Nuclear Waste Policy Act of 1982.

1.7 Facility Modifications and History

1.7.1 Introduction

Historically, RINSC had been utilized for research in environmental chemistry. Since the facility is located at the University of Rhode Island Graduate School of Oceanography, it enjoyed a large user group of environmental researchers. Graduate students and several research groups employed neutron activation analysis extensively. The RINSC has several laboratories, a clean room and a mass spectrometer room. A Center for Atmospheric Chemistry Studies building adjacent to the RINSC provided ready access for faculty, students and others doing environmental monitoring.

The Physics Department of the University of Rhode Island developed an extensive neutron scattering program at RINSC. This program attracted physicists from several different countries and many doctoral students and graduate students have obtained their degrees based on research conducted at the RINSC. In addition to the high flux necessary for this work, RINSC offers these students the opportunity to obtain extensive daily neutron beam time on three instruments. Availability of reactor services to a broad spectrum of users stimulates creativity. As in the case of chemists, physics students get much hands-on experience working with nuclear technology. Such opportunities are not as readily available at the federal reactors. Department of Energy grants have been received to upgrade reactor systems. This support will ensure the continued strong development of the basic research programs at the RINSC.

The RIAEC has developed a research plan that supports a wide range of bio-medical science research. Due to the unique nature of the RIAEC's organizational structure, it has been able to attract researchers from across the United States and internationally. A mandate to serve all research endeavors enables the facility to attract a broad spectrum of users as evidenced by the new initiatives in bioscience and medicine that are being developed by private companies and collaborations of government and educational institutions. Because the facility is not part of any one educational institution, it is not tied to one specific program. The RIAEC has demonstrated the ability to rapidly enter into cooperative agreements with a broad spectrum of users and provide support to all researchers.

Some example of current research initiatives include:

Boron Neutron Capture Therapy:

Neutron beams at URRs have been used in clinical trials to study the efficacy of boron neutron capture therapy for the treatment of brain tumors. To minimize collateral damage to healthy tissues in such procedures, an advanced epithermal neutron filter has been designed by Dr. Karl Ott of Purdue University for installation in the thermal column of the RINSC reactor. Epithermal neutrons are expected to penetrate healthy regions of the brain with minimal collisions and deposit the bulk of their energy in malignant cells. Dr. Ott's filter design is predicted to significantly improve performance over previous beams. It will provide a 5 to 7 fold enhancement of the dose-rate of the reactor beam specifically in the epithermal neutron energy range. Concomitantly, a great reduction in the intensity of associated gamma rays and of high energy and thermal neutrons should be achieved. For cancer therapy, these results will produce increased beam penetration and decreased side effects due to radiation damage to normal tissues around tumors.

Dr. Ott has been working closely with Argonne National Laboratory (ANL) in the design of the filter and he has signed a fifteen-year lease with the state of Rhode Island for use of the RINSC facility. He has also contracted out work on an improved boron based compound and is collaborating with a team from Brown University (BU) that is developing a gadolinium based compound for use in the treatment of brain cancers.

Assay of Tissue and Fluids Utilizing Neutron Activation Analysis (NAA) of Stable Isotopes:

Over the past several years, the RINSC has worked with BioPal, Inc. in the development of a new application for NAA that has the potential to generate significant usage of even the lowest flux URR's. This technique is a new generation, high-precision alternative to traditional, radio labeled research products that provides a state-of-the-art assay service based on NAA technology. There are several major advantages to this approach which include:

- Non-radioactive - Stable isotopes are used as labels. No NRC license or expensive laboratory equipment required by users.
- No shipping of radioactive material is required.
- High Sensitivity - Significant enhancement in sensitivity over traditional radioimmunoassay.
- Multiple Labels - A large number of stable isotopes can be used.
- Archiving and Re assay - NAA is non-destructive.
- High Throughput - Ultimately thousands of samples/day can be processed at one reactor.
- Low Cost - Recent improvements in detector technology and electronics have improved throughput efficiency and sensitivity.
- Platform Technology - many research and clinical assays can be adapted to this technology.

This technique is currently utilizing stable labeled micro-spheres and stable markers for a number of deposition studies and for DNA synthesis. Custom NAA services are available. Stable isotope labeling kits are being developed to measure antibodies, peptides, and other cellular probes. Methods are being tested that focus on cancer detection, measurement of extra-cellular space and blood volume and protein-bound iodine.

Advanced Isotope Research:

Many URRs have active programs to produce radioisotopes for scientific and industrial applications. Some of these isotopes are used as tracers in chemical, nutritional, and environmental studies, while others are used in the synthesis of radiopharmaceuticals. Radiopharmaceuticals are used both as diagnostic and as therapeutic agents. Significant funding will be required to develop fully functional radioisotope production.

Environmental Research:

The significant environmental expertise of RINSC researchers has resulted in the formulation of a commercial venture called Microinorganics, Inc. This company has two laboratories at RINSC and has become actively involved in environmental sampling work for state and federal agencies. Current projects include sampling of the Providence River and Narragansett Bay for the Department of Environmental Management and the Environmental Protection Agency, trace metal work for the Narragansett Bay Commission and other projects associated with water supplies and landfills. This highly successful and environmentally important project was able to succeed because of the expertise resident in the researchers at the RINSC and the support of the facility.

1.7.2 Original Design and Construction

The RINSC has successfully been in operation for more than 38 years. Over the years, The State of Rhode Island has provided significant financial support toward improvement of the facility through a capital budget program and increased operating support. Recently, all four roofs of the RINSC facility have been replaced and the building exterior has been restored and painted. New security and fire alarm systems have been installed. A new central air conditioning system in the office wing was recently installed. The de-mineralized water system was replaced and an emergency core cooling system has been installed. Private funds were utilized to completely refurbish four laboratories. Underground radioactive waste storage tanks were removed and replaced by tanks located inside the facility. The Rhode Island Cancer Council provided a grant to equip the new biomedical laboratory. Dr. Karl Ott is currently evaluating bids to construct an extensive medical treatment building adjoining the reactor building. It is evident from these investments that there is a long-term commitment to the continued success of Rhode Island's only nuclear facility.

The following two sections detail the chronology of the amendment and changes to the facility Technical Specifications approved by the NRC and a brief listing of the facility modernization.

The listing highlights mostly new equipment replacements and facility system upgrades. Minor equipment upgrades (e.g., switches, charts, gauges, meters, etc.) are not included. Any work on safety items pertaining to the operation of the reactor are kept tract of by utilizing a work permit and QA/QC program.

Fuel replacements and graphite reflector replacements are not addressed in this section.

1.7.3 License Amendments

1.	Change #1 - Change delay tank to Aluminum	11-19-64
2.	Change #2 - Liquid radioactive waste goes to retention tanks	4-14-65
3.	Change #3 - Change heat exchanger to stainless steel, revise secondary coolant activity limit, change area monitoring system, and modify the requirement for HV scram	4-14-67
4.	Amendment #1 - Allow 2-MW operation	9-10-68
5.	Change #5 - Change inlet alarm from 110°F to 113°F	1-22-69
6.	Amendment #2 - Increase U^{235} from 4.0 Kilograms to 7.5 Kilograms	10-21-70
7.	Change #6 - Allow use of irradiated fuel in determining core reactivity	12-15-71
8.	Amendment #3 - Increase U^{235} from 7.5 Kilograms to 10.0 Kilograms	7-14-73
9.	Change #7 - Remove 30-element core size limit	5-21-73

	and restate blade drop times	
10.	Amendment #4 - Increase U^{235} from 10.0 Kilograms to 10.4 Kilograms	12-06-74
11.	Amendment #5 - Replacement of stack monitor	1-10-78
12.	Amendment #6 - Removes U^{235} core weight limit	10-02-78
13.	Amendment #7 - Changes to Security Plan	6-17-80
14.	Amendment #8 - Allows use of UAl_x or U^3O^8 fuel elements	7-31-80
15.	Amendment #9 - Changes to Security Plan	5-12-81
16.	Amendment #10 - Replaces specification for the primary water makeup and cleanup systems	3-02-82
17.	Amendment #11 - Allows use of different graphite reflectors	10-18-82
18.	Amendment #12 - Changes to liquid waste effluents	4-13-84
19.	Amendment #13 - Adds 1% shutdown margin	2-19-85
20.	Amendment #14 - Changes secondary pH upper limit from 7.5 to 9.0	5-20-85
21.	Amendment #15 - Redefine restricted area	10-27-87
22.	Amendment #16 - Clarify the exhaust sources and rates entering the stack	10-20-88
23.	Amendment #17 - Change order to LEU fuel	3-17-93
24.	Amendment #18 - Complete revision of T.S. to standards	3-09-94
25.	Amendment #19 - Delete requirement for daily measurement of secondary activity	1-19-95
26.	Amendment #20 - Clarify operation of the stack monitor	9-22-95
27.	Amendment #21 - Change RSC&RUC to NRSC	2-12-96
28.	Amendment #22 - Change to extend specific annual tests for 1996 only	12-18-96
29.	Amendment #23 - Remove Inlet temperature scram as a safety limit and limiting safety system setting; add pool temperature scram	6-17-97
30.	Amendment #24 - Re-word Airborne Eff., Emergency Exhaust System, Remove Inlet Temperature as a LSSS and minor syntax and reference changes	9-23-97
31.	Amendment #25 – Correct omission of Amendment #24	11-12-97
32.	Amendment #26 - Change bases of emergency generator system; change Director & RSO qualifications	11-22-99
33.	Amendment #27 – Extends the license expiration date To July 21, 2004	7-28-2000
34.	Amendment #28 – Change pool water level scram check Frequency	8-2-2001

1.7.4 Reactor Systems Modifications

1.7.4.1 LEU Conversion and Power Upgrade Program

As part of the LEU core design process, critical operating parameters for operation at 5 $MW_{(t)}$ were determined. The studies for the LEU conversion included calculations for operation at power levels up to 5 $MW_{(t)}$ and for the advanced core design. This was done to ensure that the process did not compromise the future capabilities of the reactor. The studies for the LEU conversion contain all core information necessary for a safety analysis supporting an upgrade to 5 $MW_{(t)}$.

The LEU conversion program consisted of several phases: design of the new core, completion of a Safety Analysis Report (SAR), shipment of the spent HEU fuel, modifications to reactor systems and new core loading and the test program. The power upgrade program consisted of adding an additional cooling loop and an emergency core cooling system, (ECCS not required for reactor operation at 2 $MW_{(t)}$ and below), and calculations to verify actual system performance with model predictions. This phase is described in further detail in Chapter 5.

The reactor shutdown time for conversion of the new fuel was utilized to install a new 3-$MW_{(t)}$ cooling tower and new wide range power level instruments and a larger secondary piping system which were provided through DOE Reactor Instrumentation Program grants. Following conversion to LEU, efforts continued to develop the systems necessary to upgrade to a proposed 5 $MW_{(t)}$ operation. An Emergency Core Cooling System (ECCS) was designed and installed in 1994-1995. A second primary loop and associated secondary system were installed in 1995 and 1996. In addition, the primary pump diaphragm isolation valves were replaced with low d/p butterfly valves. As part of the primary flow upgrade, new components with a larger range were added to the reactor flow sensing circuits to monitor the increased flow capability.

Throughout the upgrade program, valuable assistance was received from Argonne National Laboratory in conducting extensive thermo-hydraulic, nucleonic, and accident analysis studies that refined the new system operating characteristics. Valuable funding support was received from the Department of Energy throughout this endeavor.

1.7.4.2 LEU Conversion - Design Phase

The design phase focused on six basic criteria and objectives of the conversion program. These were:

- Use the standard design DOE fuel plate;
- Fuel bum-up greater than the 14% of the HEU fuel;
- Optimize the thermal flux in the beam ports;
- Include a flux trap for small sample irradiation;
- Maintain 5 $MW_{(t)}$ capability; and
- Cost similar to a replacement HEU core.

The neutronic core design was performed using the standard LEU fuel plate provided by DOE. This plate is thinner and contains considerably more uranium-235 than the HEU plate [redacted] Because of the heavier fuel loading of the LEU plate, a major concern was that the core might become so small that the control blades might lose their effectiveness. Many core configurations were considered.[1] These studies included consideration of:

- 18 fuel plate elements;
- 22 fuel plate elements;
- several fuel element arrangements;
- graphite and beryllium reflectors;
- relocation of the regulating blade position; and
- Use of a stainless steel regulating blade.

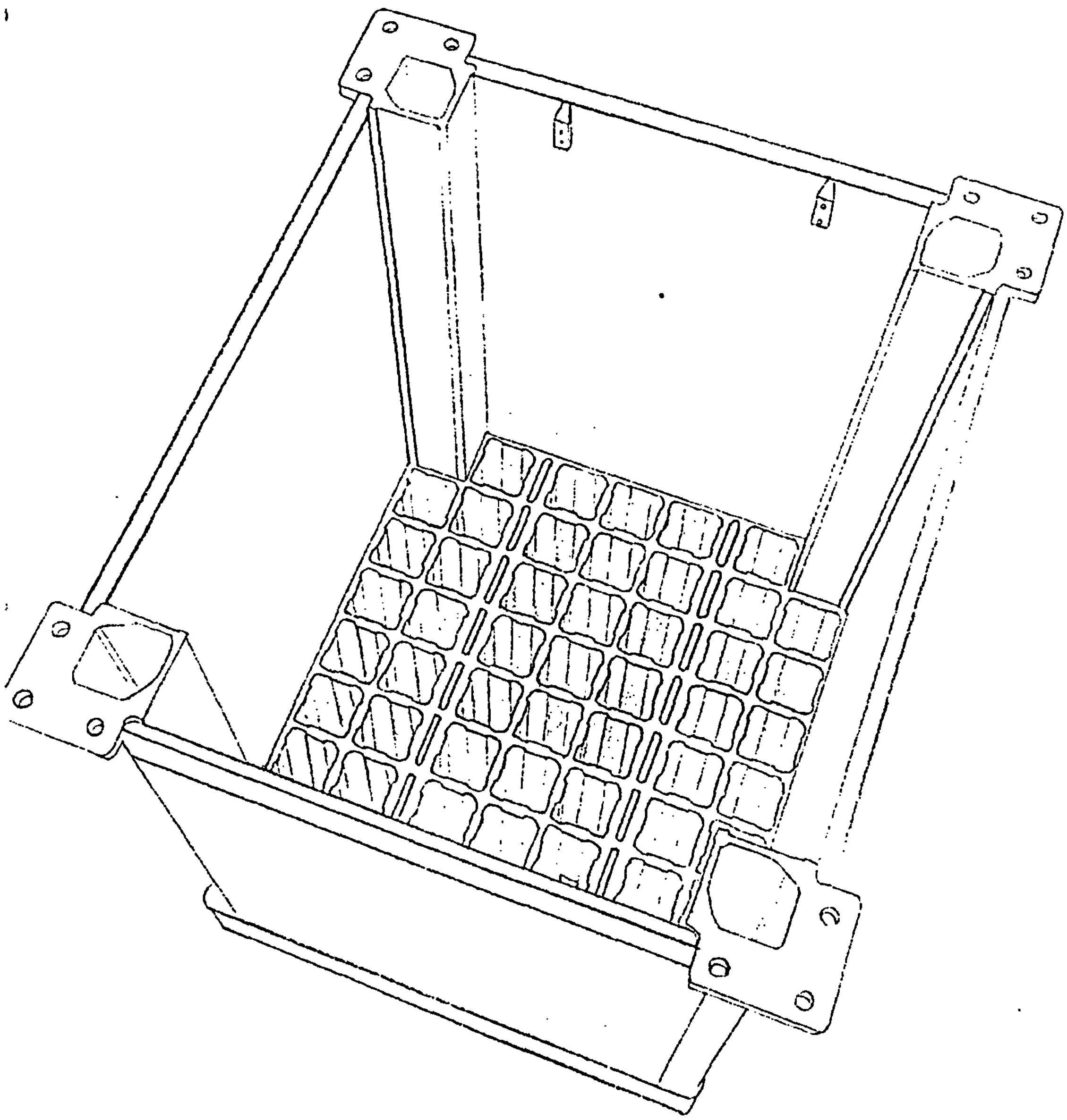
The final LEU installation resulted in a 22-plate fuel element, a stainless steel regulating blade with a relocated position in the core, and various fuel and reflector arrangements. The facility now has satisfactorily operated with the LEU fuel since September 24, 1993.

(1) DiMeglio, A.F., Matos, J.E., and Freese, K.E.: Conversion, Core Redesign and Upgrade of Rhode Island Atomic Energy Commission Reactor, Proceedings of the 1987 International Meeting on Reduced Enrichment for Research and Test Reactors, Buenos Aires, Argentina, 28 September - 10 October, 1987.

1.8 FACILITY MODERNIZATIONS AND HISTORY	DATE
Reactor Room Exhaust Isolation Valve - Installation	1966
Fire Sprinkler - South Basement Area	1968
Reactor Room Intake Isolation Valve	1969 & 1973
New Laboratory Building Addition	1971
Cleanup Demineralizer Tank-Replaced	1977 & 1978
Primary System Heat Exchanger-Replaced	1973
Fire Sprinklers - North Basement Area	1984
Reactor Room Crane Upgrade - (Analysis in 1987)	1988
Reactor Building Roof Surface-Replaced	1991
Reactor Room Air Intake Isolation Valve & Duct-Replaced	1992
Regulating Rod-Replaced	1993
Cooling Tower #1-Replaced	1993
Shared Heating System Installation	1993
LEU (Fuel & Beryllium Reflectors) Upgrade	1993
Emergency Core Cooling System	1995
Primary Pump Replace & Piping Upgrade	1996
Heat Exchanger #2-Installation	1996
Primary System Diaphragm Valves-Replaced with Butterfly Valves	1997
Makeup Demineralized Water System-Replaced	1997
Reactor Console Equipment	1997
Area Radiation Monitors-Replaced	1998
Fire detection and alarm system replaced	1999

Vehicle Barriers installed @ front & rear of confinement building	2003

[redacted]



Grid Box

Figure 1-2

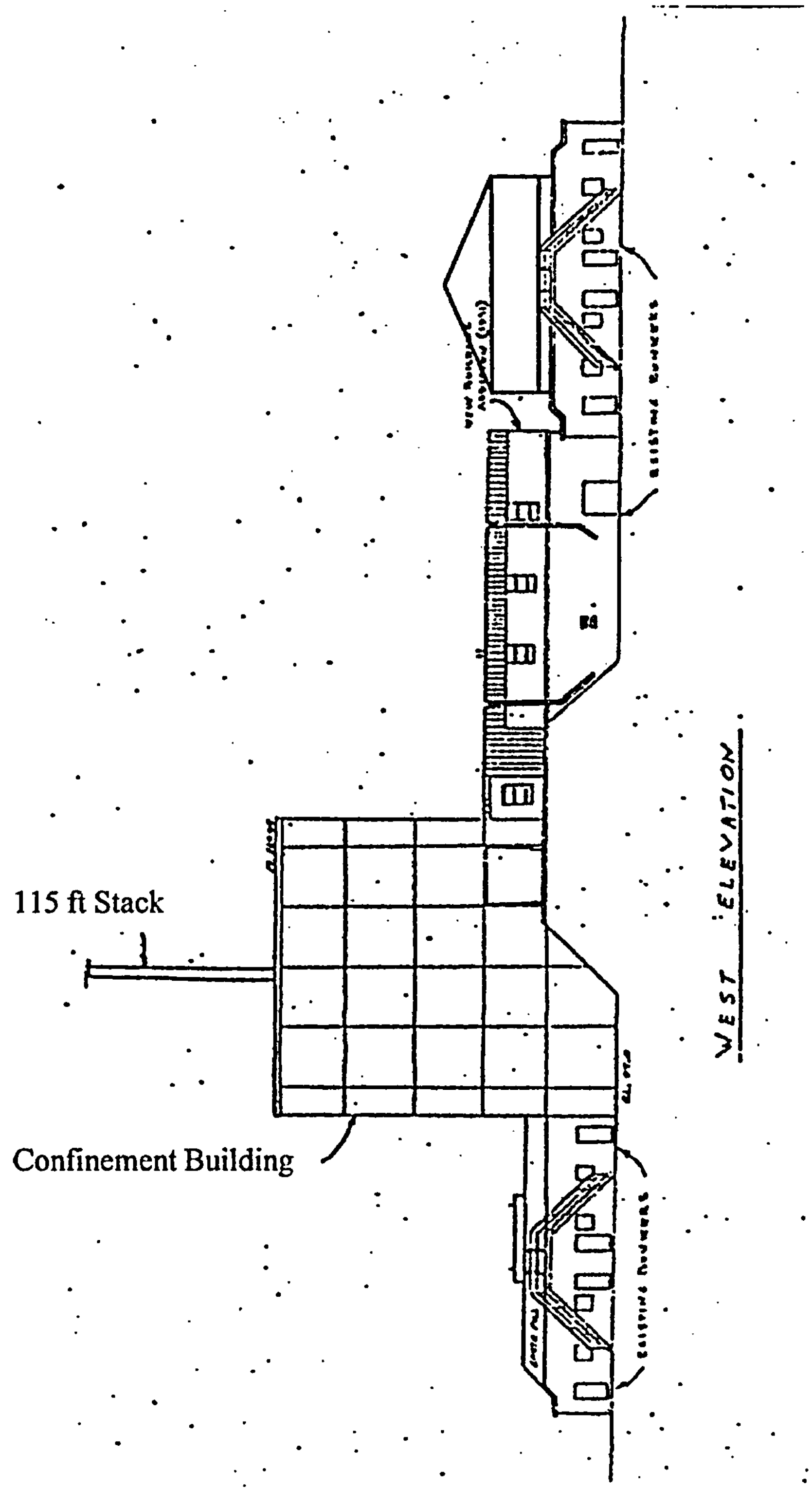


Figure 1-3

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2.0 SITE CHARACTERISTICS

This chapter discusses and describes the geographical, geological, seismological, hydrological, and meteorological characteristics of the reactor facility site and vicinity in conjunction with present and projected population distributions, industrial facilities, land use, and site activities and controls.

The conclusion reached in this chapter and throughout the SAR is that the site is well suited for the location of the facility when considering the relatively benign operating characteristics of the reactor, including the MHA.

2.1 Geography and Demography

2.1.1 Site Location and Description

The RINSC is located on a three-acre site in southern Rhode Island. Figure 2-1 shows the site relative to Narragansett Bay (scale: 1in = 2,000 ft). The three-acre parcel is listed in the Town of Narragansett tax assessor's office as Map N-C, Lot 7 and is leased to the RI Atomic Energy Commission from the RI Department of Higher Education. It is located in the Town of Narragansett in Washington County. The location of the reactor building and nearby URI Narragansett Bay Campus (NBC) buildings are depicted on Figure 2-2 (not to scale) and the reactor building is shown on the aerial plan (Figure 2-3, scale 1 inch = 400 feet). The reactor building sits atop a hill, about 70 feet above mean sea level, overlooking Narragansett Bay. The site area was originally a part of a 27-acre former military reservation (Ft. Kearney) that consists of old concrete bunkers and gun mounts. The area below the reactor building consists of concrete above dense soil. Figure 2-4 shows the location of the site as outlined on the FEMA flood map. Note the reactor building is sited at least 56 feet above the nearest flood elevation (V 10) of 14 feet.

2.1.2 Operational Boundary and Zone Area Maps

There are three areas of concern regarding the normal operation, safety, and emergency actions associated with the reactor facility: (1) the area within the operations Boundary, (2) the area within the site boundary, and (3) the Emergency Planning Zone (EPZ).

The operations boundary is defined as the reactor confinement building. The area within this boundary is a "restricted access" area over which the Director of the RINSC has direct authority and control of all activities, normal and emergency. There are pre-established evacuation routes and procedures known to personnel frequenting this area. The operations boundary is within the site boundary.

The site boundary is delineated on Figure 2-5. The minimum distance from the operations boundary to the site boundary is approximately 140 feet to the southeast of the RINSC facility.

The EPZ boundary is also the reactor confinement building. Emergency plans have been developed to ensure that prompt and effective actions can be taken to protect the public in the event of an accident.

2.1.3 Population Distribution

The Rhode Island Statewide Planning Office has provided the following data for the Town of Narragansett:

The 1990 census population was 14,985
The 1995 census population was 14,878
The 2000 census population was 16,361.

The area is well developed and no great changes in population are projected. Most of the surrounding areas are residential. Existing land use surrounding the facility is as follows:

Narragansett Bay Campus: Figure 2-2 shows the buildings on the campus that services 200-300 students and faculty;

Narragansett Industrial Park: located less than 1/4 mile west along South Ferry Road that houses several light industrial buildings and businesses;

West Passage Estates: a residential area to the north of South Ferry Road containing single family homes of about one (1) unit per 1/2 acre; and

Areas within one mile: scattered residential units and a small business strip along Route 1A to the west.

2.2 Nearby Industrial, Transportation And Military Facilities

2.2.1 Industry

There are no major industrial facilities in the Narragansett area that need be of concern from the safety standpoint. The area's economy is mostly based on tourism, with some small-town government jobs and most jobs concentrated in the infrastructure supporting tourism (e.g., restaurants, shopping, etc.). The Port of Providence and the Port of Davisville are the closest locations for shipping, with most water traffic entering Narragansett Bay on the East Passage (i.e., the east side of the Island of Jamestown).

2.2.2 Transportation

2.2.2.1 Highway Transportation

The Narragansett Bay Campus is accessed by South Ferry Road, a small road east of U.S. Scenic Route 1A which connects Narragansett to Saunderstown and Wickford (small towns to the north). A larger road, U.S. Route 1 which is a half mile to the west of

1A, is accessed by Bridgetown Road. U.S. Route 1 connects points south, such as South Kingstown and Westerly, to major cities to the north such as Warwick and Providence. Interstate Route 95 which is approximately 20 miles to the west can be accessed using Rhode Island Route 138 from U.S. Route 1 (See Figure 2-1).

2.2.2.2 Airports

There are two state airports within 20 miles of the facility. The closest is a seldom-used airport named Quonset State Airport in North Kingstown, approximately 10 miles to the north. Generally this airport used by the Air National Guard. The T. F. Green Airport in Warwick, approximately 20 miles to the north, is the main Rhode Island airport for passenger and commercial flights. There are two small private airports about 20 miles to the west, Richmond and Westerly Airports. Due to the distance, number of flights, or size of aircraft, these airports are not considered to present a credible hazard to the RINSC.

2.2.2.3 Water Transportation

The RINSC is located on the west side of the Narragansett Bay (West Passage). Light fishing craft, some cargo transportation and pleasure boating form most of the water traffic that is mostly seasonal. The main shipping traffic, to and from the Port of Providence, takes place on the area of the bay between Conanicut Island and Aquidneck Island (East Passage).

2.2.2.4 Rail Transportation

The only rail system nearby is the Amtrak Railway located in West Kingstown, approximately 6 miles to the west. The rail runs approximately north and south and handles passenger and shipping service from the south (Westerly) and into Massachusetts to the north. The RINSC is not on the notification list for any shipments of hazardous materials along this route.

2.2.3 Military Facilities

There is one military facility in the vicinity of Narragansett. The Naval Education and Training Center at Newport is the home for the Naval War College, the Naval Underwater Weapons Center and several school commands. While there are no active navy ships at the piers, several Coast Guard ships are home-ported at the base. There are no threats posed to the RINSC from this facility. There is also an Air National Guard facility at the Quonset Point Air Port

2.2.4 Analysis of Potential Accidents at Facilities

There are no nearby industrial, transportation, or military facilities with the potential of causing a credible accident (which could prevent a safe reactor shutdown or result in a release of radioactive material from the reactor facility) that would exceed the general public exposure limits of 10 CFR 20.

The basic design and structure of the facility provides significant protection for the reactor. The core is located near the bottom of a 32-foot deep, aluminum-lined concrete pool. This and the immense size of the biological shield coupled with its being constructed atop a military concrete bunker and gun mount provide excellent protection against natural phenomena that could result in damage to the reactor core.
The front and rear of the confinement building is protected at a minimum of 50 feet with vehicle barriers of the bollard type at the front and the jersey type at the rear.

2.3 Meteorology

2.3.1 General and Local Climate

2.3.2 Site Meteorology

2.3.2.1 Temperatures

Figure 2-9 shows average temperatures, maximum and minimum, and a two-year-in-ten-year maximum and a minimum on a month by month basis. During the reporting period, the highest temperature was 93°F, occurring in July. The lowest recorded temperature was -8°F, occurring in January.

2.3.2.2 Precipitation

Figure 2-9 shows the average precipitation, maximum and minimum, and a two-year-in-ten-year maximum and minimum on a month-by-month basis. The average annual precipitation is about 48.0 inches per year. The highest monthly averages occur in November and December.

2.3.2.3 Humidity

2.3.2.4 Winds

Figure 2-9 shows the data relative to Washington County. The prevailing winds are from the southwest and northwest.

2.3.2.5 Severe Weather

Tornadoes have been rare in Rhode Island. IF they do occur, they are generally not severe and, in most cases, cause only minor damage to trees or light buildings. The reactor building has survived past hurricanes with only damage to the reactor building roof in 1991. Snow storms and severe cold weather have not presented problems in the past. The emergency generator provides adequate backup for the emergency lighting and the evacuation system, if required.

2.4 Hydrology

2.4.1 Surface Water

There are no major streams or rivers in the area and local flooding is not possible. Natural runoff flows away from the site towards Narragansett Bay as can be seen from the general topography (Figure 2-1).

2.4.2 Storm Water Drainage System

Surface flooding is not a factor since the Bay Campus has a storm drainage system that intercepts local runoff and discharges it away from the site. Storm water piping is below the lowest RINSC floor level.

2.4.3 Sanitary Sewer System

2.4.4 Ground Water

2.4.5 Flooding

The Flood Insurance Rate Map is depicted in Figure 2-4. It shows that the RINSC is situated at an elevation of about 70 feet above mean sea level and that the nearest flood elevation (V10) is at 14 feet. The highest historic storm tide for the Narragansett Bay Shore has been 25 feet during the Hurricane of 1938. Due to the lack of nearby streams and the adjacent down slope to the shoreline, no credible source of flooding exists.

2.4.6 Accidental Release of Liquid Effluents

The probability of an accidental release of radioactive liquid effluents from the RINSC in surface waters is extremely low. Two RINSC systems may contain radioactive liquid: (1) the reactor coolant and (2) the reactor water clean up system. All of the components for these systems; reactor tank, pumps, heat exchangers, filters, resin tanks, valves, and piping, are located within the RINSC reactor and equipment rooms. Any contaminated water leakage from this equipment will be wiped up and disposed of as discussed in Chapter 11.

2.5 Geology, Seismology, And Geo-technical Engineering

2.5.1 Regional and Site Geology

The geology of the state was taken into account in determining the potential for an amplification of ground motions during a seismic event. Most of the state is characterized by glacial till plains overlaying rock of the Esmond Igneous Suite. Till is generally composed of unsorted rocks of varying sizes and is typically a stable, consolidated geological formation. Figure 2-6 shows the basic geology of Rhode Island.

2.5.2 Seismology

The Narragansett area is located in Seismic Zone 2 of the Rhode Island Building Code. In general, seismic activity is low (See Figure 2-8). Rhode Island is located on the North

American Tectonic plate in the northeastern United States. Historically, there has been little seismic activity centered in the state. Only a few earthquakes of modified Mercalli (MM) intensity V or greater have been centered within Rhode Island during the past 200 to 300 years. Figure 2-7 presents the locations of some of the stronger earthquakes in Rhode Island. According to a recent Seismic Vulnerability Study of Rhode Island by George Tsiatas, a modified Mercalli intensity level VI was about the strongest. The report summarizes the possibility of MMI VII earthquake. The state classifies buildings according to class. The reactor building can be classified as a Class 81 (reinforced concrete, low-rise building).

2.5.2.1 Summary and Conclusions

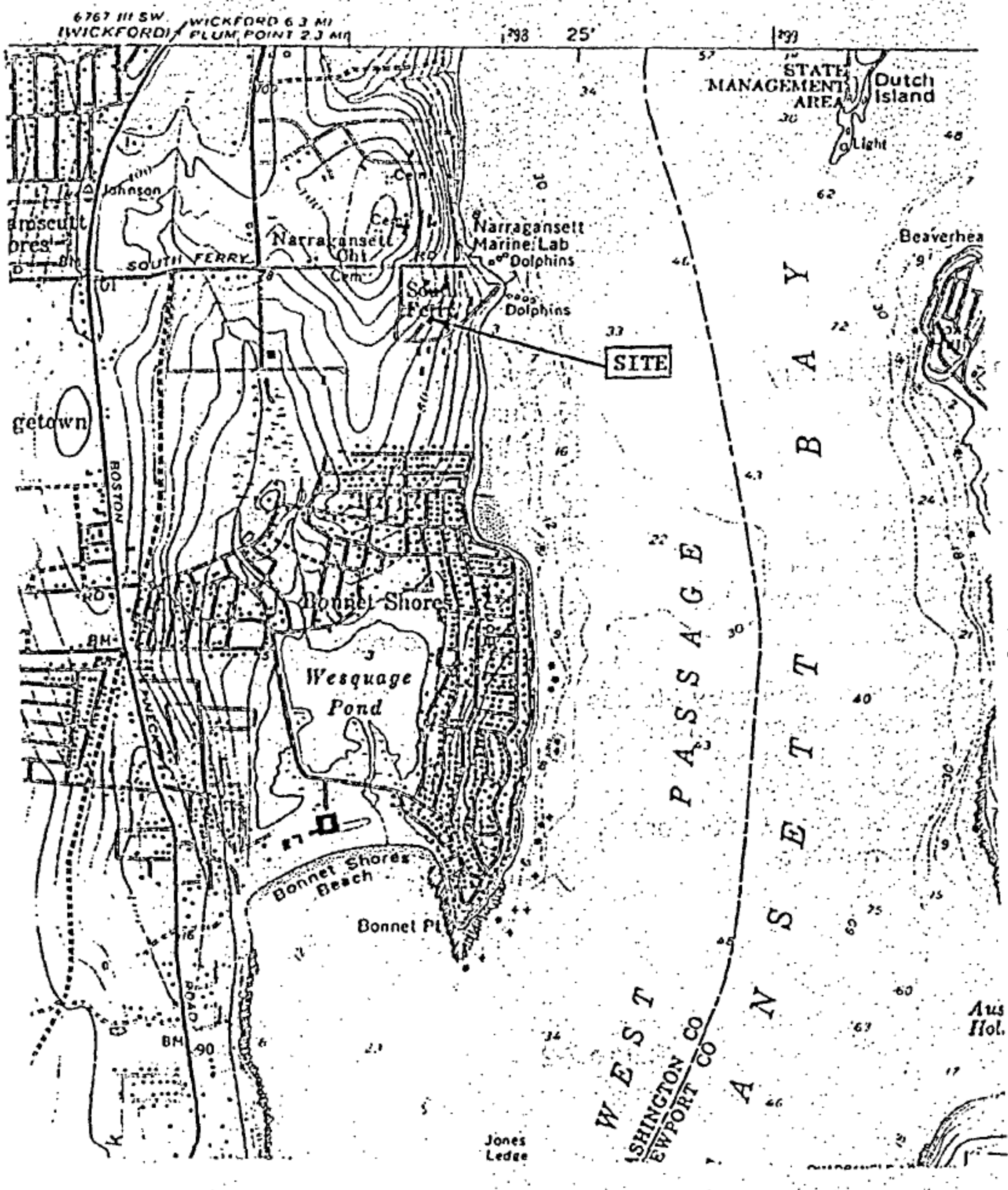
Although some seismic activity occurs in Washington County, it is relatively mild and the probability of damage to the RINSC remains light. The distance from the closest known fault to the RINSC far exceeds the siting requirements of the ANSI/ANS 15.7, Section 3.2(1) that states "No proposed facility shall be located closer than 400 meters from the surface location of a known capable fault."

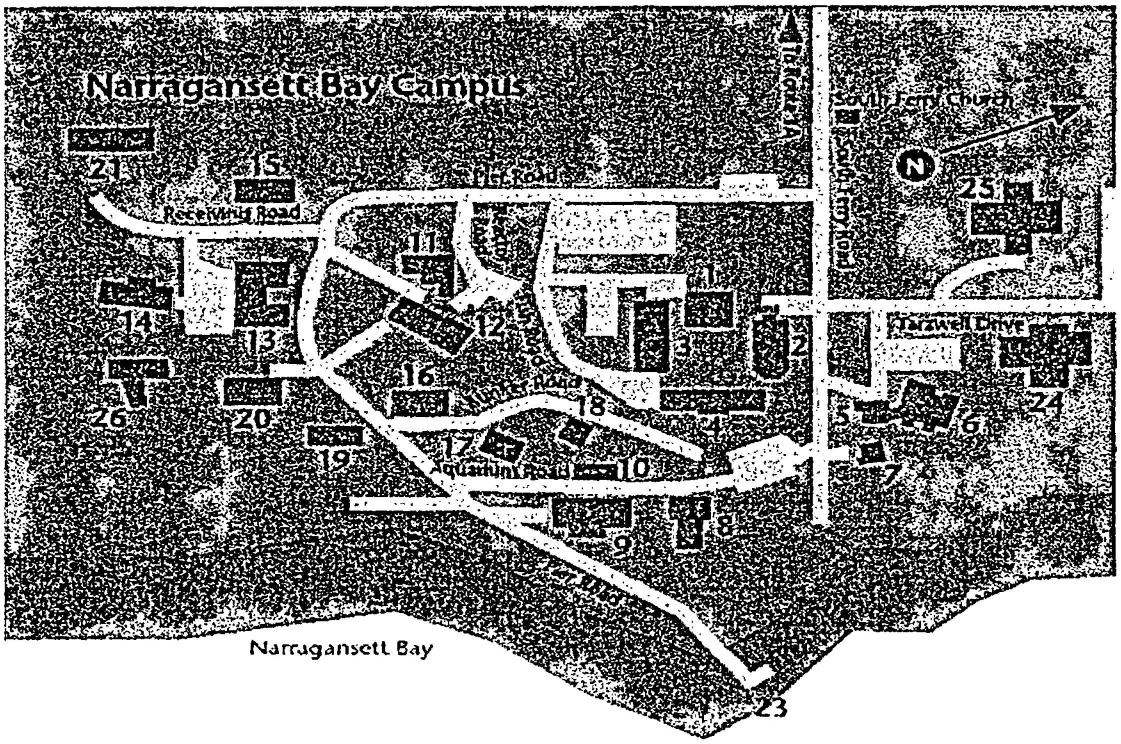
1 Figure2-1

SITE LOCATION PLAN
FOR THE
RHODE ISLAND NUCLEAR SCIENCE CENTER

REFERENCE: USGS NARRAGANSETT PIER
QUADRANGLE 7.5 MINUTE SERIES (TOPOGRAPHIC)

[redacted]

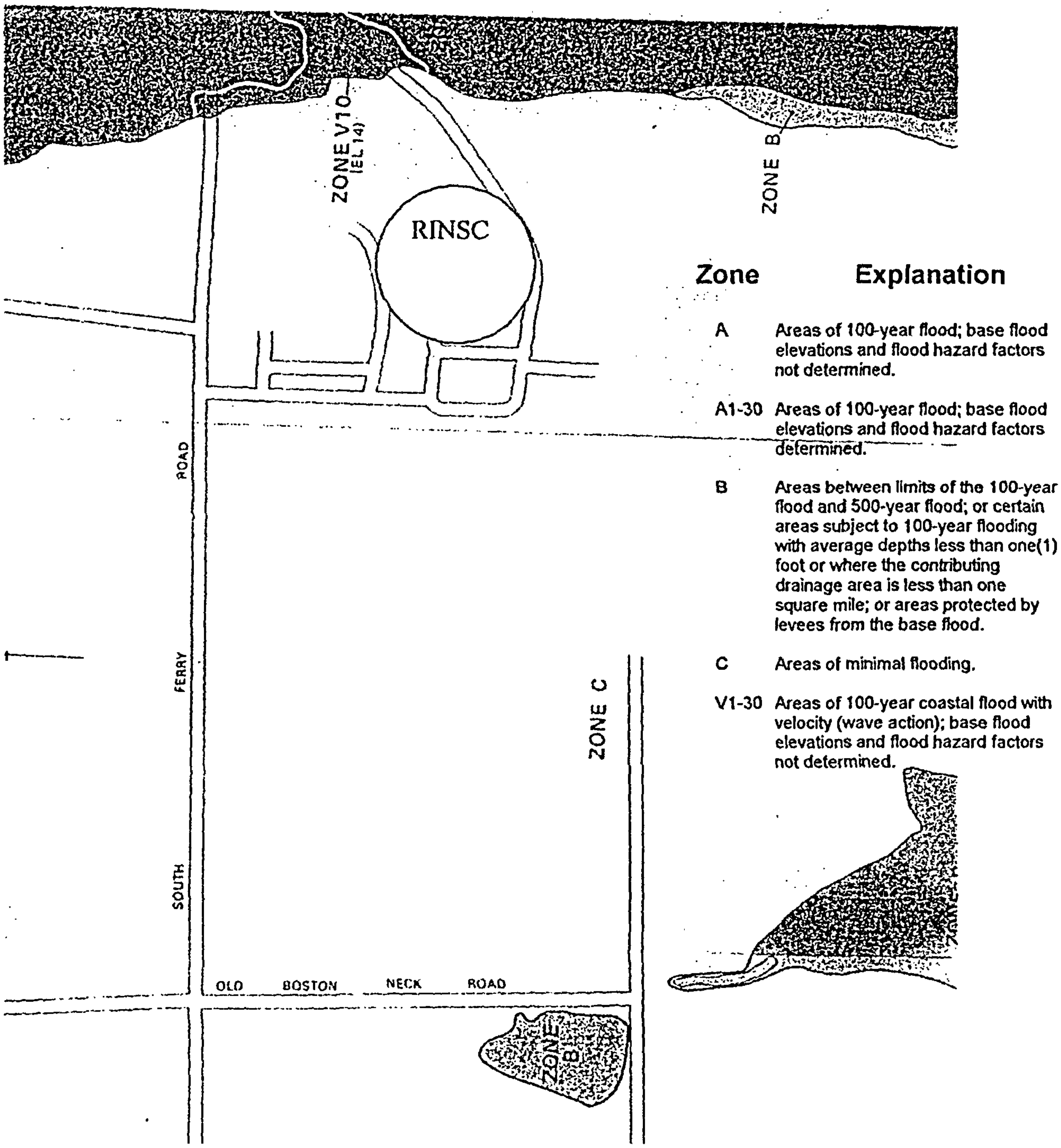




1. Pell Marine Science Library
2. Watkins Laboratory
3. Horn Laboratory
4. Fish Laboratory
5. Marine Resources Building
6. Coastal Institute Building
7. Mosby Center/Food Services
8. Marine Ecosystems Research Lab
9. Aquarium Building
10. Aquarium Annex
11. Center for Atmospheric Chemistry Studies
12. Rhode Island Nuclear Science Center
13. Middleton Building/South Laboratory
14. Sheets Building/Ocean Engineering
15. Marine Geological Samples Laboratory
16. Technical Services Building
17. Marine Building
18. Bunker C
19. Perkins Small Boat Facility
20. Maintenance Building/Marine Office
21. Furado Building/Central Receiving
23. Research Vessel Pier
24. EPA/Environmental Research Lab
25. NOAA/NMFS/NorthEast Fisheries
26. Ocean Technology Center

Figure 2-2

Figure 2-3



Flood Insurance Rate Map Effective: December 7, 1971

Figure 2-4

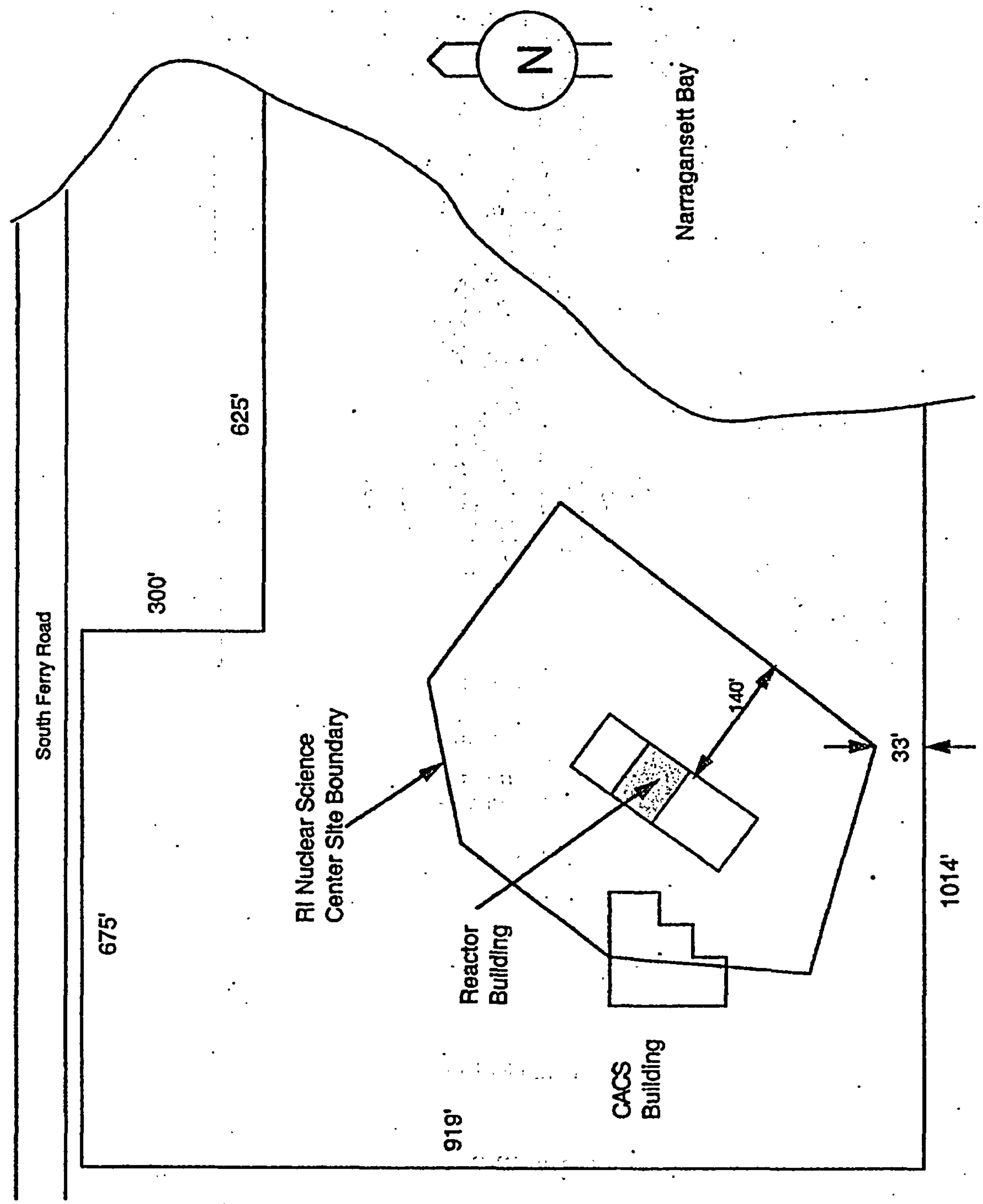
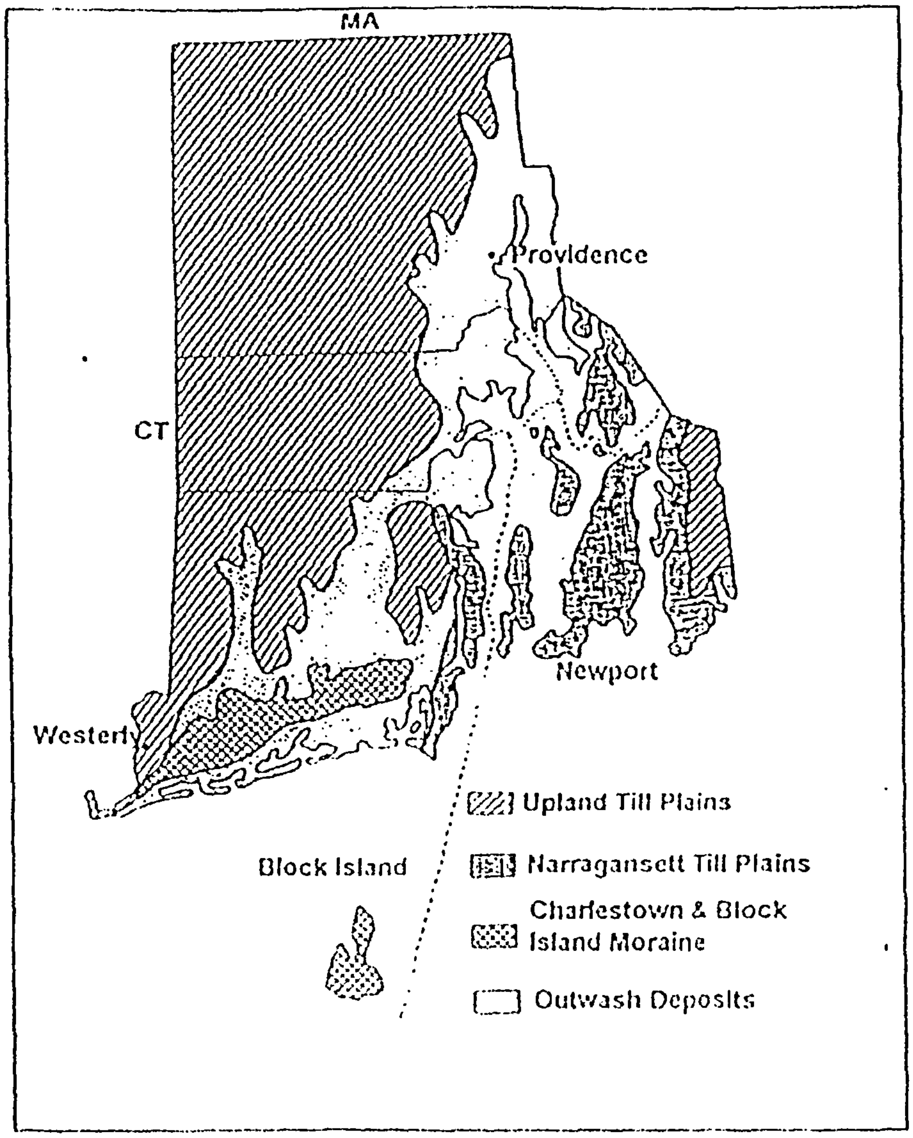
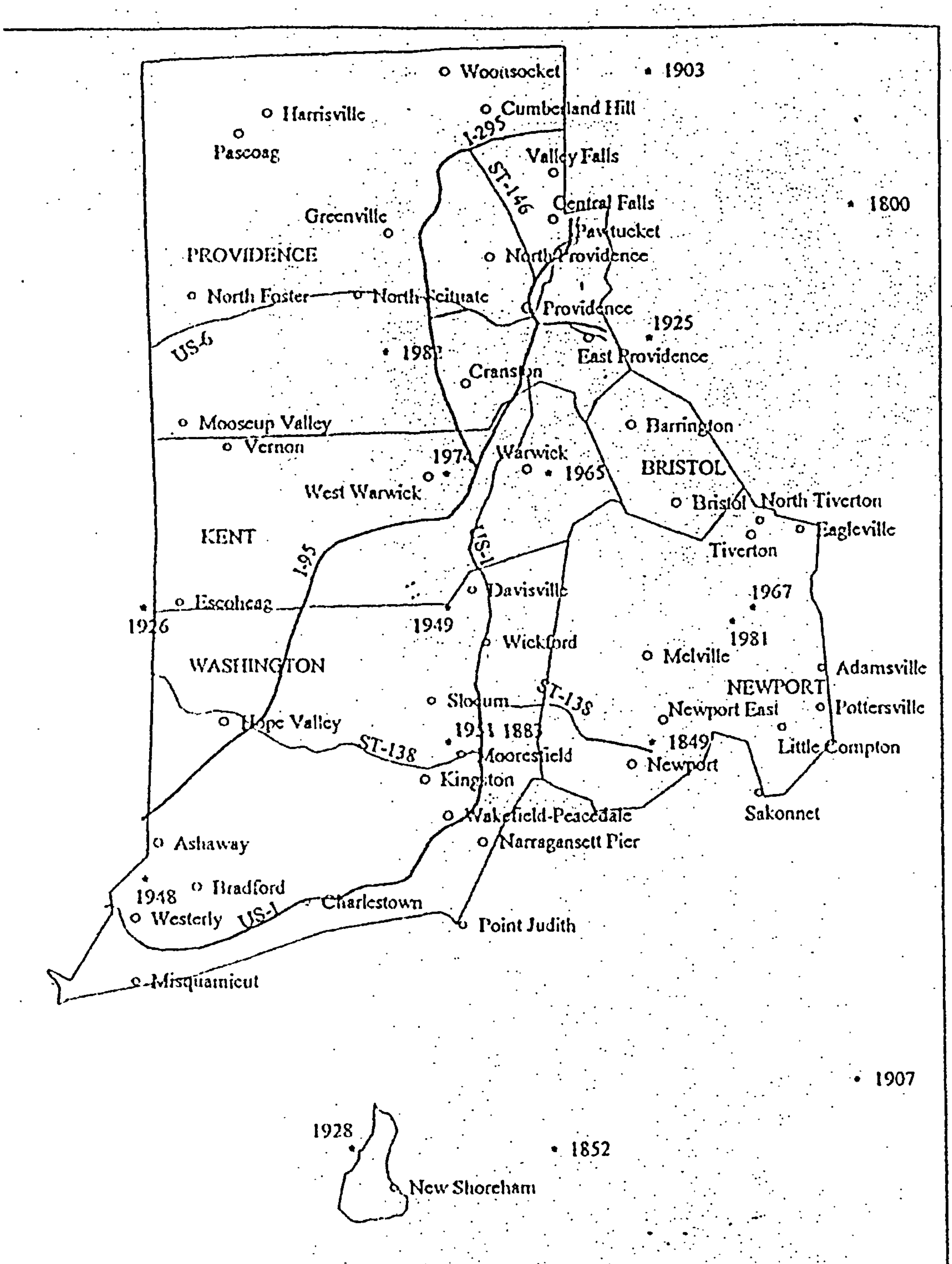

N
Narragansett Bay
South Ferry Road
625'
300'
675'
919'
1014'
33'
140'
RI Nuclear Science Center Site Boundary
Reactor Building
CACS Building

Figure 2-5



Rhode Island Geology

Figure 2-6



Rhode Island Earthquakes

Figure 2-7

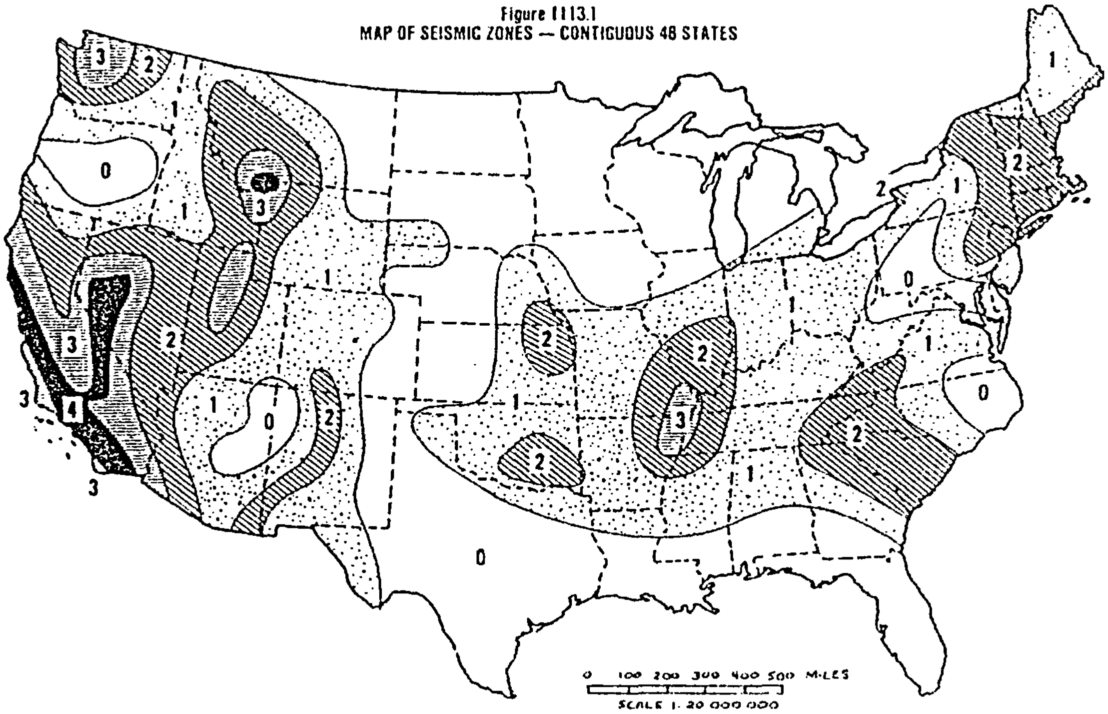


Figure 1113.1
MAP OF SEISMIC ZONES — CONTIGUOUS 48 STATES
0 100 200 300 400 500 MILES
SCALE 1: 20 000 000

Figure 2-8

Monthly Averages for Newport Rhode Island												
	January	February	March	April	May	June	July	August	September	October	November	December
High Temperature (F / C)	37 / 3	38 / 4	46 / 8	57 / 14	67 / 20	77 / 25	82 / 28	81 / 27	74 / 24	64 / 18	53 / 12	41 / 5
Low Temperature (F / C)	19 / -7	21 / -6	29 / -2	38 / 3	47 / 14	57 / 17	63 / 17	62 / 17	54 / 12	43 / 6	35 / 2	24 / -4
Precipitation (in / mm)	4 / 99	4 / 92	4 / 103	4 / 104	4 / 96	3 / 85	3 / 81	4 / 92	3 / 88	4 / 94	4 / 112	4 / 111
Snow (in / cm)	11 / 28	8 / 21	5 / 13	trace/trace	0 / 0	0 / 0	0 / 0	0 / 0	0 / 0	0 / 0	trace/trace	7 / 17
Wind Speed (mph / kmh)	12 / 19	12 / 20	13 / 21	13 / 21	11 / 18	11 / 17	10 / 16	10 / 16	10 / 16	10 / 17	11 / 18	12 / 19
Wind Direction	WNW	WNW	WNW	WNW	SSE	South	South	South	South	WNW	WNW	WNW
Cloud Cover (out of 8)	4.8	4.9	5.1	5.1	5.3	5.1	5.1	4.9	4.7	4.4	5.1	5.0

Figure 2-9

CHAPTER THREE

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Chapter 3

3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

This chapter identifies and describes the principal architectural and engineering design criteria for the structures, systems, and components that are required to ensure reactor facility safety and protection of the public. The material presented emphasizes the safety and protective functions and related design features that help provide defense-in-depth against the uncontrolled release of radioactive material to the environment. The bases for some of the systems discussed in this chapter are developed in other chapters and are appropriately cross-referenced, where required.

3.1 Design Criteria

The RINSC is designed and licensed for operation at a maximum steady-state power level of 2 MW. Thus, its fission-product inventory is substantially less than that of conventional nuclear power plants. In addition, a conservative upper limit of energy released for an entire year of operation would be about 700 MW-Days. These comparisons illustrate why the RINSC should be placed in a much lower risk category than conventional nuclear power plants.

The RINSC reactor does not have structures, components, or systems that are important to safety in the same context as nuclear power plants. For the RINSC, a loss of coolant event, failure of the protection system, or any other credible accident does not have the potential for causing off-site exposure comparable to those listed in the guideline for accident exposures of ANSI 15.7.

Thus, the RINSC does not have structures, components, and systems requiring a Category I classification. However, certain structures, components, and systems have been designed to withstand natural phenomena. These design considerations are discussed in the following subsections.

3.2 Meteorological Damage

Tornadoes are rare in Rhode Island. Based on the small probability of occurrence, postulated low intensity, the intermittent type of reactor operation and low fission-product inventory, no criteria for tornadoes have been established for the RINSC structure. The RINSC reactor core is protected from damage by high winds or tornadoes by virtue of its location in the thick reinforced concrete structure surrounding the reactor tank. The superstructure of the RINSC has been designed for area wind loads including those associated with the infrequent hurricanes reaching the Rhode Island coast. The reactor building has survived past hurricanes with only light damage to the facility roof in 1991. Facility design also accounts for snowstorms and severe cold weather, which has not presented significant problems in the past.

3.3 Water Damage

As discussed in Chapter 2, flooding is not expected at the RINSC site. However, even if flooding occurred, reactor safety would not be an issue since the core is located in a water pool. In the event of a severe storm or flood, the reactor will be shut down and locked if there appears to be even a remote chance of danger in operating the reactor at the time.

3.4 Seismic Damage

The Rhode Island area is classified as being in Seismic Zone 2, as defined in the Uniform Building Code (see Figure 2-8). The RINSC building, reactor foundation, shielding structure, reactor tank, and core support structure have been designed and constructed in accordance with this code. Seismic activity in the region has registered as high as Richter 5 in historical time, which indicates an upper limit on the most likely seismic event. Due to design, there is ample conservatism in the design for the maximum expected event and it is most likely that the reactor can be returned to operation without structural repairs following an earthquake likely to occur during the plant lifetime. The RINSC structures may suffer some damage from a seismic event of the highest possible yield, but, as previously noted, even in the event of the incredible seismic scenario, the resultant radiological doses would be within the ranges evaluated in Chapter 13 and the consequences found acceptable from the standpoint of public safety. In the event of a beyond design basis earthquake, the frequency of vibration of the pool unit may well be different from that of the building, causing dislocations between the reactor and immediate surroundings. Any resultant break in the primary coolant pipes would allow the pool to drain no lower than [redacted], due to the pipe location in the pool concrete and anti-siphon provision.

The facility has a seismic detector (switch) mounted on the biological shield which, when tripped, initiates an automatic reactor scram. Maintenance and testing of the detector is preformed in accordance with the RINSC Operating Procedures and Technical Specifications. The switch is set to scram at a Modified Mercalli's IV, ANSI/ANS-15.7, (rev. 1977) section 3.2 (2) requires a reactor scram for intensities V or greater. Therefore, this requirement is met. The reactor would not be started up again until an examination of the structure could be made to insure that no damage had occurred to the reactor.

3.5 Systems and Components

3.5.1 Instrumentation and Control

The instrumentation and control system for the RINSC is a traditional analog system. The RINSC reactor can be operated in two control modes: manual and automatic. The operations are controlled from the reactor console mode control and the blade control panels. The manual and automatic control modes are used for reactor operation from source level to 100% power. These two modes are used for manual reactor startup, change in power level, and steady-state operation.

The RINSC has five independent reactivity control blades: four safety control blades and one regulating blade. Each of the blades has its own drive mechanism and control circuit and they are operated individually. The control and regulating blades and drives are similar. The regulating blade is used to control power either manually or by automatic control.

The control-blade-drive assemblies are mounted on the reactor bridge structure. The drives are standard drive mechanisms. The mechanism for the control blade is shown in Figure 3-1. The mechanism consists of an AC motor and reduction gear, a rack and pinion, an electromagnet and armature, a dashpot assembly, and a control-blade extension shaft. Blade-position data are obtained from potentiometers. Limit switches are provided to indicate the positions limits of the blades. The normal drive speed for the control blade is 3.6 inches per minute. The normal drive speed for the regulating blade is 78 inches per minute.

During a scram, the control blade, blade extension, and magnet armature are detached from the electromagnet and drop by gravity. The dashpot assembly slows the rate of insertion near the bottom of the stroke to limit deceleration forces. Upon receipt of a scram signal, all the control blades are released from their drives and dropped into the core. Insertion of at least three of the control blades ensures reactor shutdown. Total worth of the blades is more than adequate to maintain the core at a sub-critical level, with the most reactive blade stuck out of the core.

No conceivable malfunction of the reactivity control systems could result in a reactivity accident worse than the conditions encountered during the startup accident. As shown in Chapter 13, neither continuous blade withdrawal nor loss of coolant will cause undue heating of the fuel. Identified accidents will not result in significant movement of adjacent fuel elements or otherwise disturb the core so as to add reactivity to the system.

Since the primary coolant system operates at atmospheric pressure, control blade ejection is not a credible event. The control blades and the regulating blade cannot drop out of the core because the blades in the full down position are approximately one inch above the safety plate located near the bottom of the tank; travel out of the core in the downward position is therefore eliminated.

There are two scram loops, using different input signals, to provide redundancy in scram capability. The protection and reactivity control systems satisfy all existing design standards. Periodic checks (i.e., startup, shutdown, and maintenance procedures) of all reactor protective system channels and reactivity control systems demonstrate that they perform their intended function.

Protection System Functions:

The RINSC Reactor Protection System has been designed to initiate automatic actions to assure that fuel design limits are not exceeded by anticipated operational occurrences or accident conditions. The Reactor Protective System automatically scrams the control blades when trip settings are exceeded (Chapter 7). There are no other automatic actions required by the protective systems to keep fuel temperature limits from being exceeded. The Reactor Protective System satisfies the intent of IEEE-323-1974 in the areas of redundancy, diversity, power-loss fail-safe protection, isolation and surveillance.

The RINSC Reactor Protection System is designed to be fail-safe: any sub-channel loss that causes the channel to lose its ability to perform its intended function results in initiation of shutdown action. Protective action is manifested through several independent scram inputs arranged in series such that action by any one interrupts current to the scram magnets resulting in shutdown of the reactor. Redundancy of channels is provided and in addition, a loss of any channel due to open circuit or loss-of-power will result in a scram. Scram action is, therefore, on a one-out-of-one basis. All instrumentation is provided with testing capability.

The Reactor Protective System and the magnet power supply are, for the most part, physically and electrically isolated from the remainder of the control system. The cables between the control room and reactor bridge are supported by two separate cable trays, isolating instrumentation cabling from power cabling.

3.5.2 Environmental and Missile Design Bases

The construction of the facility precludes catastrophic rupturing of the reactor tank. There is no source in the reactor room for generating large, sustained, positive pressure differentials that would breach the reactor confinement integrity.

The amount of explosive materials allowed in the reactor room has been minimized to preclude damage to the reactor should they detonate. Each experiment containing explosives will be analyzed to show that detonation will not produce pressure or fragments that will damage the reactor. The reactor core is protected from external missiles by being surrounded by a large block of reinforced concrete. The piping systems are anchored and are imbedded in the concrete bioshield walls so that they could not conceivably affect the reactor if damaged. The probability of an event or condition resulting from dynamic effects of missiles, aircraft, etc., causing a reactor incident, is very small. In addition the confinement building is surrounded with vehicle barriers to prevent an explosive laden vehicle from getting within 50 feet.

3.5.3 Reactor Design

Safety limits are established for the reactor for period, excess reactivity, and fuel surface temperature.

Accident analyses presented in Chapter 13 show that under credible accident conditions, the safety limits will not be exceeded. Consequently, there would be no fission product release that would exceed allowable radiation levels.

Because of the fuel material and core design, there is a significant prompt negative temperature reactivity coefficient. Routine steady-state power operation is performed with the shim and regulating blades partially withdrawn. As shown in Chapters 4 and 13, the most rapid possible reactivity insertion rates are adequately compensated for by period alarm and trip provisions.

3.5.4 Electric Power Systems

The primary power distribution system that supplies commercial electric power to the RINSC is maintained by electrical utility maintenance crews.

In case of a power failure, the RINSC is provided with a 15-kW emergency backup power system (see Chapter 8).

Inspection and Testing of Electric Power Systems:
Routine surveillance and inspection of the electric power system including the emergency backup power system is conducted weekly and after any maintenance that could affect system operability.

3.5.5 Fluid Systems

A coolant system is utilized to cool reactor pool water during normal operation of the reactor. The RINSC requires no auxiliary cooling system for cooling of reactor pool water upon shutdown. Natural convection cooling is adequate to dissipate core afterheat.

The reactor pool and cooling systems operate at low pressure and temperature. The pool is open to the atmosphere, and there are no means for pressurizing the system. The reactor pool is constructed of aluminum and the primary coolant system components are aluminum or stainless steel. The system components outside the reactor pool have a low probability of serious leakage or of gross failure. Further, the design of the system is such that even though a line or component ruptures, only a small amount of water would be removed from the tank (~3 ft) (Chapter 5). Rupture of the tank is virtually impossible, since it is supported on the bottom and sides by reinforced concrete. All components containing primary coolant (i.e., reactor pool, primary coolant system, and the purification system) are constructed of aluminum and stainless steel, using standard codes for quality control. There is no requirement for leak detection in the primary coolant or purification loop since no conceivable leak condition can result in the pool water level to lower more than 16 " below the suspension frame base plate elevation with out a scram. There is also a requirement to test the secondary water for sodium –24 which would indicate a leak from the primary water into the secondary water.

The reactor pool is an open system and the maximum pressure in the primary system is that due to the static head. The primary, secondary, and purification systems are pressurized by their circulating pumps. Piping and valves in the primary and purification systems are stainless steel or aluminum and of such size to provide adequate operating margins. The secondary system components are PVC & carbon steel. Chapter 5 describes the cooling system in detail.

The RINSC water purification system design includes an Auxiliary Make Up Water System for makeup of primary coolant water.

Cooling equipment used in normal operation of the reactor is located either in the reactor room, equipment room, or outside the building with adequate space provided to permit inspection and testing of all components. Operation of the bulk coolant and cooling tower system is checked on a daily basis prior to reactor operation. During this checkout, the performance of each system is monitored with emphasis on pump outlet pressures, pressure differentials and system flow rates.

3.5.6 Reactor Confinement

The structure surrounding the reactor constitutes a confinement building rather than providing absolute containment. Because of the low fission-product inventory, leakage from the structure can be tolerated. Accident analyses are presented in Chapter 13.

Under the conditions of a loss of coolant, it is conceivable that the temperature at the reactor room could increase slightly due to heating of the air flowing through the core. However, since the building is not leak tight, it will not pressurize from the heating of the air.

Further, there is no requirement from a radiological-exposure viewpoint for a containment structure; hence, only confinement capability is provided. In addition, there is no source of energy (from an accident) that would provide a significant driving force (ΔP) if no corrective action was taken.

The confinement structure (the reactor building) is a reinforced poured concrete structure with a conventional built-up roof. The entire structure is exposed to only normal external environmental conditions and internal environmental conditions are maintained at regulated conditions.

Provisions for Confinement Testing and Inspection:
The reactor room confinement capability is checked on a daily basis prior to reactor operation and routinely during reactor operations. This check involves monitoring the pressure differentials between the reactor room and the surrounding areas. The reactor room exhaust system is checked weekly to confirm proper operation.

Control Room:
In the event of an accident where operational instructions require shutdown of the reactor, continuous or even partial occupancy of the control room is not a requirement since the reactor has been shut down and experiments in progress terminated. Exposure levels from radiation sources resulting from an accident would be significantly reduced in magnitude (due to location of the control room with respect to the reactor room). Consequently, control room radiation levels may not be higher than the allowable tolerance levels. Nevertheless, the RINSC Emergency Plan describes actions for mitigating accident situations that require control room evacuation.

3.5.7 Radioactivity Control

There is no readily available path for radioactive liquid waste to be discharged directly to the environment. Liquids in the reactor room that could subsequently be released into the environment may result from spills, washdown of the floor, etc. These liquids are collected in storage tanks within the RINSC, analyzed for radioactivity and disposed of accordingly according to procedures.

Fuel Storage and Handling and Radioactivity Control:

[redacted]

3.5.8 Missile Protection

As discussed in 3.52, Missile protection is provided for the RINSC reactor by virtue of the reinforced poured concrete building design and the location of the core that is surrounded by a thick reinforced concrete biological shield (see Chapter 1 for building design).

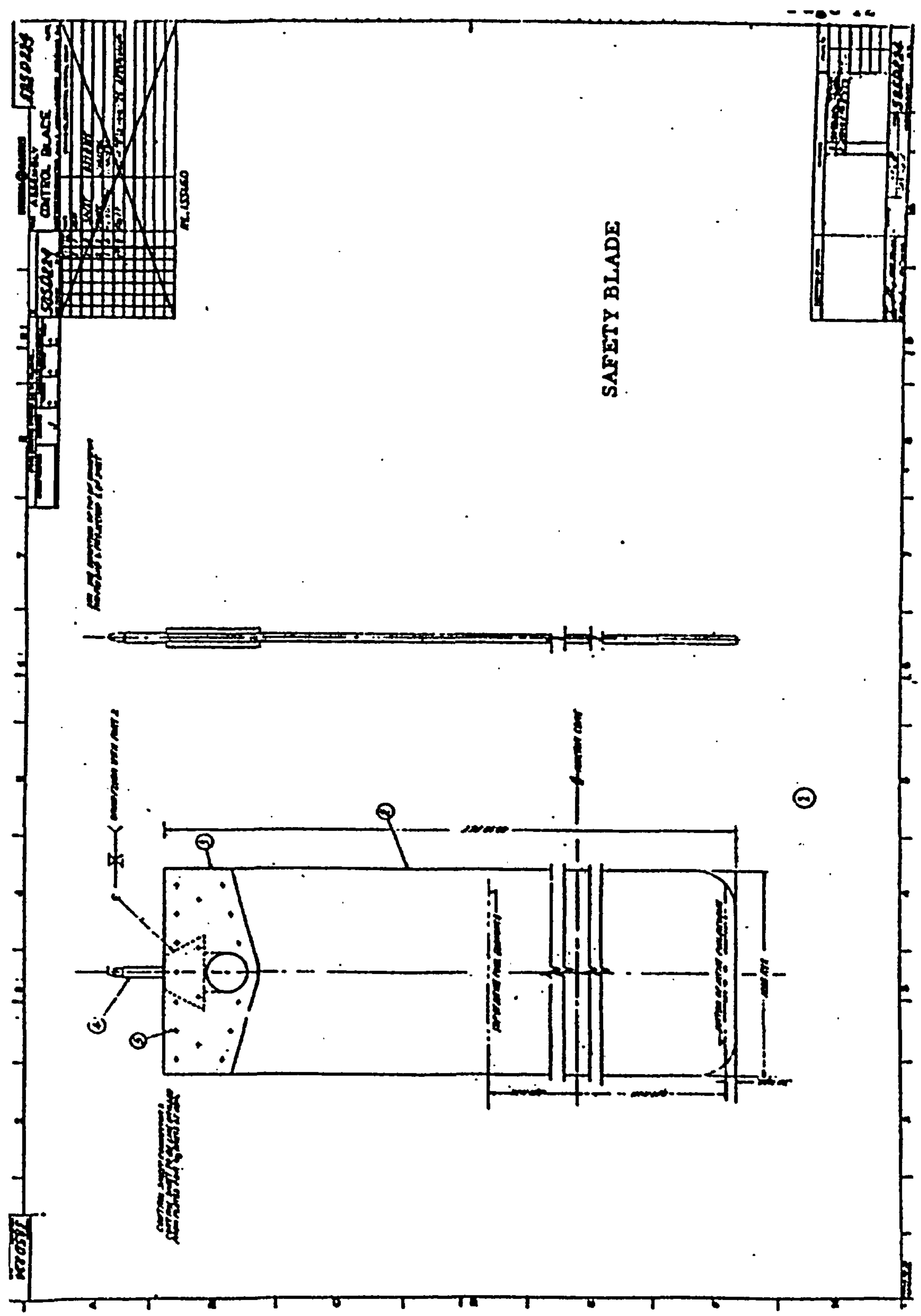

Figure 3-1

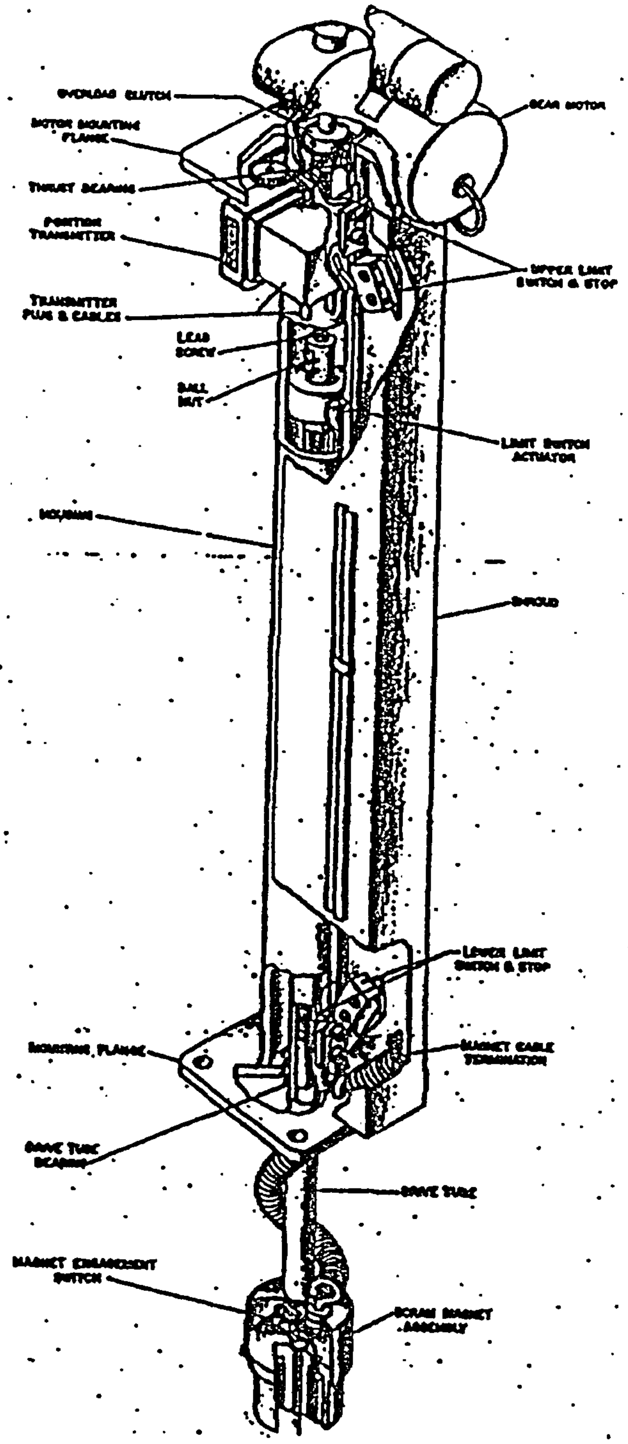

Figure 3-2

CHAPTER FOUR

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Chapter 4

4.0 REACTOR DESCRIPTION

This chapter discusses and describes the principal features, operating characteristics, and parameters of the reactor. The analysis in this chapter supports the conclusion that the reactor is conservatively designed for safe operation and shutdown under all credible operating conditions. The information in this chapter provides the design bases for many systems, subsystems, and functions discussed elsewhere in the SAR and for many of the Technical Specifications.

4.1 Summary Description

The RINSC contains a 2 $MW_{(t)}$, open-pool, water-cooled (natural and forced convection mode) reactor. Since the conversion from HEU in 1992, the core consists of LEU fuel reflected with a combination of graphite and beryllium. The fuel is Uranium Silicide-Aluminum (U_2Si_2-Al) enriched to less than 20% Uranium-235.

The reactor core assembly is located near the bottom of a 32-foot deep, aluminum-lined pool. It can be moved into any of three sections by means of a mechanical rail system located on top of the pool wall. The three sections are the high-power section (for operation at power levels above 0.1 $MW_{(t)}$ with forced circulation), the dual-storage section, and the low-power section. At power levels below 0.1 $MW_{(t)}$, the reactor may be operated, with core cooling by natural convection, in any of the sections. As spent fuel elements are generated, they are placed in the fuel storage racks located in the dual storage section and in the low-power section. The spent elements may be used as a source of gamma radiation.

The reactor accommodates an experiments position (flux trap at peak flux location) at the center of the core within a beryllium reflector element. This flux trap has a 38mm (1.5 inches) diameter and is supplied with a beryllium plug when not being used. Incore devices can be placed in irradiation baskets along the edge of the core opposite the thermal column. These devices are designed for large capacity of samples and long duration irradiations. The reactor also provides irradiation for experiments by utilizing two pneumatic tubes for small targets; 6 horizontal beam ports for long-term irradiations and neutron beam extraction experiments such as neutron scattering and neutron spectroscopy; a thermal column containing graphite for neutron radiography, etc.; and a dry irradiation room located adjacent to the low power section of the pool for gamma irradiations.

The reactor core is based on fuel elements in a rectangular array, surrounded on four sides by reflector elements. Four safety control blades and a servo-actuated regulating blade control the reactivity. The control blades move vertically within a pair of shrouds extending the length of the core. Core elements are contained in a grid box that is enclosed on four sides to confine the flow of cooling water between elements. The grid box assembly, including the drive mechanisms, is supported by the suspension frame. The elements that make up the core sit on a 7 x 9 grid plate with the four corner positions occupied by the suspension frame corner posts. These corner posts connect the grid plate to the reactor bridge that spans the open pool. The neutron detectors are suspended within the water filled corner posts. The grid plate is suspended about [redacted] below the pool water surface. The core suspension system includes the reactor bridge, the suspension frame, the locating plate, and the blade drive mechanisms.

The core assembly is cooled by water under either natural or forced convection mode. Single primary pump operation circulates approximately 1,950 gpm of coolant through the grid box assembly. The heat from the pool and primary coolant systems is transferred to the secondary coolant system by means of heat exchangers. The heat is then dissipated to the atmosphere through cooling towers.

4.2 Reactor Core

The core assembly consists of the following components: the reactor fuel, the control blades, the neutron moderator and reflector, the neutron source, and the core support structure. The control blade system is described in greater detail in Chapter 7, Instrumentation and Control Systems.

The core consists of a 7 by 9 array of 3-inch-square elements, with the center array filled with fuel elements, and the four corners occupied by the suspension posts. Graphite and beryllium reflector elements surround the fuel section while five radiation baskets normally making up row 9 (Figure 4-1). A beryllium reflector with flux trap occupies core position D-5 in the center of the core.

The locating plate, made of heavy steel, spans the upper end of the suspension frame to provide support and location for the control blade drive mechanisms (Section 4,2,2 below). The control blade drive guide tubes are flanged to the bottom of this locating plate.

4.2.1 Reactor Fuel

Each fuel element consists of two aluminum side plates and 22 equally spaced flat LEU fuel plates. The fuel, with an active length of 24 inches, is sandwiched between aluminum cladding on each side. When assembled in the fuel element, the plates are separated by a gap for water passage. Two identical end boxes position the fuel elements in the grid [redacted]. The elements are approximately 40 inches long, including the end boxes. The elements may be inverted and rotated 180 degrees to achieve more efficient utilization of fuel.

The core elements are supported and enclosed on four sides by the grid box (Figure 4-1). The grid box is approximately 28 inches long, 24 inches wide, and 36 inches high. The bottom is an aluminum grid plate with a 7-by-9 array of square holes, spaced to conform to the basic 3-inch element module. Four corner posts, attached to the lower end of the suspension frame, support the grid box and can contain neutron detectors. All of the grid box components are made of aluminum.

Currently two core configurations are authorized for used in the RINSC reactor. These being the 14 element LEU startup and subsequent cores analyzed during the HEU to LEU conversion process and the larger 17 element core designed to produce a higher neutron leakage flux into the thermal column which is discussed in section 4.7.

4.2.2 Control Blades

Reactor control for startup and shutdown is accomplished by four blade-type control blades working vertically within a pair of shrouds located parallel to the major axis of the core. Blades and shrouds are depicted in Figure 4-1 by the two vertical sections that divide the core into three regions. The poison material of the blades is boron carbide and aluminum that is sandwiched between aluminum side plates. The poison section is 40.5 inches long, 25 inches of which provide active control of the core; the remaining 15.5 inches connect the poison section to the drive tube. The shrouds act as guides for the control blades throughout its travel. When a control blade is fully withdrawn from the core, at least 3 inches of the control blade remains engaged in the shroud. The shroud consists of two thin aluminum plates 38 inches high, separated by aluminum spacers to provide an eighth-inch water annulus around the blade. The shroud is fastened to the sides of the grid box by screws. Small flow holes at the bottom of the shroud minimize the effect of viscous damping on the blade drop time.

These four control blades are actuated by electromechanical drives that position, hold, and scram each control blade. The drives, mounted on the locating plate above the core, are coupled to the control blades through electromagnets that provide gravity scram when de-energized. The drive mechanisms include a drive motor, worm-gear reducers, slip clutch, ball-bearing screw assembly, limit switches, scram magnet, and assembly housing. Related instruments to the drives give a continuous indication of position within 0.02 of an inch. The drive motor is a reversible electric motor with an integral worm-gear assembly to reduce speed and prevent drift of the control blade. A mechanical slip clutch on the output shaft limits the force on the control blade to approximately 75 foot-pounds. A ball-bearing screw and nut are utilized to raise and lower the control blade.

In order to minimize friction and possible binding, large clearances are provided in guide bearings of the control shafts. All lubricants are sealed to prevent leakage into the reactor pool. The working parts of the drive are enclosed in the drive tube, which is sealed off at the lower end by attachment to the scram magnet assembly, or to the solid coupling in the case of the regulating blade. The vertical travel of the drive does not allow the magnet assembly to enter the pool water. These design factors preclude any significant contamination of the pool water with lubricants from the drive mechanisms. The control drive mechanism can operate through a stroke of 32 inches at a maximum speed of 3.6 inches per minute in either direction. Coasting of the mechanism is limited to less than 0.1 of an inch of blade travel.

Limit switches at the ends of the stroke de-energize the drive motor and operate the indicating lights. In addition, a limit switch within the scram magnet gives remote indications when the magnet engages the control blade. Scram action is initiated by de-energizing the holding magnet, which releases the extension shaft and control blade.

The release completely separates the blade from the drive mechanism. The blade is free to fall after de-energizing the holding magnet, and at this time the blade drops into the core under the force of gravity. In the case of power failure, the blades drop automatically. To recover the control blade after a scram, the drive mechanism is run down and the magnet attaches to the top of the control blade shaft.

The regulating blade is actuated by an automatic servo control system to compensate for small changes in reactivity. The blade has a maximum worth of 0.7% $\Delta k/k$. The blade is a 25-inch long by 2½-inch square stainless steel channel, lock-screwed to the servo regulating drive shaft. The element guide tube is a 3-inch square aluminum shell seated in the reflector region of the grid.

The servo-controlled drive automatically regulates reactor power within closer limits than those attainable by using the control blades. The regulating blade is driven by a stepper motor-controlled mechanism. Remote position indication is provided to the control console. A solid coupling replaces the holding magnet; therefore the regulating blade does not scram. The maximum speed of travel of the regulating blade is 78 inches per minute, with a total stroke of 22 inches.

4.2.3 Neutron Moderators and Reflectors

The graphite reflector element is a block contained in a 3-inch square aluminum can equipped [redacted] to facilitate handling. AGOT reactor-grade graphite is used. By reducing neutron flux leakage from the active core, the reflector elements increase the neutron flux at the core perimeter and improve utilization of the fuel. The graphite in the reflector element extends about 3 inches above and below the active 24-inch length of the adjacent fuel elements. The thin-walled aluminum can is evacuated to collapse the walls onto the graphite and thus provide good heat transfer to the pool water. Flexure of the thin wall will accommodate swelling of the graphite under irradiation.

The design of the element allows for this thermal expansion to a maximum increase in graphite dimensions of 1.1% due to irradiation growth and gas evolution from an integrated flux of 2×10^{21} nvt (expansion based on a more than two-year, full-power operation factor). Numerous tests under conditions similar to those expected in an open pool reactor reveal no significant changes in graphite properties in this application. Because of the longevity of the use of graphite, new replacement elements were installed after existing elements reached their life expectancy.

As a result of the LEU studies and conversion, beryllium reflectors were introduced into the new cores for efficiency and operational purposes. All the new reflectors are handled with the existing facility handling tools. For the 17 fuel element core, three graphite reflector elements facing the thermal column are removed and replaced with three additional fuel elements.

4.2.4 Neutron Startup Source

An antimony-beryllium neutron source is provided for routine startup of the reactor. This main source consists of an antimony bar that is one inch in diameter by six inches long, surrounded by a thick beryllium cylinder and sheathed in a watertight aluminum jacket. Provision is made for remote handling. Operation of the reactor at an average power of 10 kilowatts will maintain the neutron source strength at 10 curies, and this source strength will produce over 0.3 milliwatts in the reactor when it is shut down with the control blades in the core. The source assembly rests inside an irradiation basket on a support bar, whose position ensures that the source is in the active region of the core. The irradiation basket is an aluminum shell, 3 inches square by 42 inches long, located in row 9 of the core assembly. In addition to the main neutron source indicated above, two small-capsulated plutonium-beryllium neutron sources (1.3 x 10^6 n/cm^2/sec each) are provided for initial startup and low-power operation of the reactor. The smaller sources are placed in the core separate for the large antimony-beryllium source. The PuBe source holder [redacted] must be removed at power levels above 10 kilowatts.

4.2.5 Core Support Structure

The frame, suspended from the bridge, is an all-aluminum rectangular column built up of four square corner posts that form a rigid structure to which the core box is attached at the lower end. Cross braces and stiffeners provide structural rigidity and alignment in the upper half of the frame, while coolant flow channels provide this function in the lower portion of the frame. As illustrated (Figure 5-3), stiffeners are provided on three sides of the frame, while the fourth side is open to provide access to the core. This open side allows ready access to any one of the spaces in the grid box.

4.3 Reactor Pool

The reactor core sits on a 7 x 9 grid plate near the bottom of a 32-foot deep pool. This pool is lined with an aluminum liner that aids in maintaining water purity and minimizes water leakage into the concrete. Piping penetrates the pool liner structure to allow for process systems lead-ins and experimental piping to the core grid box area. Major experiment facilities converge toward the core, and afford ample opportunity for the simultaneous performance of a number of different experiments.

There is a bridge assembly over the pool to provide working surfaces for inspection, maintenance, and fuel handling. The design of the bridge and frame are such that it can be mechanically moved along a rail system on top of the pool-level wall. The rigid steel structure is a unique double bridge, with a second floor above the component support level. The second floor is supported independently over the bridge wheels, and allows easy access to all reactor components without danger of vibration-induced shutdown of the reactor.

The entire bridge can be moved by a hand crank and gear drive between the high-power and low-power pool sections. The bridge moves on rails that rest on the parapet walls of the pool. The bridge position is interlocked with the reactor scram circuit to prevent movement during operations. The volume of the reactor pool, including the pool coolant systems, is approximately 40,000 gallons.

4.4 Biological Shield

The reactor is shielded so that radiation levels at all points above and outside the pool are kept below one mrem/hr. In order to achieve the required shielding, the reactor core assembly is maintained below 24 feet of water. The shield to the side of the core consists of at least three feet of water in addition to [redacted] of concrete or equivalent shielding in the low-power section of the pool. The beam ports and the thermal column are shielded by dense concrete plugs and steel door, respectively. The purpose of the biological shield is to provide radiation shielding for personnel working in and around the reactor facility and is adequate to meet the applicable personnel radiation protection requirements of 10 CFR 20. The immense size of the biological shield also provides excellent protection against natural phenomena that could result in damage to the reactor core assembly.

4.5 Nuclear Design

The initial core and a subsequent improved flux core have been analyzed and authorized for use in the RINSC. The LEU conversion core consists of a compact configuration using 22 standard plates per fuel element and a combination of graphite and beryllium reflectors. Figure 4-1 shows the startup configuration of the conversion core that consists of 14 fuel elements. The fuel elements contain a total of [redacted] of U-235 each. A central beryllium piece with a 38mm diameter vertical hole is incorporated as a flux trap. The regulating blade is stainless steel and was moved one grid position from the original HEU core so as to be adjacent to the fuel elements. The LEU fuel used is uranium silicide-aluminum dispersion fuel approved for use by the NRC under NUREG-1313. The core design allowed the use of the existing graphite reflectors along with beryllium reflectors.

Because of the one shift operation, the xenon behavior of this core is cyclical and this core can be operated as long as it remains possible to achieve start-up on a Friday morning. Using computer simulation, this core has been "run down" until a Friday morning startup is no longer possible. The reactivity requirements for Xenon, Samarium, long lived fission products, control, and the cold-hot swing is approximately 3% which allowed for approximately four months of operation before it is not possible to start up. The reactivity balance is shown in Table 4-1.

After this initial operation, ten beryllium and ten graphite reflector elements are reconfigured to provide additional reactivity. Figure 4-6 also presents this second core showing the fuel remaining in each fuel element after the initial startup core operation. This configuration allows for an additional 70 weeks of 7-hours/day, 5-days/week operation. The reactivity balance is shown in Table 4-2.

Following this second phase of operation, the graphite and beryllium reflectors are again reconfigured. This third core is shown in Figure 4-6 that also shows the fuel in each element at the start of this phase. Table 4-2 again presents the reactivity balance that allows for an additional 60 weeks of operation.

TABLE 4-1
Reactivity Balances on the Friday Morning of the Last Week of Operation for Ten Cores from Startup to Equilibrium

	Reflector Changes Only			4 Burned Elements Removed and 4 Fresh elements Added in Corners						
	Startup % Δk/k	Core 2 %Δk/k	Core 3 %Δk/k	Core 4 %Δk/k	Core 5 %Δk/k	Core 6 %Δk/k	Core 7 %Δk/k	Core 8 %Δk/k	Core 9 %Δk/k	Core 10 %Δk/k
Fresh Cold Clean	3.00	5.19	6.92	6.92	6.92	6.92	6.92	6.92	6.92	6.92
Reactivity Losses										
Burnup	0.30	1.85	3.17	3.08	3.09	3.12	3.07	3.06	3.06	3.06
Xenon	1.54	1.54	1.54	1.54	1.54	1.54	1.54	1.54	1.54	1.54
Samarium	0.57	0.73	0.73	0.73	0.73	0.73	0.73	0.73	0.73	0.73
Long-Lived F.P.	0.09	0.57	0.97	1.06	1.03	1.03	1.07	1.06	1.07	1.07
Cold-Hot Swing	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30	0.30
Control	0.20	0.20	0.20	0.20	0.20	0.20	0.20	0.20	0.20	0.20
Total Losses	**3.00**	**5.19**	**6.91**	**6.91**	**6.89**	**6.92**	**6.91**	**6.89**	**6.90**	**6.90**

TABLE 4-2
Calculated Data and BOC Excess Reactivity for First Ten Cores

	Core Lifetime Weeks	Accum. Weeks	Operation Years	BOC Excess %Δk/k
Startup	14	14	0.3	3.0
Core 2	70	84	1.6	4.1
Core 3	60	144	2.8	3.7
Core 4	33	177	3.4	3.0
Core 5	51	228	4.4	3.6
Core 6	66	294	5.7	4.0
Core 7	54	348	6.7	3.9
Core 8	53	401	7.7	3.9
Core 9	57	458	8.8	4.0
Core 19	57	515	9.9	4.0

Note that the core is now almost completely beryllium reflected. The core has operated for several years and refueling is now required. Refueling consists of removing the four elements with the most burn-up, placing four fresh elements in the core corner positions, and placing the remaining used fuel elements in the remaining positions with those elements containing the least fuel nearest the center of the core. This process provides the flattest flux and greatest neutron leakage. Eventually an equilibrium core will be reached.

Figure 4-6 presents this eventual equilibrium core where the four elements with the most burn-up have been discharged and four fresh elements have been added to the edge of the core. The average discharge burn-up for this equilibrium core is about 21%.

Reactivity coefficients were calculated separately for changes in coolant temperature, coolant density, and fuel temperature using a 3D diffusion model of the LEU core. The three physical effects contributing to these coefficients are: (1) hardening of the neutron spectrum due to increasing the water temperature only, (2) the increase in neutron leakage due to decreasing the water density only, and (3) the increase in absorption due to doppler broadening of the U-238 resonance as the temperature of the fuel meat increases. The reactivity coefficients between 20°C and 40°C are shown in Table 4-3 along with the void coefficient for a uniform 1% change in the coolant density for all fuel elements.

TABLE 4-3

	LEU Startup Core	LEU Transition Core 2	LEU Equilib. Core 10
Kinetics Parameters			
Delayed Neutron Fraction, β-eff, %	0.782	0.776	0.764
Prompt Neutron Generation, time, μs	66.20	66.00	68.30
Reactivity Coefficients : 20-40°C			
Change in Water Temperature Only $\Delta k/k \times 10^{-4}$ /°C	-0.80	-0.86	-0.89
Change in Water Density Only $\Delta k/k \times 10^{-4}$ /°C	-0.82	-0.75	-0.69
Coolant Coeff., $\Delta k/k \times 10^{-4}$ /°C	-1.60	-1.60	-1.60
Doppler Coeff., $\Delta k/k \times 10^{-4}$ /°C Fuel	-0.18	-0.18	-0.18
Temperature Coeff*, $\Delta k/k \times 10^{-4}$ /°C	-1.80	-1.80	-1.80
Δk/k/% Void	-0.0027	-0.0025	-0.0023

*Fuel and coolant temperature changes were assumed to be the same here. The fuel temperature rise will be larger than that of the coolant. Change in Reactivity = (Coolant Coefficient) x $\Delta T_{coolant}$ + (Doppler Coefficient) x ΔT_{fuel}

Because of their harder neutron spectrum, the LEU cores have smaller reactivity coefficients due to changes in water temperature only, but larger reactivity coefficients due to changes in water density only. As burnup increases in the LEU cores, the neutron spectrum becomes softer, the reactivity coefficient due to changes in water temperature increases, and the reactivity coefficient due to changes in water density decreases. The RINSC Technical Specifications reflect these calculated values. Measurements of the coefficients are taken to confirm the calculated values when appropriate.

Power distributions and nuclear power peaking factors were calculated using the diffusion theory model for the LEU cores. From the point of view of thermal-hydraulic safety margins, the most important neutronic parameter is the total 3D power peaking factor (the absolute peak power density in a fuel assembly divided by the average power density in the core). The total power peaking factor is defined here as the product of two components: (1) a radial factor defined as the average power density in each assembly divided by the average power density in the core and (2) an assembly factor defined as the peak power density in each assembly divided by the average power density in that assembly. The assembly factor is a point-wise factor computed at the mesh interval edge and includes both planar and axial power peaking. Power distributions for the LEU cores are shown in Tables 4-4A, 4-4B, and 4-4C.

TABLE 4-4A
RINSC LEU STARTUP CORE POWER PEAKING FACTORS
All Blades Fully Withdrawn

Area	Average Power	Peak Power	Radial	Element	Total
E3	43.0178	100.295	0.86	2.33	2.00
E4	51.8491	115.751	1.03	2.23	2.30
E5	58.2067	128.071	1.16	2.20	2.55
E6	52.0210	115.829	1.04	2.23	2.31
E7	42.6059	100.259	0.85	2.35	2.00
D3	45.4918	86.962	0.91	1.91	1.73
D4	58.7650	131.842	1.17	2.24	2.62
D6	59.0484	132.027	1.18	2.24	2.63
D7	44.4948	76.501	0.89	1.72	1.52
C3	43.0178	100.295	0.86	2.33	2.00
C4	51.8491	115.751	1.03	2.23	2.30
C5	58.2067	128.071	1.16	2.20	2.55
C6	52.0210	115.829	1.04	2.23	2.31
C7	42.6059	100.259	0.85	2.35	2.00
FUEL	50.2287	132.027			

All Blades 50% Withdrawn

Area	Average Power	Peak Power	Radial	Element	Total
E3	40.7418	110.359	0.81	2.71	2.20
E4	50.3451	126.867	1.00	2.52	2.53
E5	57.5876	148.542	1.15	2.58	2.96
E6	50.4967	126.946	1.01	2.51	2.53
E7	40.3213	110.301	0.80	2.74	2.20
D3	48.8511	98.835	0.97	2.02	1.97
D4	63.9026	153.397	1.27	2.40	3.05
D6	64.1589	153.570	1.28	2.39	3.06
D7	47.4674	87.542	0.94	1.84	1.74
C3	40.7418	110.359	0.81	2.71	2.20
C4	50.3451	126.867	1.00	2.52	2.53
C5	57.5876	148.542	1.15	2.58	2.96
C6	50.4967	126.946	1.01	2.51	2.53
C7	40.3213	110.301	0.80	2.74	2.20
FUEL	50.2404	153.570			

TABLE 4-4B
RINSC LEU EQUILIBRIUM CORE 10 POWER PEAKING FACTORS
All Blades Fully Withdrawn

Area	Average Power	Peak Power	Radial	Element	Total
E3	45.0534	106.770	0.90	2.37	2.12
E4	51.2560	116.395	1.02	2.27	2.32
E5	53.9262	108.068	1.07	2.00	2.15
E6	51.3449	111.248	1.02	2.17	2.21
E7	47.7459	111.585	0.95	2.34	2.22
D3	45.3506	87.860	0.90	1.94	1.75
D4	54.7329	118.019	1.09	2.16	2.35
D6	56.6276	118.874	1.13	2.10	2.36
D7	48.4466	83.877	0.96	1.73	1.67
C3	45.0534	106.770	0.90	2.37	2.12
C4	51.2560	116.395	1.02	2.27	2.32
C5	53.9262	108.068	1.07	2.00	2.15
C6	51.3449	111.248	1.02	2.17	2.21
C7	47.7459	111.585	0.95	2.34	2.22
FUEL	50.2722	118.874			

All Blades 50% Withdrawn

Area	Average Power	Peak Power	Radial	Element	Total
E3	42.8196	119.259	0.85	2.79	2.37
E4	49.5608	129.708	0.99	2.62	2.58
E5	53.0974	128.564	1.06	2.42	2.56
E6	49.7684	125.308	0.99	2.52	2.49
E7	45.6472	124.949	0.91	2.74	2.49
D3	48.8135	101.667	0.97	2.08	2.02
D4	59.4384	139.914	1.18	2.35	2.78
D6	61.5902	141.058	1.23	2.29	2.81
D7	52.0959	96.088	1.04	1.84	1.91
C3	42.8196	119.259	0.85	2.79	2.37
C4	49.5808	129.708	0.99	2.62	2.58
C5	53.0974	128.564	1.06	2.42	2.56
C6	49.7684	125.308	0.99	2.52	2.49
C7	45.6472	124.949	0.91	2.74	2.49
FUEL	50.2661	141.058			

TABLE 4-4C
RINSC LEU TRANSITION CORE 2 POWER PEAKING FACTORS
All Blades Fully Withdrawn

Area	Average Power	Peak Power	Radial	Element	Total
E3	42.8832	104.661	0.85	2.44	2.08
E4	52.4493	122.084	1.04	2.33	2.43
E5	59.0606	127.120	1.18	2.15	2.53
E6	52.6646	122.232	1.05	2.32	2.43
E7	42.5705	104.825	0.85	2.46	2.09
D3	44.4891	84.943	0.89	1.91	1.69
D4	57.8374	130.228	1.15	2.25	2.59
D6	58.1548	130.437	1.16	2.24	2.60
D7	43.6143	75.018	0.87	1.72	1.49
C3	42.8832	104.661	0.85	2.44	2.08
C4	52.4493	122.084	1.04	2.23	2.43
C5	59.0606	127.120	1.18	2.15	2.53
C6	52.6646	122.232	1.05	2.32	2.43
C7	42.5705	104.825	0.85	2.46	2.09
FUEL	50.2394	130.437			

All Blades 50% Withdrawn

Area	Average Power	Peak Power	Radial	Element	Total
E3	40.6936	116.990	0.81	2.87	2.33
E4	50.8722	135.749	1.01	2.67	2.70
E5	58.3491	149.086	1.16	2.56	2.97
E6	51.0585	135.917	1.02	2.66	2.70
E7	40.3526	117.121	0.80	2.90	2.33
D3	47.9444	97.535	0.95	2.03	1.94
D4	63.0423	153.222	1.25	2.43	3.05
D6	63.3316	153.422	1.26	2.42	3.05
D7	46.6823	87.388	0.93	1.87	1.74
C3	40.6936	116.990	0.81	2.87	2.33
C4	50.8722	135.749	1.01	2.67	2.70
C5	58.3491	149.086	1.16	2.56	2.97
C6	51.0585	135.917	1.02	2.66	2.70
C7	40.3526	17.121	0.80	2.90	2.33
FUEL	50.2609	153.422			

Blade worth calculations were performed with the regulating blade inserted in the core. The individual blade worths were not calculated because of flux shifting (tilting). It is important to note that the LEU shutdown margin calculation conservatively exceeds the RINSC Technical Specification of a minimum shutdown margin of 1% with blade 3 full out and the regulating blade full out. Table 4-5 shows the LEU reactivity data.

TABLE 4-5
LEU Reactivity Data and Power Peaking Factors

	Start-up Core	Core 2	Core 10
Excess Reactivity	3.1	4.1	4.0
Shutdown Margin	6.25	5.69	5.93
Reg. Blade Worth	0.41	0.41	0.47
Total Power Peaking Factor/Grid Position (Control Blades Full Out)	2.64/D6	2.6/D6	2.36/D6
Total Power Peaking Factor/Grid Position (Control Blades 50% Out)	3.06/D6	3.05/D6	2.81/D6

4.6 Thermal-Hydraulic Design

The thermal hydraulic studies for the RINSC reactor have been a joint effort by the RINSC staff and by Argonne National Laboratory (ANL). The RINSC primary and secondary coolant systems have been updated providing redundant coolant loops. Although not required for operation at 2 MW, two-loop operation is possible. Core flow and bypass flows were calculated. The analysis at 2 MW is for either loop #1 or #2 over a range of flows. For the primary loop (steady-state operation), the measured nominal flow is about 1,950 gpm for single pump operation.

The computer codes used for the analysis are PLTEMP, and NATCON. Data generation and code runs were performed by either RINSC or by ANL. The LEU core was rigidly remodeled by ANL with small inconsistencies corrected from previous analyses. Bypass flow was determined to be very important in determining actual core flows. Steady-state hot channel and natural convection computations were performed for the LEU startup core prior to conversion from HEU to LEU fuel. Additional calculations were performed for the larger 17-fuel element core.

The parameters used in the "PLTEMP" and "NATCON" programs were calculated using the LEU fuel element and the two core configurations. The physical dimensions of core components used were obtained from current drawings. In addition to the normal dimensions of core components used in the thermal-hydraulic analysis, the LEU fuel thermal conductivity was calculated and, the "Critical Velocity for Fuel Plate Deformation" was studied (See Reference 4-Y).

4.6.1 Hot Spot Factors

The use of LEU fuel elements necessitated an evaluation of the engineering hot spot factors to be used in the single channel analysis. The RINSC prepared a report entitled "Report on the Determination of Hot Spot Factors for the Rhode Island Nuclear Science

Research Reactor Using LEU Fuel (Reference 4-Y). This report yielded the following results:

F_b (Bulk Water Temperature Rise) = 1.62
F_q (Heat Flux) = 1.46
F_h (Heat Transfer) = 1.41

These factors were used in the following single channel (Hot Channel) analysis to determine a "limiting power level" based upon incipient boiling utilizing the PLTEMP Program.

4.6.2 Steady State Single Hot Channel Analysis

Computer runs using PLTEMP were run for the single channel analysis using the derived hot channel factors. Flow rates and power levels were varied to provide sufficient information for limiting power level and core flow determinations. The hot channel occurs with the blades in a 50% out condition, F2 = 1.536 in D-6 position. (From Table 4-5, the total power peaking factor = 3.06). The values in Tables 4-9 and 4-10 reflect this data as well as the engineering hot spot factors.

The tabulated results for axial factors of 1.32 (blades full out) and for axial factors of 1.536 (blades 50% out) are shown in Tables 4-9 and 4-10. The results are also presented as a "Hot Channel Fuel Surface Graph" depicting "Fuel Temperature" vs. "Total Core Flow." (See Figure 4-7).

TABLE 4-9
STEADY STATE HOT CHANNEL ANALYSIS
F AXIAL = 1.536

ΔP (MPA)	Total Flow (GPM)	T Surface 2 MW	T Surface 2.2 MW	T Surface 2.4 MW	T Surface 2.6 MW	T SAT	T onb
0.0025	1105	122.26	122.57	122.86	123.10	115.82	122.1
0.0030	1223	122.27	122.57	122.87	123.15	115.82	122.1
0.0035	1333	119.27	122.58	122.88	123.16	115.82	122.1
0.0040	1447	114.99	121.60	122.89	123.17	115.82	122.1
0.0045	1533	111.44	117.71	122.90	123.19	115.82	122.1
0.0050	1627	108.40	114.41	120.36	123.20	115.82	122.1
0.0055	1715	105.74	111.54	117.24	122.93	115.82	122.1
0.0060	1800	103.39	109.00	114.53	120.00	115.82	122.1
0.0065	1883	101.30	106.75	112.10	117.39	115.82	122.1
0.0070	1963	99.42	104.71	109.93	115.06	115.82	122.1

TABLE 4-10
STEADY STATE HOT CHANNEL ANALYSIS
F AXIAL = 1.32

ΔP (MPA)	Total Flow (GPM)	T Surface 2 MW	T Surface 2.2 MW	T Surface 2.4 MW	T Surface 2.6 MW	T SAT	T onb
0.0025	1105	120.80	122.09	122.36	122.62	115.82	122.1
0.0030	1223	115.01	121.52	122.37	122.63	115.82	122.1
0.0035	1333	110.36	116.52	122.38	122.64	115.82	122.1
0.0040	1447	106.55	112.40	118.16	122.65	115.82	122.1
0.0045	1533	103.38	108.94	114.44	119.86	115.82	122.1
0.0050	1627	100.67	105.99	111.26	116.48	115.82	122.1
0.0055	1715	98.30	103.43	108.49	113.51	115.82	122.1
0.0060	1800	96.21	101.18	106.07	110.90	115.82	122.1
0.0065	1883	94.36	99.17	103.91	108.58	115.82	122.1
0.0070	1963	92.69	97.36	101.97	106.51	115.82	122.1

The normal primary flow rate for the reactor is about 1950 gpm. From the data it can be seen that incipient boiling (T_{onb}) occurs at about 2.6 MW or 130% of the normal 2 MW power level. At a reduced flow of about 1700 GPM incipient boiling is reached at about 2.4 MW or 120% of normal power. The limiting safety settings are then set at the values as shown below:

Normal Power Level	Over Power Trip (scram)
2 MW	120% (2.4 MW)
Normal Flow	Reduced Flow Trip (scram)
1950 gpm	1700 gpm

The hot channel surface temperature of the fuel resulting at the 1580 GPM pump flow is from Table 4-9 about 110°C. The corresponding coolant velocity from the PLTEMP output for 1533 GPM (ΔP = 0.0045) = 1.44 M/S and for 1627 GPM (ΔP = 0.0050) = 1.53 m/s. An extrapolated value for the 1580 GPM condition is about 1.48 m/s.

Based on the average and hot channel studies the LEU fuel plates remain below ONB temperatures at all operating and scram settings and are acceptable for the startup and subsequent cores for 2 MW operation.

4.6.3 Steady State Full Core Analysis

The PLTEMP Program was used to analyze the full core for the axial peaking factors of 1.32 (blades full out) and 1.536 (blades 50% out) core conditions. The full core analysis included the various components (fuel elements, reflectors, baskets etc.) as shown in Figure 4-1. This analysis initially determines the flow rates for the fuel portion of the core, the by-pass flow through the other components and the total core flow versus the pressure drop across the core. This data is tabulated in Table 4-11. The fuel plate surface temperature versus flow rate is shown in Table 4-12. It is important to note the maximum fuel plate surface temperature does not vary by more than ±3.5 degrees centigrade for the two axial factors (F axial = 1.32 and 1.536). From the

TABLE 4-11
LEU FULL CORE ANALYSIS – 14 ELEMENT CORE 2 MW

DP (MPA)	Core Flow (kg/s)	By Pass Flow (kg/s)	Total Flow (kg/s)	Core Flow (gpm)	By-Pass (gpm)	Total Flow (gpm)
0.0040	57.14	33.66	90.80	910.8	536.2	1447
0.0045	61.07	35.13	96.20	973.5	559.5	1533
0.0050	64.83	37.21	102.04	1033.4	593.6	1627
0.0055	68.43	39.19	107.62	1090.8	624.2	1715
0.0060	71.89	41.09	112.98	1146.0	654.0	1800
0.0065	75.23	24.77	118.16	1199.0	684.0	1883
0.0070				1250.0	719.0	1963

TABLE 4-12

DP (MPA)	By-Pass Flow (gpm)	Core Flow (gpm)	Total Flow (gpm)	Outlet Bulk Temp °C F axial = 1.32	Plate Surface Temp °C F axial = 1.32	Outlet Bulk Temp °C F axial = 1.536	Plate Surface Temp °C F axial = 1.536
0.0040	536.2	910.8	1447	51.81	68.03	51.81	70.69
0.0045	559.5	973.5	1533	51.20	66.65	51.20	69.18
0.0050	593.6	1033.4	1627	50.69	65.29	50.69	67.89
0.0055	624.2	1090.8	1715	50.25	64.28	50.25	66.78
0.0060	654.0	1146.0	1800	49.86	63.56	49.86	65.80
0.0065	684.0	1199.0	1883	49.53	62.77	49.53	64.94
0.0070	719.0	1250.0	1969	49.23	62.07	49.23	64.16

Notes: (1) Normal primary pump operation 1950 gpm
(2) Calculations based on inlet temperature to core of 42.3°C
(3) Fuel element data based on the radial factor of 1.18

tabulated data and our pump flow of about 1950 gpm, the core flow (1243 gpm ±) and a by-pass flow (707 gpm ±) was determined. The corresponding ΔP = 0.0055 MPA. The results are graphically depicted as LEU core "Flow" vs "DP." (See Figure 4-8).

The output of the program also determines a number of other parameters. A list of these for the steady state 2 MW operation case is shown in Table 4-13. This "full core" computer run produces this type of data for all of the 14 fuel elements. Table 4-13 shows typical output data for the element #8 (D-6 core position). A program option can also produce data along a selected "single channel" of a fuel element. The axial peaking factor vs. relative blade position for the core is tabulated for both the blades full out condition (F axial = 1.32) and the blades 50% out (F axial = 1.536). This data was obtained from the nucleonic studies of the core described in Section 4.5. Table 4-14 gives the power fraction and peaking factors of each element. Table 4-15 shows the axial distribution for the blades out and blades 50% out conditions. This data was input to the PLTEMP Program to calculate the various parameters at a point-by-point basis along the axial plate length. Table 4-16 shows the output data for a typical single channel (not a hot channel) for an arbitrarily selected element at the normal core flow

(DP = 0.0055 MPA). It should be noted that the highest power peaking factor occurs in core position D-6, for both blades full out and blades 50% out situations.

TABLE 4-13

The PLTEMP full core analysis for each fuel element calculates additional parameters. Typical output values are shown for these parameters. (Core Position D-6 data)

PARAMETER		VALUE
Average Surface Temperature	°C	64.45
Clad Temperature (average)	°C	64.78
Channel Flow Rate	(kg/s)	0.2298
Velocity	(m/s)	1.605
Outlet Pressure	(MPA)	0.1727
CHF (Critical Heat Flux)	(MW/M^2)	2.39
Flow Instability	(MW/M^2)	0.889
Exit Saturation Heat Flux	(MW/M^2)	1.087
Minimum DNB Ratio		8.461
F axial		1.32
F radial		1.18
Average Heat Flux	(MW/M^2)	0.100241
DP	(MPA)	0.0055

Core position D-6 has the highest fraction of power generation (0.084) in the core. (F radial = 1.18)

TABLE 4-14

FRACTION OF POWER IN FUEL	CORE POSITION	PEAKING FACTORS
0.0613334	E3	0.86
0.0734574	E4	1.03
0.0827287	E5	1.16
0.0741706	E6	1.04
0.0606202	E7	0.85
0.0648993	D3	0.91
0.0828908	D4	1.17
0.0841551	**D6**	**1.18**
0.0634729	D7	0.89
0.0613334	C3	0.86
0.0734574	C4	1.03
0.0826898	C5	1.16
0.0741706	C6	1.04
0.0606202	C7	0.85

TABLE 4-15
AXIAL DISTRIBUTION

INTERFACE	RELATIVE DISTANCE	BLADES FULL OUT	BLADES 50% OUT
1	0.00	0.4105	0.0553
2	0.05	0.5506	0.2682
3	0.10	0.6836	0.4759
4	0.05	0.8076	0.6744
5	0.20	0.9219	0.8597
6	0.25	1.0228	1.0283
7	0.30	1.1111	1.1770
8	0.35	1.1850	1.3028
9	0.40	1.2435	1.4032
10	0.45	1.2858	1.4764
11	0.50	1.3200	1.5360
12	0.55	1.2858	1.4764
13	0.60	1.2435	1.4032
14	0.65	1.1850	1.3028
15	0.70	1.1111	1.1770
16	0.75	1.0228	1.0283
17	0.80	0.9219	0.8597
18	0.85	0.8076	0.6744
19	0.90	0.6836	0.4759
20	0.95	0.5506	0.2682
21	1.00	0.4105	0.0553

TABLE 4-16
TYPE 1, ELEMENT 8

	PLATE 21, SIDE 2				**CHANNEL 22**				**PLATE 22, SIDE 1**
	Q-MW/M2	T°C	T°C	H-W/M2/K	T°C	H-W/M2/K	T°C	T°C	Q-MW/M2
HEIGHT	HT FLUX	FUEL	CLAD	<--->	COOLANT	<--->	CLAD	FUEL	HT FLUX
0.0148	0.0191	44.27	44.20	10285	42.34	10285	45.53	45.64	0.0338
0.0443	0.0440	46.91	46.76	10299	42.49	10299	49.82	50.07	0.0755
0.0738	0.0690	49.56	49.33	10323	42.74	10323	54.04	54.44	0.1167
0.1033	0.0906	52.16	51.85	10357	43.10	10357	58.11	58.63	0.1554
0.1329	0.1115	54.66	54.29	10402	43.57	10402	61.96	62.60	0.1913
0.1624	0.1304	57.03	56.60	10450	44.12	10430	65.52	66.27	0.2237
0.1919	0.1466	59.20	58.71	10503	44.75	10503	68.70	69.54	0.2515
0.2214	0.1600	61.13	60.60	10563	45.45	10563	71.43	72.35	0.2745
0.2510	0.1703	62.79	62.22	10628	46.20	10628	73.68	4.66	0.2921
0.2805	0.1781	64.24	63.65	10699	47.00	10699	75.55	6.58	0.3055
0.3100	0.1781	64.94	64.34	10772	47.80	10772	76.17	7.20	0.3055
0.3395	0.1703	64.88	64.30	10845	49.60	10845	75.54	6.52	0.2921
0.3691	0.1600	64.55	64.02	10914	49.36	10914	74.50	5.43	0.2745
0.3986	0.1466	63.91	63.42	10971	50.06	10971	72.98	3.83	0.2515
0.4281	0.1304	62.96	62.52	11023	50.69	11023	70.98	1.73	0.2237
0.4576	0.1116	61.70	61.33	11069	51.24	11069	68.54	9.18	0.1915
0.4872	0.0907	60.17	59.87	11107	51.70	11107	65.71	6.23	0.1556
0.5167	0.0680	58.40	58.17	11138	52.06	11138	62.54	2.93	0.1167
0.5462	0.0440	56.41	56.26	11160	52.32	11160	59.08	9.34	0.0755
0.5757	0.0191	54.24	54.18	11172	52.46	11172	55.40	5.51	0.0328
0.5905	0.0065	53.11	53.09	11176	52.51	11176	53.51	3.55	0.0112
	(OUTLET)								

4.6.4 Rising Power Transient Analysis

Since the plate temp program is a steady state analysis program, the determination of a transient involving a rising power scram is not possible. However, with a few simple assumptions, an estimate can be made of the fuel surface temperature at the projected power level at the scram condition.

RISING POWER TRANSIENT CALCULATION

Assumptions:

- The reactor period rises on a period of slightly greater than the blade rundown override of 7 seconds.
- Reactor power increases until the 2.4 MW scram setting is reached, and the blades drop to the full-in position within 1.0 second.

The actual trip power level can be calculated as follows:

$$P = P_0 e^{T/\tau} = 2.4e^{(1/7)} = 2.4(2.718)^{.1429} = 2.78\ MW$$

From Table 4-9, a power level of 2.6 MW has a calculated surface temperature of 122.93°C, which is slightly above ONB (normal flow). The expected temperature at 2.78

MW would be slightly higher, but for a hot channel analysis, the ONB region would not present a problem for the LEU fuel.

4.6.5 Natural Convection

The Natural Convection Analysis for the LEU core was performed using the NATCON Program. The program was run for both the regular channel and the "hot channel" conditions. Both cases were run for the blades full out situation (F axial = 1.32) and the blades 50% out (F axial = 1.536). The results are shown in Table 4-17.

TABLE 4-17
NATURAL CONVECTION

REGULAR CHANNEL **BLADES 50% OUT, FAXIAL = 1.536**

Power Level (kW)	Exit Temp. °C	Maximum Wall Temp. °C	Incipient Boiling °C	T sat-T wall (117.34-Tw)	Radial Peaking Factor	Margin to Incipient Boiling °C
10	51.93	52.01	117.57	65.34	2	65.56
100	68.41	71.22	118.28	46.12	2	46.06
200	77.29	84.71	118.88	32.63	2	34.17
300	83.80	96.48	119.21	20.86	2	22.73
500	93.69	117.85	119.85	-0.51	2	2.00
520.4	94.56	119.89	119.89	-2.55	2	0.00

HOT CHANNEL

10	59.18	58.89	117.72	58.45	2	58.83
100	86.63	92.81	118.56	24.53	2	25.75
209.1	102.11	119.34	119.20	-2.00	2	-0.40

REGULAR CHANNEL **BLADES FULL OUT, FAXIAL = 1.32**

Power Level (kW)	Exit Temp.	Maximum Wall Temp. °C	Incipient Boiling °C	T sat-T wall (117.34-TW)	Radial Peaking Factor	Margin to Incipient Boiling °C
10	52.04	52.26	117.60	65.08	2	65.34
100	68.72	71.58	118.19	45.76	2	46.61
200	77.69	84.25	118.72	33.09	2	34.47
300	84.26	95.11	119.00	22.23	2	23.89
400	89.61	105.28	119.35	12.06	2	14.07
500	94.24	114.45	119.67	2.89	2	5.22
558.45	96.69	119.80	119.80	-2.46	2	0.00

HOT CHANNEL

10	59.38	59.80	117.65	57.54	2	57.85
100	87.12	92.92	118.43	24.42	2	25.51
217.3	103.66	119.14	119.03	-1.80	2	0.00

Note that for the most conservative case, (hot channel) the power level is 209.1 kW, using incipient boiling as the limiting parameter. The maximum wall temperature was calculated as a function of axial length and the value was tabulated form the data. The program run terminates when the fuel surface temperature reaches incipient boiling.

Since the 209.1 kW exceeds the current licensed power level of 100 kW for natural convection, no change is deemed necessary in the licensed maximum natural convection power level of 100 kW.

4.7 Improved Flux Core Modification

The main objective in the modification of the RINSC core was to find a technically and economically feasible model that, without changing the operating power, would increase the leakage of fast neutrons from the core to the thermal column for research and other reasons. Several options were considered, including removing graphite reflectors near the thermal column and moving the fuel matrix toward the thermal column. However, many of these options were found to be technically infeasible and overly expensive. A suggestion by the ANL RERTR group eventually lead to an expanded core design, in which three graphite reflector elements near the thermal column would be replaced by three fresh LEU fuel elements (see Figure 4-1). Such an arrangement leads to a significant increase in neutron leakage from the core into the thermal column.

4.8 Thermal Analyses

Two fresh core models were chosen for the thermal hydraulic calculations that would produce highest possible fuel and cladding temperatures under normal reactor operation at 2 MW reactor power. The upper diagram in Figure 4 represents the original RINSC startup core, while the lower diagram represents an expanded startup core with the three additional fuel elements on the thermal column side.

A full core model of the RINSC reactor was run using the PLTEMP[6] code in an iterative manner to determine the pressure drop that would yield the flow rate corresponding to one pump operation. A single plate model for the hot channel was then run to determine the peak fuel, clad and coolant temperatures. The coolant inlet temperature was assumed to be 42.2°C (108°F) because it is the design inlet temperature. The hot channel factors, bypass flow geometry, and axial power distribution remain unchanged from current operating conditions in the 14 fuel element core.

Figure 4-5 shows the power peaking factors obtained from the DIF3D diffusion theory calculations with the control blades 50% withdrawn. From the point of view of thermal-hydraulic safety margins, the most important neutronic parameter is the total 3D power peaking factor (the absolute peak power density in a fuel assembly divided by the average power density in the core). The total power peaking factor is defined here as the product of three components: (1) a radial factor defined as the average power density in each assembly divided by the average power density in the core, (2) a planar factor defined as the peak power density in the limiting plane in each fuel assembly divided by the average power density in that plane, and (3) the axial peak to average power density in the assembly defined as the maximum ratio of the peak to average power density in the plate. The planar and axial factors are point-wise factors computed at the mesh interval edge and includes both planar and axial power peaking.

Tables 4-18 and 4-19 show the results of the thermal analysis for both core configurations, with the control blades withdrawn 50% and fully out. There is a substantial margin to boiling and flow instability in all cases.

Table 4-18
Power Peaking Factors and Thermal Hydraulic Parameters in Hot Channel Calculations

No. of Elements in Core	Control Rod Position	Peaking Factor Radial	Planar	Axial	Total	Pressure Drop Mpa	psi	Coolant Flow Rate gpm
14	50% Out	1.258	1.562	1.536	3.02	0.00677	0.98	1870
14	Out	1.178	1.695	1.32	2.64	0.00677	0.98	1870
17	50% Out	1.424	1.493	1.536	3.27	0.00677	0.98	2195[1]
17	Out	1.330	1.621	1.32	2.85	0.00677	0.98	2195[1]

[1]Extimated for 17 element core.
F_b (hot channel factor for bulk water temperature rise) = 1.25.
F_q (hot channel factor for heat flux) = 1.26.
F_h (hot channel factor for heat transfer coefficient) = 1.33.

TABLE 4-19
Safety Margins and Peak Temperatures in Hot Channels

No. of Elements in Core	Control Rod Position	Margin to Onset of Nucleate Boiling	Margin to Flow Instability	Margin to Departure from Nucleate Boiling	Maximum Temperatures °C/°F Fuel	Clad	Coolant
14	50% Out	1.80	4.50	6.96	89.5/193	88.4/191	52.8/127
14	Out	2.14	5.14	7.96	84.8/185	83.8/183	53.1/128
17	50% Out	2.13	5.03	7.80	84.9/185	84.0/183	52.1/126
17	Out	2.39	5.78	8.95	80.6/177	79.8/176	52.3/126

Replacing three graphite reflectors with fuel elements would lead to lower average fuel, cladding, and coolant temperatures, because the reactor power would be spread over more fuel elements, and because there would be an increase in the total coolant flow through the core due to the increase in the number of flow channels available. Results of the thermal analysis show there is substantial margin for boiling and flow instability in all cases.

The addition of three fuel elements near the stainless steel regulating blade in position D2 would cause an increase in the blade worth. The safety specifications require the worth of the regulating blade to be less than β (the prompt neutron fraction).

Calculations were performed to predict the worth of the regulating blade in the expanded core. Results of these calculations are summarized in Table 4-20. The calculated worth of the regulating blade in the expanded core is estimated to be about 0.5%Δk/k. The determination of the actual effect that the expanded core configuration would have on the regulating blade can only be made by making the core change, and performing a control blade calibration. If the calibration were to show that the regulating blade had a measured worth close to the safety limit, the reflector elements could be rearranged to reduce the worth of the regulating blade.

TABLE 4-20
Regulating Rod Worth in the Expanded Core

	DIF3D Δk/k	MCNP Δk/k
CORE 2 EOC		
Equilibrium Xe	0.281%	0.198 ±0.071%
No Xe	0.387%	
Expanded Core with 3 new fuel elements		
No Xe	0.586%	0.508 ±0.056%

Table 4-21 shows total excess reactivity in terms of K_{eff}. The initial estimates were made with diffusion theory calculations, which are in good agreement with results of more detailed Monte Carlo calculations. Both computational methods suggest that the expanded core would have approximately 1.6%Δk/k greater excess reactivity at equilibrium Xenon, than the present core under the same conditions, and that the expanded core would have a total excess reactivity of 4%Δk/k with no Xenon poisoning in the core.

TABLE 4-21
K_{eff} obtained from DIF3D and MCNP Calculations

	DIF3D	MCNP
Core 2, EOC, equilibrium Xe	1.00028	0.99312 ±0.00049
Expanded Core, 3 New Fuel, equilibrium Xe	1.01689	1.00915 ±0.00052
Expanded Core, 3 New Fuel, No Xe	1.04402	1.04250 ±0.00054

Table 4-22 shows the shutdown margin calculation for the expanded core. The calculations took into account the depletion of the fuel in core 2 and assumed that the core was at room temperature, with no Xenon poisoning, and that an experiment worth limit of +0.6%Δk/k was present at the core. The most reactive control blade and regulating blade were assumed to be fully out of the core. The estimated shutdown margin of 4.83%Δk/k is far more conservative than the 1%Δk/k technical specification limit.

TABLE 4-22
Reactor Shutdown Margin in the New Expanded Core

	Reactivity Worth
	Δk/k
Core excess reactivity (cold/no Xe)	4.1%
Worth of control rods with the most reactive rod (CR2) and regulating rod fully out	-9.8%
Worth of experiments	0.6%
Shutdown Margin =	-5.1%

4.9 Conclusions

Expanding the RINSC core configuration to include three fuel elements in place of graphite reflector elements along the thermal column side of the core is expected to increase fast neutron leakage into the thermal column by over 300%. The worth of the current regulating blade would increase from 0.2%Δk/k to about 0.5%Δk/k. The estimated core excess reactivity for this configuration is about 4.1%Δk/k without Xenon poisoning. The predicted shutdown margin is about 4.8%Δk/k subcritical with the assumption that the core is loaded with a maximum positive worth experiment. All of these reactivities are within the limits imposed by the facility license.

4. 10 References

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3. J. F. Briesmeister, "MCNP-A General Monte Carlo N-Particle Transport Code," LA-12625-M, Los Alamos National Laboratory, Nov. 1993.

4. J. R. Deen, W. L. Woodruff and C. I. Costescu, "WIMS-D4M User Manual Rev. 0", ANL/RERTR/TM-23, Argonne National Laboratory, July 1995.

5. B. J. Toppel, "A User's Guide for the REBUS-3 Fuel Cycle Analysis Capability," ANL-83-2, March 1983.

6. K. Mishima, K. Kanda and T. Shibata, "Thermal-hydraulic Analysis for Core Conversion to Use of Low Enrichment-uranium Fuels in KUR," KURRI-TR-258 (1984).

Figure 4-1

14 Element Core

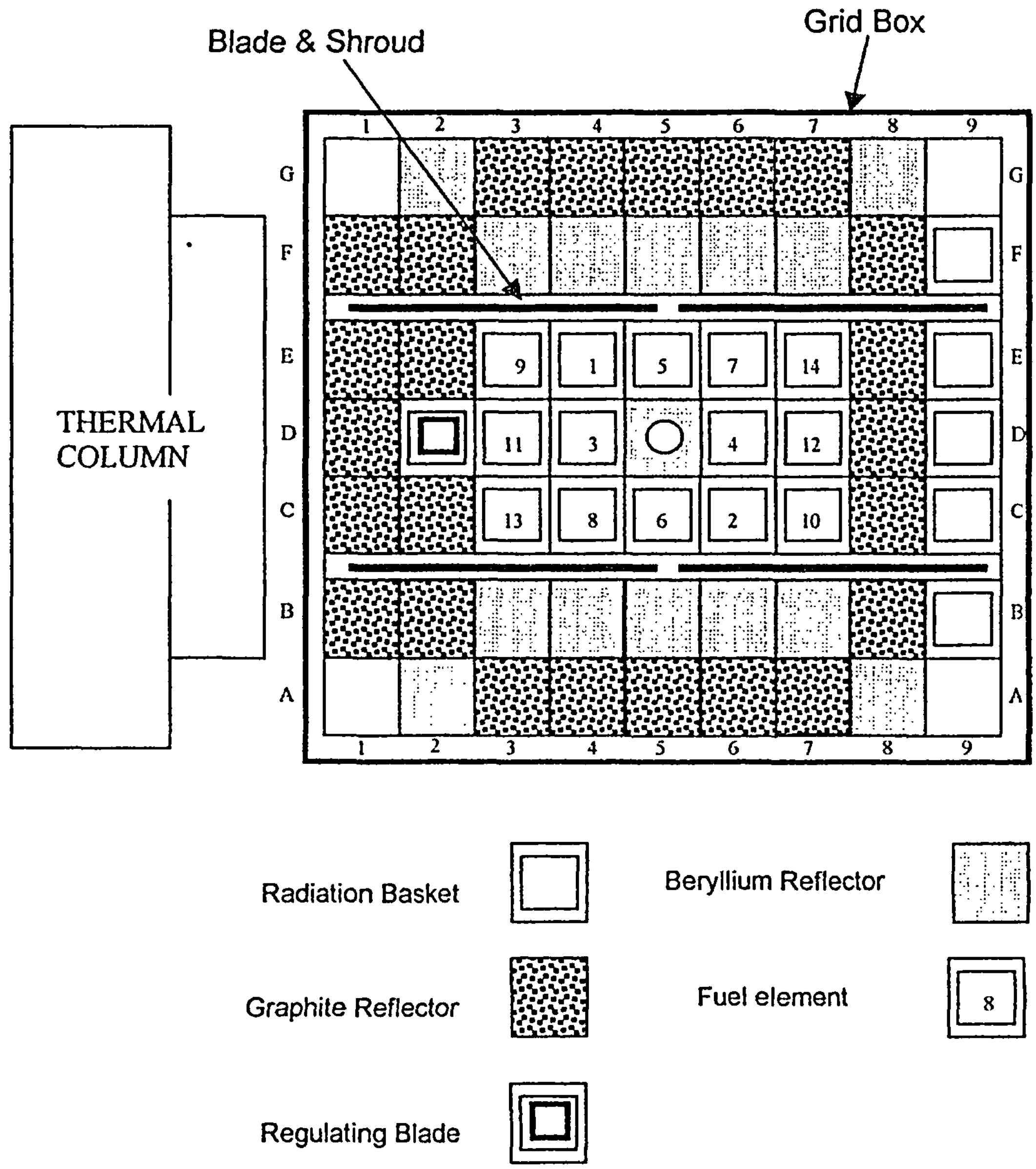


Figure 4-2

17 Fuel Element Core

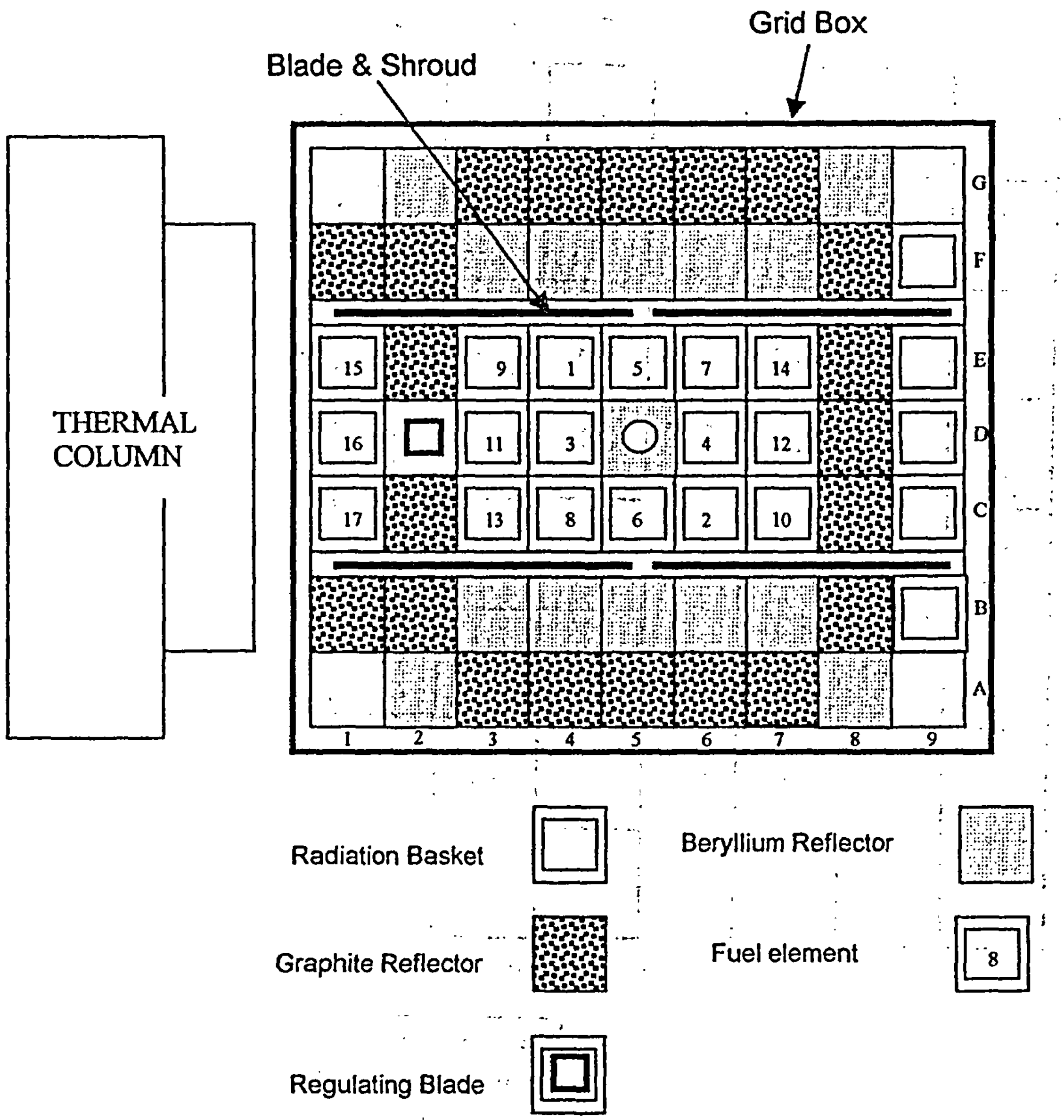


Figure 4-3
Power Peaking Factors in the Thermal Hydraulic Calculations

E	D	C	
G	G	G	8
0.607 1.768 1.536 2.192 5.8%	0.930 1.169 1.536 1.670 6.6%	0.807 1.768 1.536 2.192 5.6%	7
1.011 1.630 1.536 2.531 7.2%	1.261 1.563 1.536 3.026 9.0%	1.011 1.630 1.536 2.531 7.2%	6
1.151 1.654 1.536 2.925 8.2%	Be	1.151 1.654 1.536 2.925 8.2%	5
1.009 1.632 1.536 2.530 7.2%	1.258 1.563 1.536 3.019 9.0%	1.039 1.632 1.536 2.530 7.2%	4
0.818 1.749 1.536 2.197 5.8%	0.961 1.312 1.536 1.936 6.9%	0.818 1.749 1.536 2.197 5.8%	3
G	Reg Rod	G	2
G	G	G	1

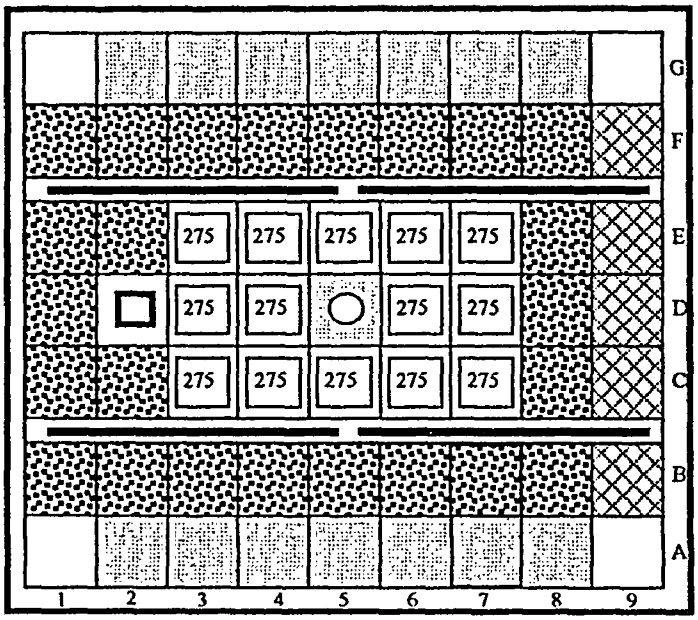
Startup Core

E	D	C	
G	G	G	8
0.823 1.766 1.536 2.231 4.8%	0.944 1.191 1.536 1.727 5.6%	0.823 1.766 1.536 2.231 4.8%	7
1.052 1.631 1.536 2.635 6.2%	1.317 1.572 1.536 3.173 7.7%	1.052 1.631 1.536 2.635 6.2%	6
1.239 1.640 1.536 3.122 7.3%	Be	1.239 1.640 1.536 3.122 7.3%	5
1.134 1.591 1.536 2.770 6.7%	1.413 1.500 1.536 3.256 8.3%	1.134 1.591 1.536 2.770 6.7%	4
0.993 1.654 1.536 2.538 5.9%	1.180 1.410 1.536 2.556 6.9%	0.999 1.654 1.536 2.538 5.9%	3
G	Reg Rod	G	2
0.526 2.228 1.536 1.800 3.1%	0.602 2.073 1.536 1.917 3.5%	0.526 2.228 1.536 1.800 3.1%	1

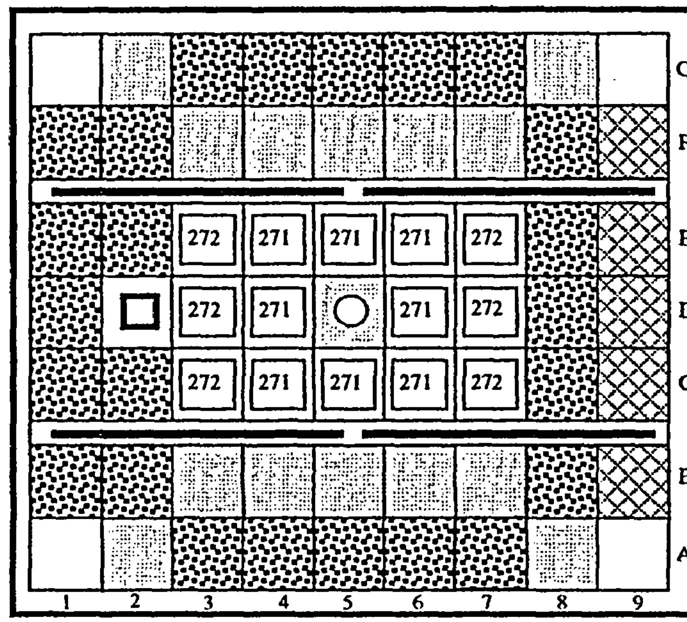
Startup Core with 3 more Fuel Elements

Radial peaking factor
Planar peaking factor
Axial peaking factor
Total peaking factor
% of total power in element

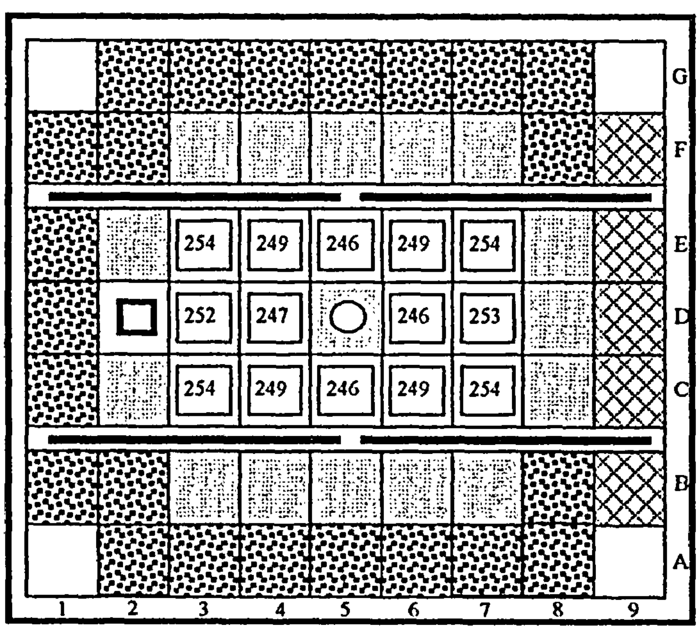
STARTUP CORE
Excess Reactivity = 3.0%Δk/k
Core Lifetime: ~14 Wks (~560 Full Power Hrs)



CORE 2
Excess Reactivity = 4.1%Δk/k
Core Lifetime: ~70 Wks (~2800 Full Power Hrs)



CORE 3
Excess Reactivity = 3.7%Δk/k
Core Lifetime: ~60 Wks (~2400 Full Power Hrs)



EQUILIBRIUM CORE
Excess Reactivity = 3-4%Δk/k
Core Lifetime: ~57 Wks (~2300 Full Power Hrs)

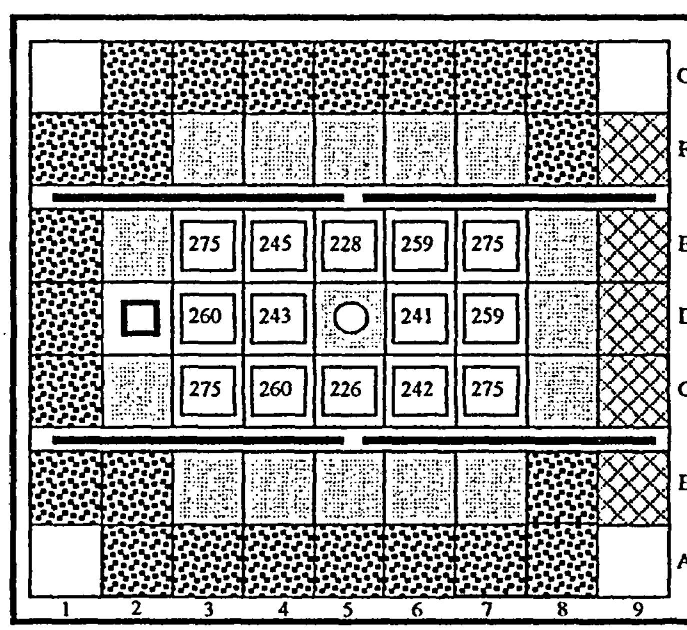


Figure 4-4

CHAPTER FIVE

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LIST OF FIGURES

Chapter 5

5.0 REACTOR COOLANT SYSTEMS

5.1 Summary Description

The RINSC reactor is an open pool type reactor that uses de-mineralized water for primary coolant, shielding, and reactor moderator; and city water for secondary coolant. The core may be cooled by natural convection at power levels up to 0.1 $MW_{(t)}$, and is cooled by forced convection at power levels above 0.1 $MW_{(t)}$. In natural convection mode cooling, heat from the core is transferred to the primary cooling water in the pool, where it is dissipated to the surrounding environment. In forced convection mode cooling, heat is transferred from the primary cooling water to two heat exchangers, which pass the heat to secondary cooling water loops, which in turn, dissipate the heat to the surrounding environment via two cooling towers. Past operating history has shown that only one cooling loop is necessary to provide sufficient heat removal for 2 $MW_{(t)}$ operation.

Figure 5-1 shows a cutaway view of the RINSC reactor pool. Since the pool is open to the atmosphere, the depth of water acts as radiation shielding to personnel at the pool top. The primary make-up system maintains the pool water level so that it is approximately 25 feet above the core. Makeup water is supplied from the normal city water supply, and is passed through a de-mineralizing system as generally described in Section 5.5. Pool water quality is maintained with a dedicated cleanup system as generally described in Section 5.4.

5.2 Primary Coolant System

5.2.1 Primary Coolant System Components

5.2.1.1 Reactor Pool

Figure 5-2 shows the reactor pool, which consists of three sections. The first section is a circular high-power section, which opens into a rectangular fuel storage section, which in turn opens into another rectangular low-power section. The sections can be separated by divider openings that allow for individual draining of either the high or low power section. The pool is lined with a 1/4-inch aluminum liner to minimize water leakage into the concrete walls. The dimensions of each of the sections are:

[redacted]

A pool divider gate opening is located between each pool section. The openings are [redacted] restrictions in the width of the pool. The gate is constructed of watertight aluminum, and is capable of covering the gate openings to permit independent draining of either of the power sections.

There are no penetrations in the pool floor. The pools walls of the high power section are penetrated by the primary coolant inlet and outlet connections, six beam ports, a through port, two rabbit tubes, and a thermal column. The thermal column penetration is at [redacted] The fuel storage section and low power section walls have no penetrations. All penetrations are seal welded.

The concrete walls of the pool and the water within serve as the biological shield for personnel in the reactor building. Consequently, in order to prevent the possibility of accidentally siphoning water out of the pool, the primary coolant inlet and outlet lines have anti-siphon loops to prevent siphoning through the primary coolant lines in the event of a piping failure. Each anti-siphon loop consists of a three-inch line that goes from the respective inlet or outlet line in the concrete, to above the water level, and then out through the pool wall. The anti-siphon loop associated with the inlet line has a check valve on it to prevent water from the inlet line from bypassing the inlet flow channel by going through the loop. The loop associated with the outlet line extends down below the water surface to prevent air from getting sucked into the line when the primary pump(s) is (are) running. If the primary system has a piping failure that causes a siphon to start, as the water level falls, air will be drawn through the anti-siphon loops and break the siphoning action.

A dry irradiation facility is encased in the pool wall at the end of the low power section (east end of the pool). There is no penetration between the wall and the room. Reactor operation up to 0.1 $MW_{(t)}$ is permitted in this section, to provide gamma and neutron radiation to this facility. Also, spent fuel elements may be used to supply gamma radiation to the facility.

The core is suspended from the top of the pool, and may be positioned anywhere along the center line of the pool. Operation using forced convection is only possible in the high power section, where a connection is made between the primary inlet and outlet coolant pipes, and the coolant flow channels of the suspension frame.

5.2.1.2 Coolant Flow Channels

The primary inlet and outlet pipes from the primary cooling loops extend through the west end of the high power section pool wall, 12 feet below the surface of the pool water. The pipe ends are fitted with 10-inch pressure couplings. When the core is positioned in this section, the couplings mate with corresponding inlet and outlet flow channel connections. The flow channels are made of aluminum, and are positioned on opposite sides of the core suspension frame, as shown in Figure 5-3. The inlet flow channel opens to the pool, just above the top of the core. The outlet channel draws from a plenum below the core.

Each flow channel is equipped with a coolant header gate, which is a hinged gate between the flow channel and the open pool. The gates are located approximately two feet below the couplings to the inlet and outlet pipes, on the west faces of the flow channels. The inlet channel gate opens into the channel so that at the designed coolant flow rate, the gate is pushed closed by the force of the water against it. When flow drops below 1,000 gpm, a counter weight causes the gate to open. The outlet channel gate opens to the outside of the channel, and it has a paddle that extends into the channel flow stream. When coolant is flowing through the channel at the designed coolant flow rate, the gate is forced closed by water flow against the paddle. The weight of the paddle causes the gate to open under low flow conditions. Both gates are connected to float switches that change state when their respective gate changes state, providing an indication of low flow through the channels.

5.2.1.3 Delay Tank

Primary cooling water from the pool outlet pipe flows to a 3000-gallon delay tank, which delays the coolant water sufficiently for the N^{16} radioactivity in the water to decay. The inlet and outlets to the tank have a baffle plate to reduce mixing of the incoming water with the water that is ready to exit the tank. The tank is made of aluminum. The top of the tank is vented through a 3 inch diameter aluminum pipe extending above the pool water level, and into the suction line of the reactor room exhaust blower, which discharges to the stack. There is a manual gate valve on the delay tank inlet, and manual butterfly valves in the primary pump loops so that the delay tank can be isolated if needed. Coolant flow from the delay tank outlet is directed to the suction side of the primary pumps. Refer to Figure 5-4.

5.2.1.4 Primary System Cooling Loops

From the delay tank, the primary cooling system is divided into two loops. Each loop consists of a primary pump, and a heat exchanger. The two loops circulate primary water independently, from the delay tank to the heat exchanger associated with the loop. From the heat exchangers, the loops merge to form the primary inlet line that feeds the cooled water to the reactor pool. Refer to Figure 5-4. Since the primary cooling system is a closed loop system, any radioactive materials that diffuse into the primary coolant are confined to the pool, delay tank, and primary loops, while allowing heat transfer to the non-radioactive secondary side of the heat exchangers. Past operating history has shown that single loop cooling is sufficient for 2 $MW_{(t)}$ operation of the reactor.

5.2.1.5 Primary Cooling Pumps

The primary pumps are located in the primary system between the delay tank and the heat exchangers. They are stainless steel construction. Individual pump measured flows are approximately 1,950 gpm, and combined flow is approximately 3,600 gpm with both units running. Past operating history has shown that the location of the primary pumps downstream of the delay tank provides an adequate means of minimizing radiation damage to the pump seals.

5.2.1.6 Heat Exchangers

The loop #1 heat exchanger is a plate type, single pass unit that is sized to transfer equivalent heat for operation up to 3 $MW_{(t)}$. It is stainless steel construction, rated at 2,000 gpm on the primary side, and 1,600 gpm on the secondary side.

The loop #2 heat exchanger is a shell and tube exchanger, sized to transfer equivalent heat for operation up to 2.5 $MW_{(t)}$. It is stainless steel construction, rated at 1,500 gpm on both, the primary and secondary sides.

Past operating history has shown each of these to provide sufficient heat removal for 2 $MW_{(t)}$ operation of the reactor, independently of each other.

5.2.2 Primary Coolant System Operation

5.2.2.1 Natural Convection Mode (≤ 0.1 MW Operation)

In natural convection cooling mode, the rise of heated water through the core causes water to be drawn from the pool, into the outlet flow channel through the open flow channel gate, down through the channel to the plenum beneath the core, and back up through the core. Any water rise in the inlet channel is directed back into the pool through the open flow coolant header gate.

5.2.2.2 Forced Convection Mode (> 0.1 MW Operation to 2 MW)

In forced convection cooling mode, the inlet flow channel takes cooled water from the inlet pipe, and guides it down to the top of the core, where it is drawn through the core as a result of suction on the outlet flow channel, provided by the primary pumps. Heat from the reactor core is transferred to the water, which is drawn into the plenum beneath the core, and directed up through the outlet flow channel and, into the outlet pipe. From the outlet pipe, the primary water is circulated to the delay tank, where circulation is delayed for approximately 90 seconds. From the delay tank, the water is directed through the cooling pumps to the primary heat exchangers where the heat from the primary coolant is transferred to the secondary coolant. Primary water from the heat exchangers merges into the inlet pipe, which carries the water back to the pool.

5.3 Secondary Coolant System

5.3.1 Secondary Coolant System Components

5.3.1.1 Secondary Coolant Pumps

The secondary coolant system transfers the heat from the primary coolant system to the cooling towers. Refer to Figure 5-5. The secondary coolant pump for each cooling loop is located in the return line from the cooling tower to the heat exchanger. Operating data indicates that the loop #1 pump provides a flow rate of approximately 1100 gpm, while the pump for loop #2 provides a flow rate of approximately 1300 gpm. These flow rates, in conjunction with the heat removal capacities of the cooling towers, have proven to be sufficient for removing enough heat from the primary system to operate the reactor at 2 $MW_{(t)}$, during the summer, with only one cooling loop running.

5.3.1.2 Cooling Towers

The cooling tower for loop #1 is an induced cross flow unit with vertical air discharge through a belt driven axial flow fan. The fan is equipped with a 20 hp (1800 rpm) main motor and a 5 hp (1200 rpm) secondary motor. This allows for two air-flow speeds to control temperature. Principal structural construction is of type 304 stainless steel angles and channels. The casing is made of fiber reinforced polyester panels. The cold water basin is also fiber reinforced polyester panels, with a hot dip galvanized steel support structure. The wet deck and eliminators are PVC. The unit is rated for cooling water from 103.1°F to 90.2°F at a wet bulb ambient temperature of 76°F, at a flow rate of 1600 gpm.

The cooling tower for loop #2 is a counter-flow, induced draft, cooling tower. The induced draft fan has a 20 hp variable speed drive motor for temperature control. The construction is largely fiber reinforced polyester and PVC. The unit is rated for cooling water from 106.8°F to 86.3°F at a 77°F ambient wet-bulb temperature, at a flow rate of 1000 gpm.

5.3.2 Secondary Coolant System Operation

Each primary cooling loop has a secondary loop associated with it that begins at the secondary side of the heat exchanger, and has a secondary pump and a cooling tower. Heat from the primary side of the heat exchanger is transferred to the secondary coolant, which is pumped out to the cooling tower, where the heat is dissipated to the atmosphere through evaporation cooling. City water is used as secondary coolant for both loops.

5.4 Primary Coolant Cleanup System

The primary coolant cleanup system maintains the water purity in the primary system to reduce the potential for the development of corrosion on the core components. The system circulates water from the 10 inch pool inlet line, through the cleanup pump, through a mixed bed de-mineralizer, and back into the pool through the makeup / cleanup return line. Water conductivity and pH are measured to ensure that high water quality is maintained. Historically, conductivity has been limited to a maximum of 2 micro-ohm/cm, and pH has been limited to a range between 5.5 and 7.5. This has proven to be sufficient for minimizing corrosion. When the de-mineralizer resin has become spent, as indicated by an increase in conductivity, the old resin is removed from the de-mineralizer, and is replaced with new resin. Bulk pool temperature is limited to less than 140°F to avoid damaging the resin. Quarterly isotopic analysis is performed on the primary water, in order to verify that none of the fuel elements are leaking.

In addition to the de-mineralizer loop, there is also a surface filtration system that skims water from the surface of the reactor pool through a filter to remove any dust and debris that gets into the pool and floats on the surface.

5.5 Makeup Water System

5.5.1 Primary Makeup Water System

The primary makeup system provides a supply of purified water to replace water losses in the primary coolant system. Refer to Figure 5-6. The system is supplied from city water, which enters through a backflow-preventer, and is then split into two separate systems. The water in each system goes through a five micron filter, an activated charcoal filter, two mixed bed de-mineralizers, a one micron filter, and a conductivity indicator, before merging into the makeup / cleanup return line. The two systems are independent of each other, so that one line can be used until it no longer provides the desired water purity, at which time it can be isolated, and the other system can be used. The makeup water system is controlled by a float switch, located at the surface of the reactor pool. When the switch senses a drop in the water level of one inch, it opens a solenoid valve in the makeup line to allow water flow into the pool. When the float senses that the pool is full, it closes the solenoid valve to shut the water supply off. There is a manual bypass valve that can be used to control the water flow in the event that the solenoid valve fails. There is also a check valve to prevent primary water from back flowing through the make up system.

5.5.2 Secondary Makeup Water System

City water supplies the makeup water to the secondary coolant system. Each cooling tower basin has a resistive level sensor that opens a solenoid valve to provide makeup water flow into the basin. When the basin is full, the level sensor closes the valve to the supply. Both cooling towers also have a solenoid valve controlled drain line. The drain valves are on timers that can be set to open at desired intervals to provide blow-down. Historically the blow-down interval has been set such that the pH of the secondary water has been maintained between 5.5 and 9.0, which has kept mineral buildup and corrosion to a minimum. The drain lines from the towers merge, and the water is either directed to the storm drain, or to a 3000 gallon holding tank, depending on whether or not re-use of the water is desired.

5.6 Nitrogen-16 Control System

The buildup of nitrogen-16 in the reactor pool, as well as in the primary system components is limited by the fact that the primary coolant flows from the reactor pool outlet to the delay tank that is described in section 5.2.3. The delay tank serves as a temporary holding area that allows most of the radioactive nitrogen in the water to decay before the water is circulated through the primary pumps, heat exchangers, and back to the pool. The baffle plate at the inlet of the delay tank separates the incoming water from the existing water in the tank. The separation of the water, in addition to the natural delay that the tank volume provides, causes the mean residence time for any given nitrogen atom to be about 90 seconds, thus allowing time for most of the nitrogen to decay prior to exiting the tank. An analysis of dose rates from nitrogen-16 concentrations in the primary coolant system is presented in section 11.1.3.2.

5.7 Auxiliary Systems Using Primary Coolant

There are no auxiliary systems that use primary coolant directly, however the Emergency Core Cooling System is related to the Primary Cooling System because it serves as an independent means of adding water to the primary loop in an emergency.

5.7.1 Emergency Core Cooling System (ECCS)

The Emergency Core Cooling System provides an independent source of water for adding water to the pool in the event of a loss of coolant accident. Water is supplied from the fire sprinkler system supply, through a series of manual valves, up to the top of the pool. Since the water from this system does not go through a clean-up system, it is for emergency use only, and can only be turned on manually.

[redacted]

1

[redacted]

HIGH POWER POOL STORAGE AREA LOW POWER POOL

[redacted]

REACTOR POOL OUTLINE

Figure 5-2

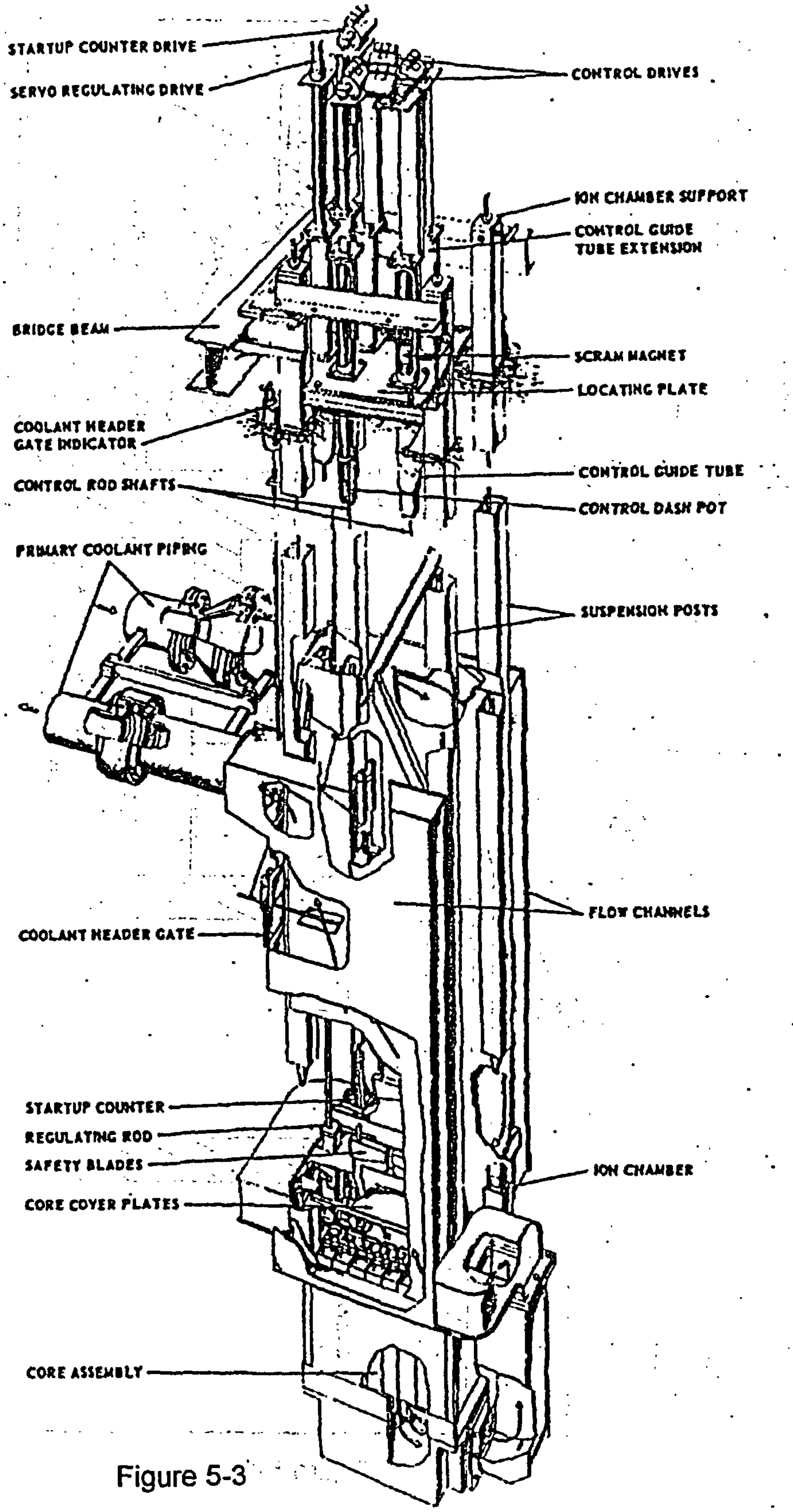
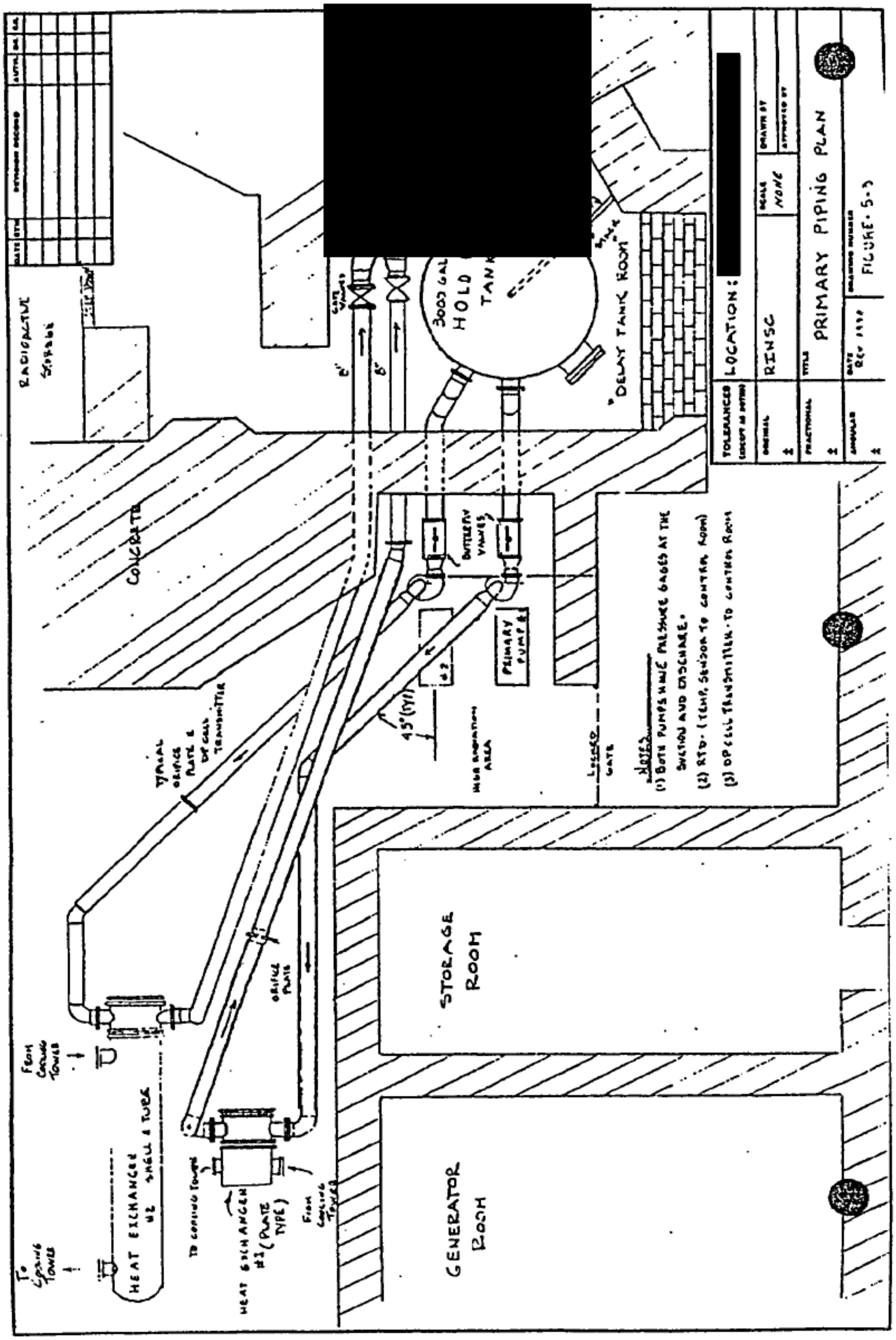
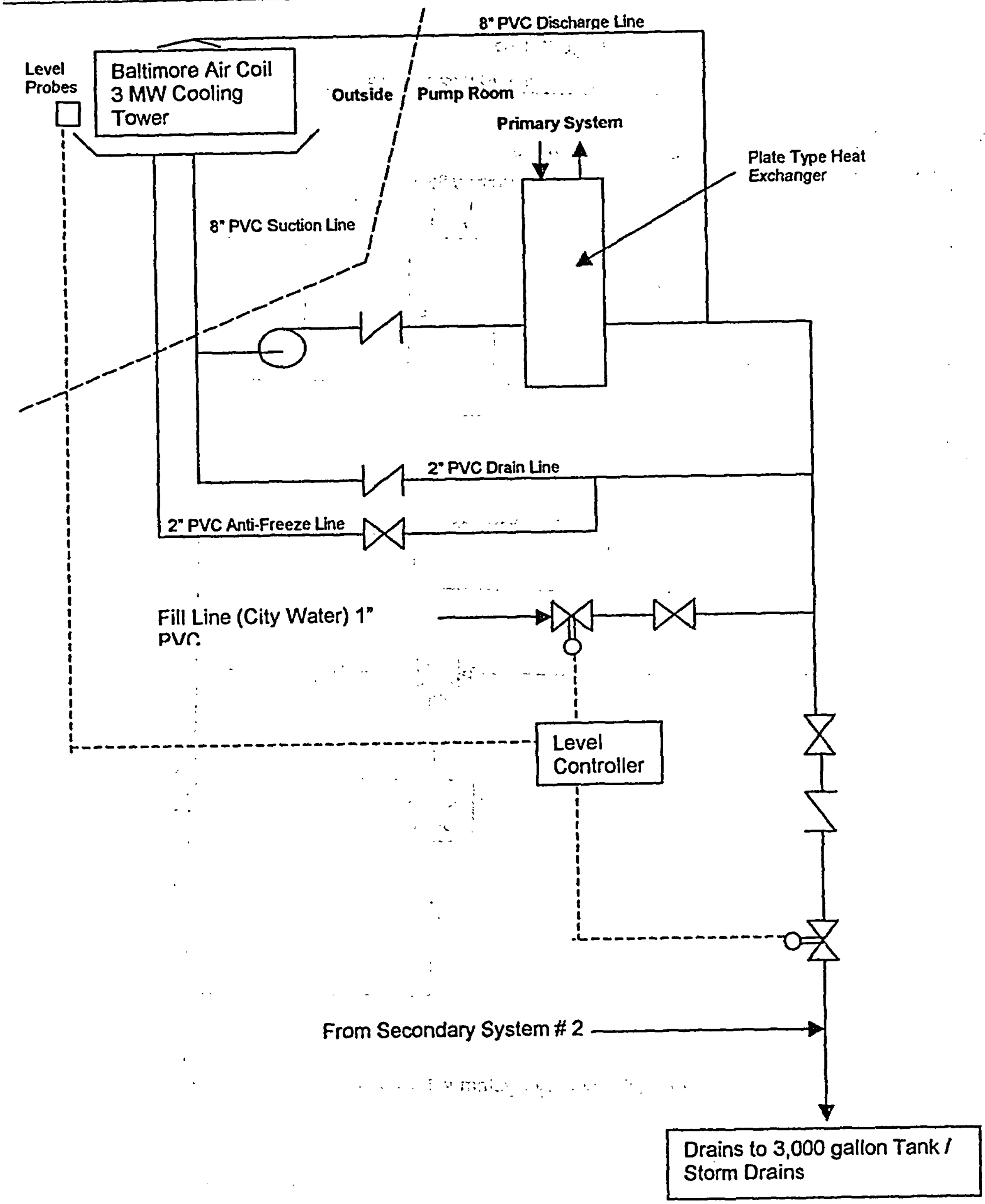

STARTUP COUNTER DRIVE
SERVO REGULATING DRIVE
CONTROL DRIVES
ION CHAMBER SUPPORT
CONTROL GUIDE TUBE EXTENSION
BRIDGE BEAM
SCRAM MAGNET
LOCATING PLATE
COOLANT HEADER GATE INDICATOR
CONTROL GUIDE TUBE
CONTROL ROD SHAFTS
CONTROL DASH POT
PRIMARY COOLANT PIPING
SUSPENSION POSTS
FLOW CHANNELS
COOLANT HEADER GATE
STARTUP COUNTER
REGULATING ROD
SAFETY BLADES
ION CHAMBER
CORE COVER PLATES
CORE ASSEMBLY

Figure 5-3

1

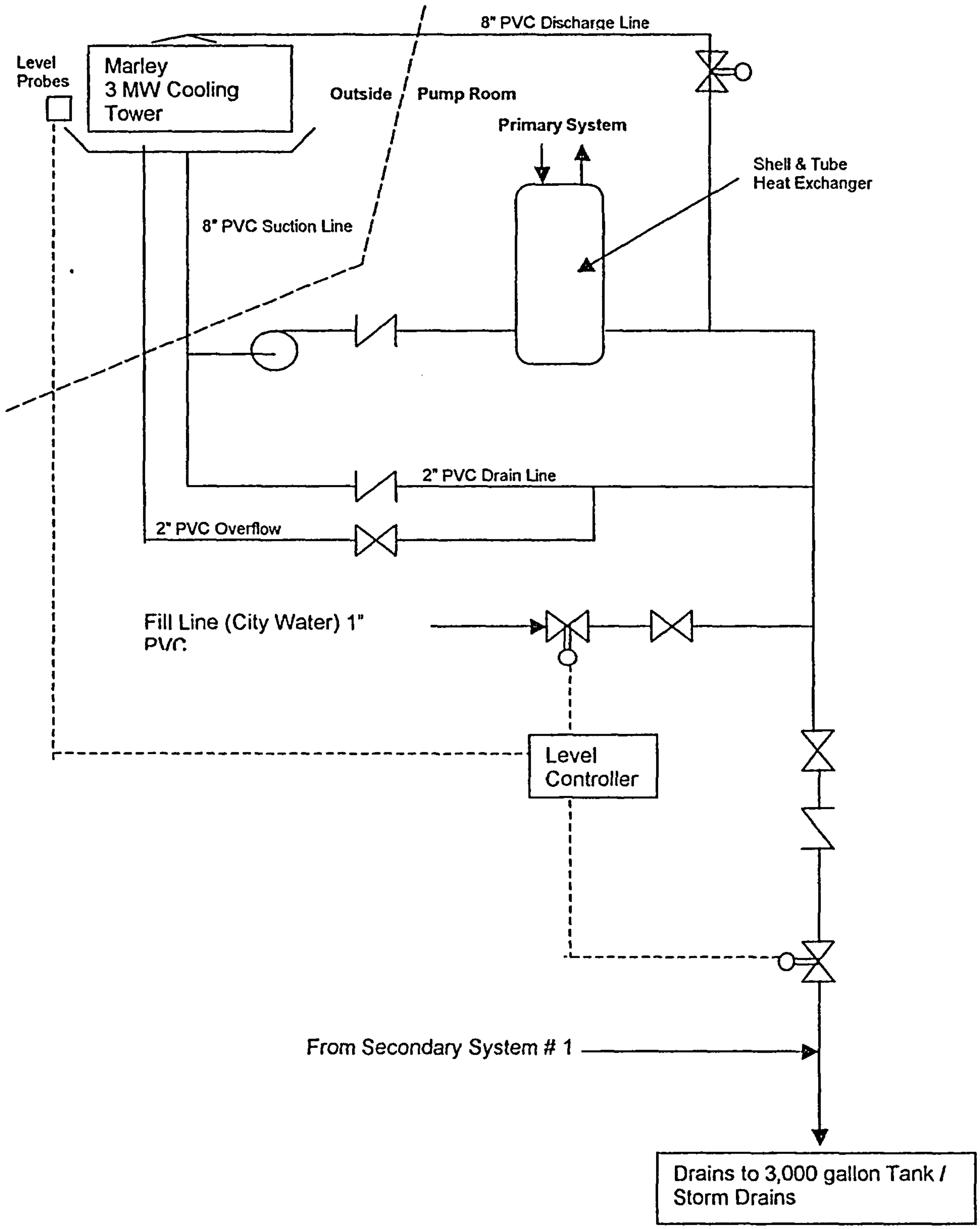

RADIOACTIVE STORAGE
CONCRETE
HEAT EXCHANGER #2 SHELL & TUBE
To Cooling Tower
From Cooling Tower
TO COOLING TOWER
HEAT EXCHANGER #1 (PLATE TYPE)
From Cooling Tower
ORIFICE PLATE
TYPICAL ORIFICE PLATE & DP CELL TRANSMITTER
45°(TYP)
#2
PRIMARY PUMP #1
HIGH RADIATION AREA
BUTTERFLY VALVES
GATE VALVES
8"
8"
3000 GAL
HOLD
TANK
"DELAY TANK ROOM"
LOCKED GATE
GENERATOR ROOM
STORAGE ROOM
NOTES
(1) BOTH PUMPS HAVE PRESSURE GAGES AT THE SUCTION AND DISCHARGE.
(2) RTD - (TEMP. SENSOR TO CONTROL ROOM)
(3) DP CELL TRANSMITTER - TO CONTROL ROOM
TOLERANCES
LOCATION: [redacted]
RINSC
SCALE NONE
DRAWN BY
APPROVED BY
TITLE PRIMARY PIPING PLAN
DATE REV 1990
DRAWING NUMBER FIGURE - 5-3

1 Figure 5-4



Secondary System # 1
Figure 5-5

Figure 5-6



Secondary System # 2

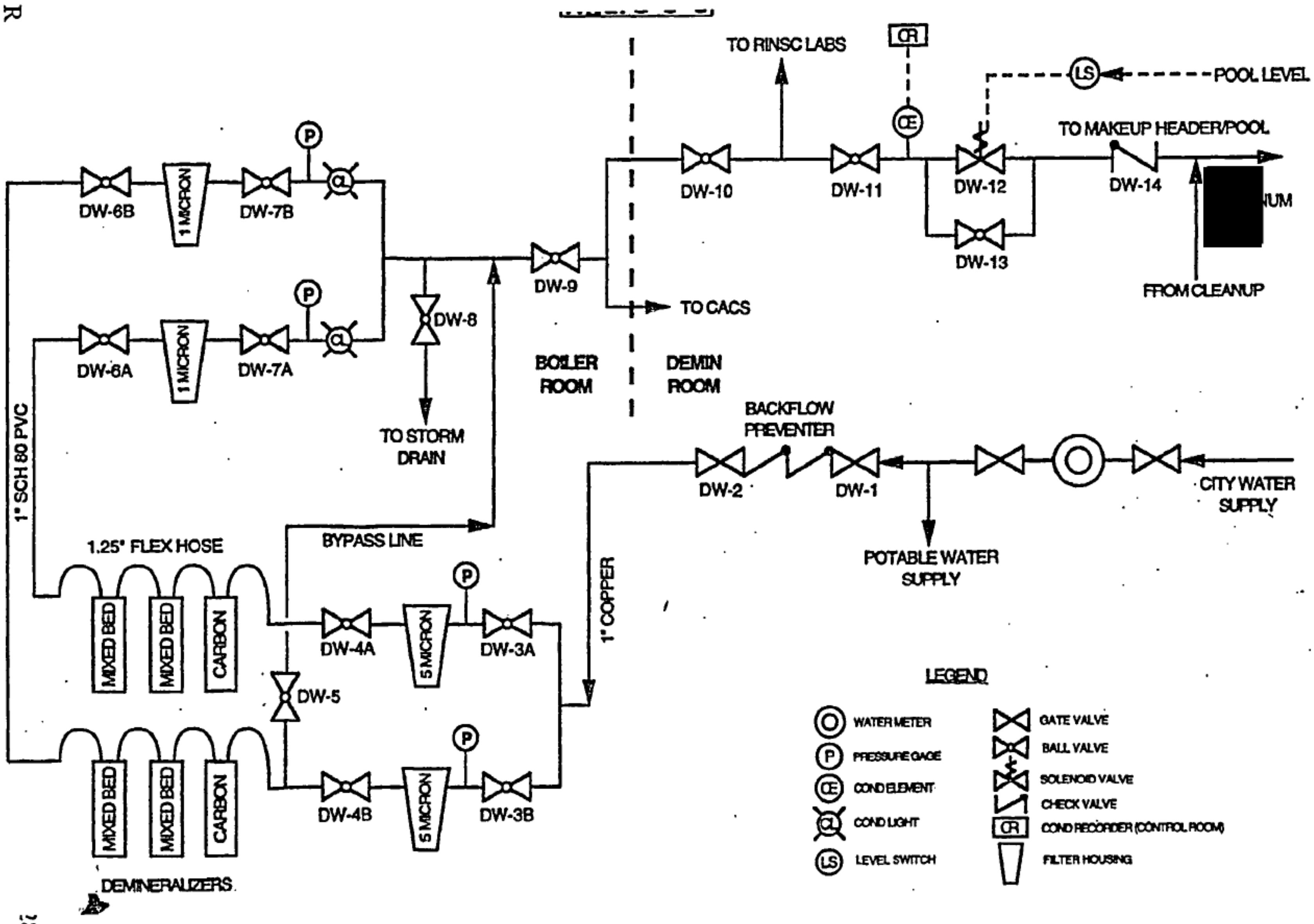

TO RINSC LABS
CR
LS
POOL LEVEL
CE
TO MAKEUP HEADER/POOL
P
DW-6B
1 MICRON
DW-7B
DW-10
DW-11
DW-12
DW-14
[redacted]UM
DW-13
DW-9
TO CACS
FROM CLEANUP
DW-6A
1 MICRON
DW-7A
DW-8
BOILER ROOM
DEMIN ROOM
TO STORM DRAIN
BACKFLOW PREVENTER
DW-2
DW-1
CITY WATER SUPPLY
1" SCH 80 PVC
BYPASS LINE
POTABLE WATER SUPPLY
1.25" FLEX HOSE
MIXED BED
MIXED BED
CARBON
DW-4A
5 MICRON
DW-3A
1" COPPER
DW-5
LEGEND
WATER METER
PRESSURE GAGE
COND ELEMENT
COND LIGHT
LEVEL SWITCH
GATE VALVE
BALL VALVE
SOLENOID VALVE
CHECK VALVE
COND RECORDER (CONTROL ROOM)
FILTER HOUSING
DW-4B
5 MICRON
DW-3B
DEMINERALIZERS

Figure 5-7

CAPTER SIX

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Chapter 6

6.0 ENGINEERED SAFETY FEATURES

This chapter discusses and describes the Engineered Safety Features (ESF) for the reactor facility. The ESF are active or passive features designed to mitigate the consequences of accidents and to keep radiological exposure to the public, the facility staff, and the environment within acceptable limits. The concept for ESF evolved from the defense-in-depth philosophy of multiple layers of design features to prevent or mitigate the release of radioactive materials to the environment during accident conditions. The need for ESF is determined by analyzing accidents that could occur, even though prudent and conservative design of the reactor facility has made the incidence of an accident very unlikely.

6.1 Summary Description

From the original operational license and its history of operation, the basic ESF of the reactor are its containment, confinement and associated emergency electrical power system. The design of these safety features is based on the maximum hypothetical accident (MHA). The function of the containment, confinement and associated emergency electrical power system is to assure that in the event of an accident which could involve the release of radioactive material, that the confinement building air is exhausted through a clean-up system and stack. This is assured by creating a flow of air into the building with a negative differential pressure between the building and the outside atmosphere. The building is required to be gas tight in the sense that a negative differential pressure can be maintained dynamically with all gas leaks occurring inward. The confinement and emergency exhaust system becomes operative when a building evacuation button is pressed. This action shall: (1) turn off all ventilation fans and the air conditioner system in the reactor building and (2) close the dampers on the ventilation intake and exhaust, other than those that are a part of the emergency exhaust system. No further action is required to establish confinement and place the emergency exhaust system in operation. An auxiliary electrical power system is provided at the site to insure the availability of power to operate the emergency exhaust system. The general description of each ESF is discussed in subsequent sections.

6.2 Detailed Descriptions

6.2.1 Confinement

Confinement (figure 6-2) is the isolation of the reactor building and maintaining a negative pressure within. When an unsafe radiological situation develops as defined in the facility operating and emergency procedures, (such as the breach of fuel element containment) the confinement system is activated by depressing any one of five emergency evacuation buttons. The evacuation alarm horns are sounded and personnel respond in accordance with the operating and emergency procedures. In the unlikely event of a release of fission products, or other airborne radioactivity, the confinement isolation system secures the normal ventilation exhaust fan, bypasses the normal ventilation supply up the stack, and closes the normal air inlet and exhaust valves. In confinement mode, the emergency exhaust system maintains a negative building pressure with a combination of controls designed to prevent escape of any large quantity of airborne activity. The emergency exhaust system filters the building air through charcoal and absolute filters and dilutes the discharge air with a supply of air from another source and then exhausting through a 115-foot tall stack.

The confinement building (reactor building) is a concrete building normally maintained under a negative pressure. Normal building air flow operation, shown in figure 6-4, results from reactor room air being removed through the reactor room exhaust blower, through the butterfly valve into the reactor exhaust stack. Air intake comes through a butterfly valve into the building heated air duct. The normal inlet and outlet butterfly valves close air tight on activation of the emergency exhaust system.

Off-gas (figure 6-4) is removed from experiment facilities through an off-gas blower and HEPA filter system. The blower removes gases from the thermal column, beam tube drains and vents etc. and discharges into the reactor room exhaust piping before the reactor room discharge butterfly valve. The pneumatic system blower also discharges its gas removal into the off-gas discharge line. The off-gas and pneumatic systems operation are more fully described in Chapters 9 and 10. The dilution air blower provides additional airflow and discharges into the stack at its base. Normal building differential pressure (dp) is measured at two locations, one in the control room, and one in the adjacent lab building.

6.2.2 Emergency Evacuation System

Figures 6-2 and 6-3 show the makeup of the emergency exhaust system. Upon activation of confinement, the reactor building exhaust system butterfly valve and the intake air butterfly valve close. The dilution air blower comes on or stays on. The off-gas blower and pneumatic system blowers are de-energized. The emergency exhaust blower is energized. The emergency exhaust blower exhausts the reactor room through the emergency filter system. The emergency filter system directs air from the reactor building through a roughing filter, an absolute particulate filter, a charcoal filter for removing radioiodine and an absolute filter for removing charcoal dust that may be contaminated with radioiodine. Each absolute filter cartridge is individually tested and certified by the manufacturer to have an efficiency of not less than 99.97% when tested with 0.3 micron diameter dioctylphthalate smoke. The minimum removal efficiency of the charcoal filters is 99% based on ORNL data and measurements performed locally.

The exhaust air is diluted and then discharged into the stack. The building differential air pressure during confinement is >0.5 inches water, gauge. Filtration efficiency is determined annually. The system is tested weekly in accordance with the facility operating procedures, Section 10.

Figure 6-1 shows the emergency power supply interface with the equipment operating the confinement system and auxiliary equipment (dilution blower).

The facility emergency power supply system provides backup power for the operation of the RINSC confinement and emergency exhaust system on loss of power. (See Section 6.2)

The generator is an ONAN Electric Generating Plant, series RJC 15 KVA, Power Plant (120/208-volt, 60-cycle, 3-phase). The unit is the original unit for the facility that has been maintained in accordance with manufacturer's recommendations throughout its lifetime. The unit provides emergency power to the emergency exhaust system blower, the ECCS electrical components, and other facility components as described in Chapter 8, and as shown in Figure 6-1. The emergency power system is considered a part of the reactor confinement system. System performance and reliability is assured through routine testing and maintenance in accordance with manufacturer recommendations as implemented in the facility operating procedures.

The generator is start-tested in accordance with operating procedures (reference: Operating Procedures, Section 10) and load-tested monthly by initiating the emergency evacuation system during generator test operation. Maintenance is conducted annually. The emergency power plant operates on bottled propane gas. The gas bottles are located outside of the reactor building near the generator room. (Maximum operating time with [redacted] of propane gas is approximately [redacted]). Reactor operating procedures preclude reactor startup if the generator fails to start or is undergoing maintenance. Section 3.3.4 of the "Emergency Plan" describes actions to be taken during a power loss.

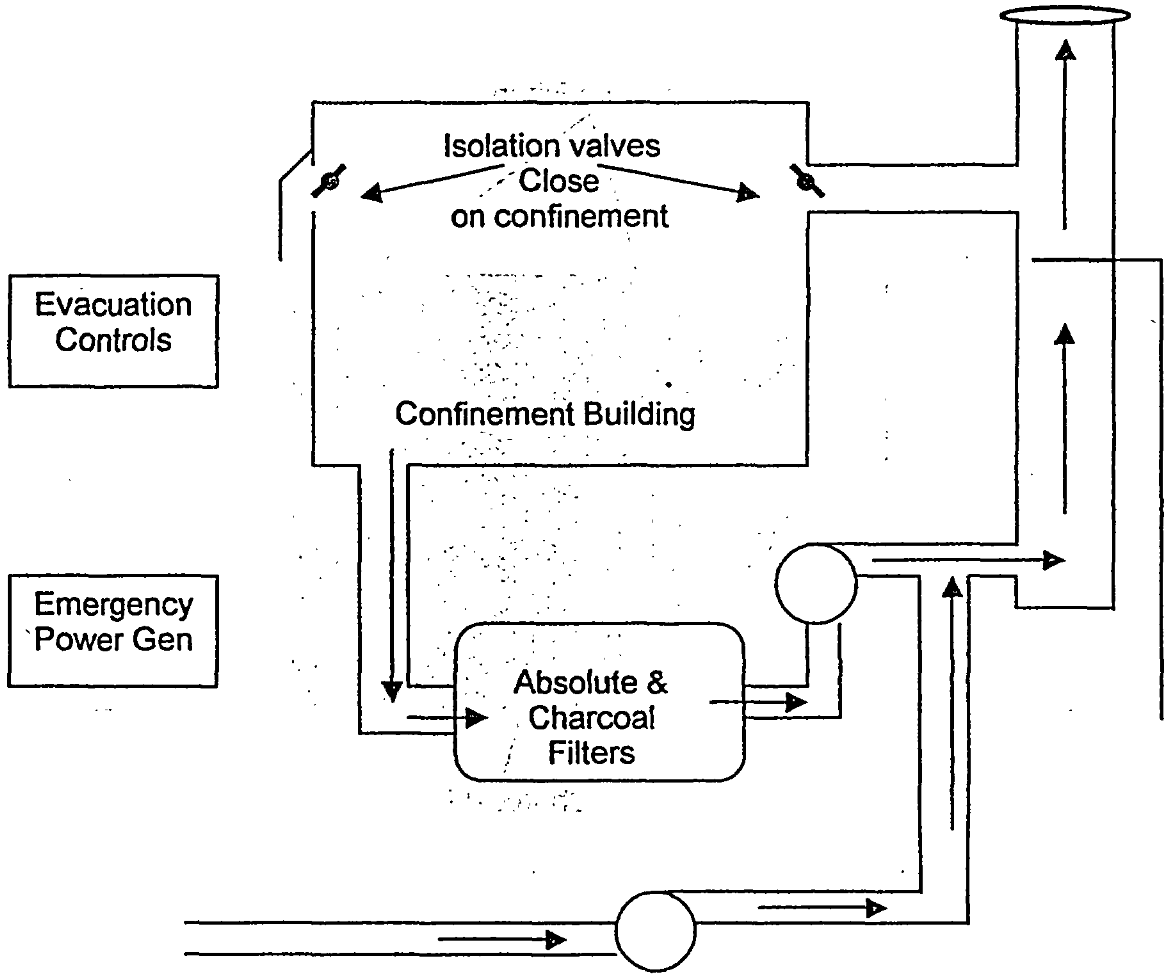
6.2.3 Containment

Containment is also considered in the design of the fuel elements, under normal conditions, the fuel plates will contain and prevent the escape of radioactive products resulting from the fission process.

[redacted]

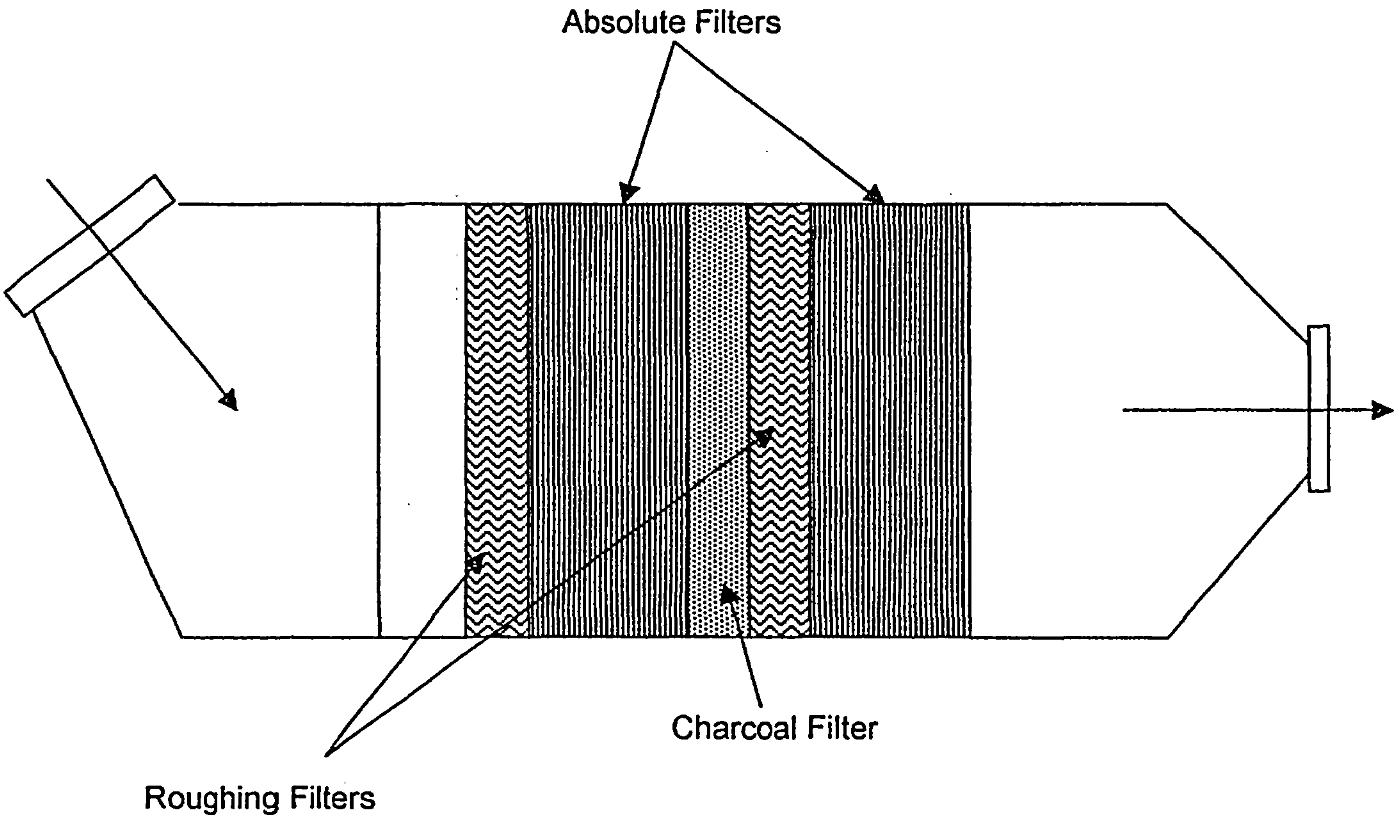
Emergency Power Supply

Figure 6-1



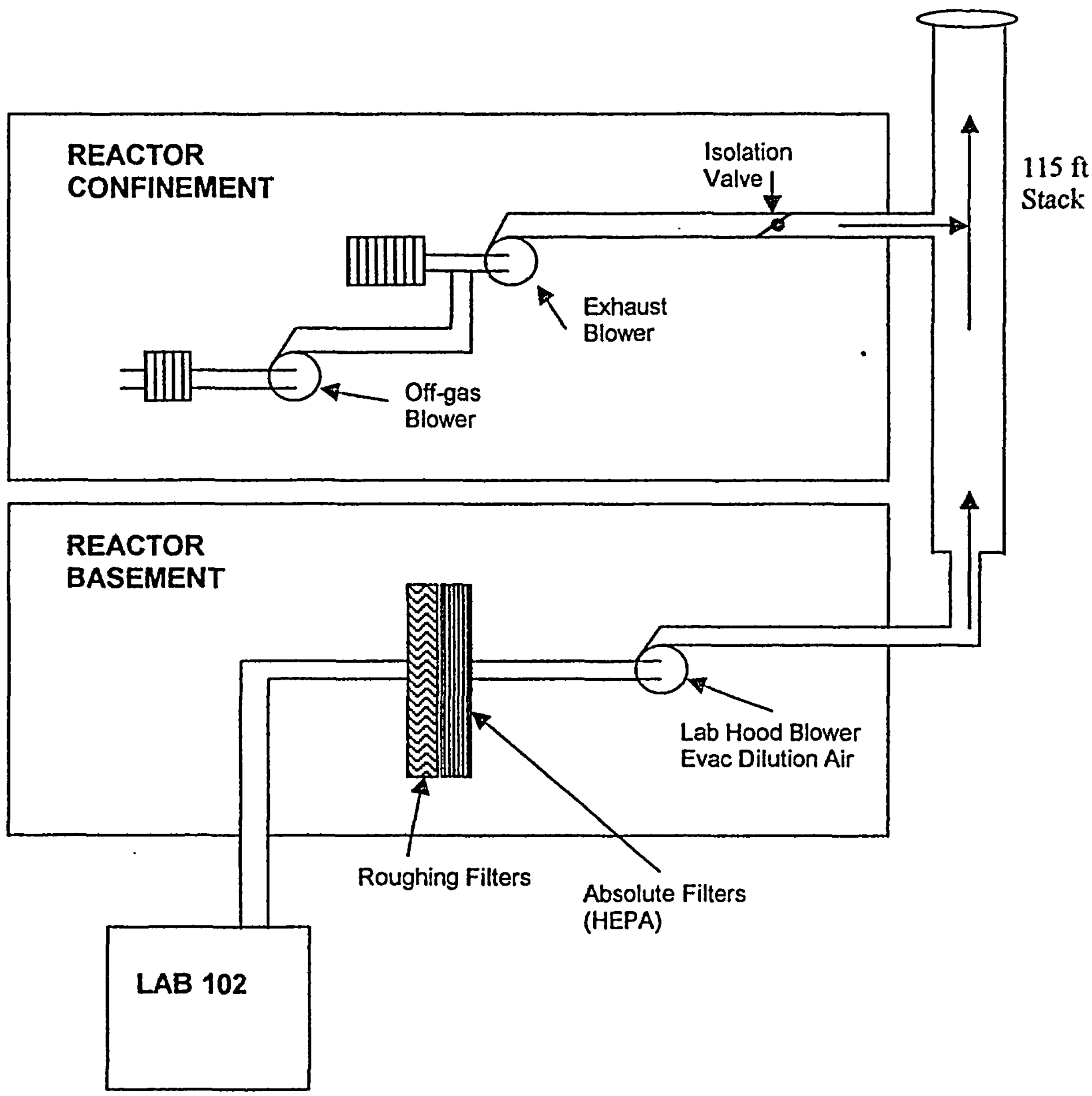
Emergency Confinement System
EVACUATION MODE

Figure 6-2



Emergency Evacuation Filter Box

Figure 6-3



Gas Removal System
NORMAL OPERATION

Figure 6-4

CHAPTER SEVEN

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Chapter 7

7.0 INSTRUMENTATION AND CONTROL

7.1 Introduction

The instrumentation and control systems of the reactor measure and display neutron flux levels in the core and supply proportional signals for interlock, alarm and scram circuits. In addition, control and position indication is provided for the safety control blades and the regulating blade. Instrumentation also monitors, cooling systems, radiation and confinement ventilation systems.

7.1.1 Design Basis

The instrumentation and control system for the RINSC reactor is designed to perform the following functions:

- Provide the operator with information on the status of the reactor and its secondary systems.
- Provide the means for insertion and withdrawal of control blades.
- Provide for manual and automatic control of the reactor power level.
- Provide sensing and control to prevent overpower conditions and automatically inserting control blades to terminate the overpower condition.

The instrumentation and control systems of the reactor include:

a. Nuclear instrumentation, to measure neutron flux at the core and to supply signals for alarm and scram circuits;
b. Area radiation monitors outside the pool;
c. Controls for the safety blades and the regulating blade;
d. Cooling system instrumentation;
d. Alarm and Indicator systems;
e. Process Control systems;

Each area, and the components within that area, is defined in the following subparagraphs. The controls and instrumentation of the reactor are outlined in block form in figure 7-4. The reactor controls and instrumentation are assembled in two cabinets: the reactor control console (Figure 7-1) and the amplifier cabinet (Figure 7-3). The process control cabinet and the motor control center are described in the following paragraphs, Process Control and Instrumentation.

7.2.1 Control Console

The control console serves as a central point for location of operating controls and instrumentation. The operator is provided with a vantage point from which to conveniently observe reactor performance and the pool area. The operator can adjust operating parameters to varying requirements when needed for tests, experiments and power level requirements.

The control console consists of a desk-type cabinet 71 inches wide, 44 inches high, and 31 inches deep. The controls and instruments required for operation of the reactor are contained in a control panel that slopes upward from the rear of the desk surface. Located on the right of the console control panel are the control blade selector switch, manual rundown switch, control blade manual control switch, and the control blade position indicators. Mounted with the position indicators are the power ON indicator, and the blade limit indicators for each control blade.

The central portion of the console is occupied by the annunciator panel. In addition to the alarm/scram indicator lights are the reset switch, the annunciator acknowledge switch, and the scram reset switch. Mounted below the annunciator panel is the manual scram switch.

Below the scram switch, in the center portion of the control panel, is the regulating blade position indicator panel. On this panel are located, below the digital readout, the mode transfer switches and the regulating blade limit indicators. The log count rate and period, wide range log count rate and period, and the linear power level indicators are located on the left side of the control panel. Below these indicators, are mounted the linear range switches, and the reactor ON indicator switch.

7.2.2 Amplifier Cabinet

The amplifier cabinet (Figure 7-3), located to the right of the control console, serves as an assembly point for location of the amplifiers, recorders, and other components used for measurement and control of reactor power. The instruments are mounted in a three-bay, relay-rack type cabinet. The upper panel of the cabinet contains the period, linear power, log count rate, and wide range log recorders. The fuses, power level selector, and master switch occupy the center of the upper panel.

The right lower bay contains the primary scram logic and the Neutron Flux Monitor. Two wide range linear channels occupy the lower panels of the center bay. The trip amplifier, servo power schedule meter and control power circuit breakers are mounted on the left bay of the cabinet. Cooling fans are mounted at the lower end of each bay to provide circulation of cooling air to the electronic components as required.

7.2.3 Power Distribution System

Power for operation of the reactor equipment is supplied through the control-power circuit breaker and the master switch from a 115-volt, 60-cycle unregulated supply, as indicated in (Figure 7–3). The master switch is key-locked in the "off" position to prevent reactor start up. The switch also provides an "on" for normal operation, and a "test" position in which the control drives may be exercised without energizing the scram magnets and withdrawing the blades. Whenever the master switch is turned "on" from "test" or "off", an interlock causes the alarm buzzer to sound for 10 seconds to warn personnel of impending startup. A time-delay mechanism prevents the control drive motors from withdrawing the blades during the 10-second delay period.

7.2.4 Unregulated Control Power Supply

Unregulated 115-volt, 60-cycle control power is supplied to the operating components of the reactor control system through circuit breaker CB2. This unregulated power is

supplied to the cubicle blowers, to the input of the 24 volt DC power supply and to the master control components.
Master switch 4S2 controls the unregulated power to the control blade drives, the power level interlocks, regulating blade drive and position indicators, and to the annunciation, pool level control, and trip actuation circuits.

7.2.5 Neutron Flux Monitor

The Neutron Flux Monitor channel contains the source range, wide range and a linear range. The common input to each range is the same fission counter detector. The source range provides for the minimum count rate to withdraw blades. The wide range provides alarm and period scram functions. The linear range provides for an additional hi-flux alarm and scram.

7.2.6 Control Blade Drive System

The four control blades are controlled by two switches: one selects the blade to be moved; the other, a switch with a spring return, has positions "OUT", "OFF", and "IN", and actuates the selected control blade. Only one blade may be raised at a time; digital readouts indicate the position (inches of travel) of each safety blade.
Indicator lights on the reactor control console show when a control blade has reached either limit of its travel. The control blade drive can be overridden by an automatic or manual scram. Control blade drive motors are interlocked against withdrawal of blades during a 10-second delay period subsequent to reactor startup, and when the log count rate meter is below the minimum number of counts per second.
A position indicator is provided for each control blade drive, to show drive position relative to the fully inserted position. Digital indication to the resolution of 0.01 inch is furnished by number wheels on a mechanical counter that is chain-driven from the ball-bearing screw of the drive. The indication is transmitted electrically through a segmented commutator in the counter to a "withdrawal in inches" indicator on the control console.
All control blades may be inserted simultaneously by using the manual rundown switch ("MANUAL RUNDOWN") located on the control console.

7.2.7 Servo-Controlled Regulating Blade Drive System

The regulating blade system provides automatic control of reactor power level by a servo system that responds to changes of neutron flux level in the reactor. The automatic system is composed of the following:

- Gamma-compensated ionization chamber and wide range linear channel;
- Servo amplifier which responds to the wide range channel signal and controls the speed and direction of the regulating blade motion through a servo motor, and;
- Power schedule selector switch that sets the reference power level.

The regulating blade motion is powered by a stepper motor control. The motor is provided with an interlock against initiation of automatic control if the reactor period is less than 30 seconds and the servo blade is not fully withdrawn. The method of position indication is by linear potentiometer and digital readout with a resolution of 0.01 inches.

7.2.8 Automatic Power Level Channel

The purpose of the automatic power level channel is to hold the reactor at a pre-selected flux level. The units forming the channel include the power schedule set switch, the power schedule indicator ("% FULL POWER"), and a servo controller. One of the wide range linear channels provides input to a closed-loop servo system that monitors the flux level; when the mode transfer circuit is set for automatic operation, the servo system operates the servo control drive attached to the regulating blade.
The desired power level is selected by means of the power schedule set switch and is indicated on the power level indicator. This establishes a reference signal for the servo amplifier, which is compared with the actual power level signal from the wide range linear channel. The difference between the two signals generates an error signal. The error signal drives the regulating blade drive motor in the proper direction to adjust the reactor power level to the reference value.
The automatic power level channel may be activated at any time provided the reactor period is greater than 30 seconds and the regulating blade is fully withdrawn. Once placed in automatic control, the system remains in automatic until the mode transfer ("MAN") button is pressed or a scram trip is initiated.

7.2.9 Wide Range Channel

The Wide Range Log Channel is used to provide period safety (scram) action as well as neutron monitoring over a wide power range. The functions are:

1. Indicate percent power.
2. Indicate power rate of change (period).
3. Provide period scram.

The intermediate range log power functions are provided by a Wide Range Channel of the Neutron Flux Monitor. The channel provides the wide-range log power function and the period scram trip signal. Reactor power level is covered from source level to 150% power. Using the AC signal from the a fission chamber, the channel combines a pulse log count rate technique for the lower five decades with a log Campbell technique for the upper five decades.
The circuitry is all solid-state modular construction; the reliability is high and the response time is adequate for any power reactor transient. The system can be calibrated and checked (including the testing of trip levels) prior to operation. Test and calibration circuits are provided which feed appropriate signals into the input of the preamplifier for checking the electronics (including the preamplifier), and a positive test for chamber and cable integrity is also provided.

7.2.10 Flux Level Safety Channels

The units that form the safety channels are: two compensated ion chambers mounted in corner posts of the suspension frame, two wide range linear channels (including the two power level indicators and range selector switches mounted on the control console), a trip amplifier, a safety selector switch, and a linear power recorder. The channels monitor power level over the total flux range produced by the core. They therefore overlap the source range and the wide range.
The safety channel signals are combined with signals from the wide range and period amplifier in a trip amplifier. The trip amplifier scrams the reactor on high flux or short period.

7.2.11 Scram Circuits

The scram circuits initiate either relay and/or electronic scram. Each circuit is discussed in a separate subparagraph below.

7.2.12 Relay Scram Circuit

The relay scram circuit controls input power to the trip amplifier. Interruption of this power de-energizes the scram magnet/amplifier and automatically shuts down the reactor. The manner in which the relay scram circuit is actuated under certain conditions is summarized below.

- <u>Manual</u> - Manual scram is initiated at the operator's discretion by actuation of the manual scram push-button that opens the relay scram circuit and de-energizes the scram relays.
- <u>Period</u> - Relay scram occurs when the period alarm relay in the wide range channel period amplifier is de-energized by a reactor period less than the preset value.
- <u>Bridge, Gate, or Coolant</u> - Relay scram occurs if the bridge is moved out of position, if the coolant header gates are opened, or if the primary coolant temperature or flow is abnormal.
- <u>Neutron Flux</u> - Relay scram occurs when the high flux relays in either wide range linear are de-energized by high flux signals in either safety channel.
- <u>Seismic Disturbance</u> - Relay scram occurs when a seismic disturbance closes the seismic trip detector contact that short-circuits the seismic trip relay coil.

7.2.13 Electronic Scram

Electronic Scram is caused by a flux level exceeding a preset value on any range in either of the wide range linear safely channels, or by exceeding a reactor period in the wide range channel. Electronic scram is initiated by electronically (bi-stable transistor-controlled amplifier) cutting off the current in each scram magnet.

7.2.14 Alarm and Indicator System

The alarm system is divided into two sections: one for the coolant variables and the other for the nuclear variables. The section used for cooling system alarms will be operative only when the power schedule switch is set to positions above 0.1 MW. When an abnormal condition develops, a buzzer sounds and the appropriate indicator light energizes to full brightness. The operator may press the acknowledgment button to silence the buzzer. When the alarm condition is corrected, and the reset button is pressed the light goes dim. The following conditions will actuate the alarm system:

- Any scram;
- neutron flux exceeding the preset value;
- reactor period less than seven seconds;
- safety blade disengaged from magnet;
- water level in pool 2 inches below normal;
- neutron channel amplifier failure;
- regulating blade at either limit of travel, when on automatic control;
- high area radiation;
- seismic disturbance;
- cooling system trouble, including high conductivity, high temperature, low flow in primary or secondary system, bridge or gate out of position.

To provide operating information for the reactor operator, the following indicating lights are provided:

- Scram;
- scram reset;
- regulating blade at either end of travel;
- regulating blade automatic control;
- safety blade magnet engaged;
- console power on;
- safety blade "IN" (distinct light for each); and
- safety blade "OUT" (distinct light for each).

7.2.15 Radiation Protection Instrumentation

The area radiation monitor system includes a remote area monitor indicating gamma radiation levels at four representative points around the reactor pool and cooling systems. An additional area monitor indicates neutron flux at the thermal column door area. The individual meters, located in the control room amplifier cabinet, indicate the gamma or neutron radiation level from the corresponding sensing element. Some channels have remote indicators and alarms. The preset level at which an alarm occurs is established from guidelines.

7.2.16 Process Control and Instrumentation

Control of the cooling system and the monitoring of its operating conditions are accomplished through the process control and instrumentation system. Except for sensing elements, instrumentation for the system is concentrated in the process control cabinet (Figure 7-2). Motor control unit relays and circuit breakers are on the panel of the motor control center, with remote start-stop buttons mounted on the process control cabinet. The process control cabinet is installed beside the amplifier cabinet and is accessible to the operator in the control room.

7.2.17 Process Control Cabinet

The process control cabinet is a sheet-metal cabinet with front panel and rack-type mounting provided for process and cooling system instrumentation. The cabinet serves as an assembly point for location of flow and temperature indicators, beamport control, pneumatic system control, motor controls and primary water conductivity indication.

7.2.18 Motor Control Center

Each of the motors is controlled by a set of remote "START" and "STOP" indicating push buttons on the process control panel located in the control room.

7.2.19 Coolant Monitoring Channels

To monitor the conditions of flow, temperature, and conductivity of the coolant system, the process system is provided with sensing, transmitting, and display instruments for each channel. The modules provide alarm and scram contacts.

7.2.20 Conductivity Channel

Coolant conductivity is measured at three points; one in the delay tank, one in the pool cleanup system, and one in the pool makeup system. The primary coolant is examined for dissolved solids by measuring the conductivity of the water at the clean up system de-mineralizer input (T-1 port). If the conductivity increases to a preset limit, the reactor is manually shutdown.

7.2.21 Temperature Channel

The temperature-sensing, resistance temperature detectors (RTD)'s operates on the principle that resistance in a wire varies in relation to temperature. The output of the RTD is connected to a process monitor/control instrument. The output is converted from resistance, digitized and displayed in degrees Fahrenheit (°F). Relay outputs interface with the alarm and scram circuits. The scram relays are setup in the fail-safe mode of operation, the relay trips when:

- The display value reaches or exceeds the set-point;
- The displayed value exceeds the high/low range of the meter.
- The meter cannot perform an A/D conversion;
- The power to the unit is off.

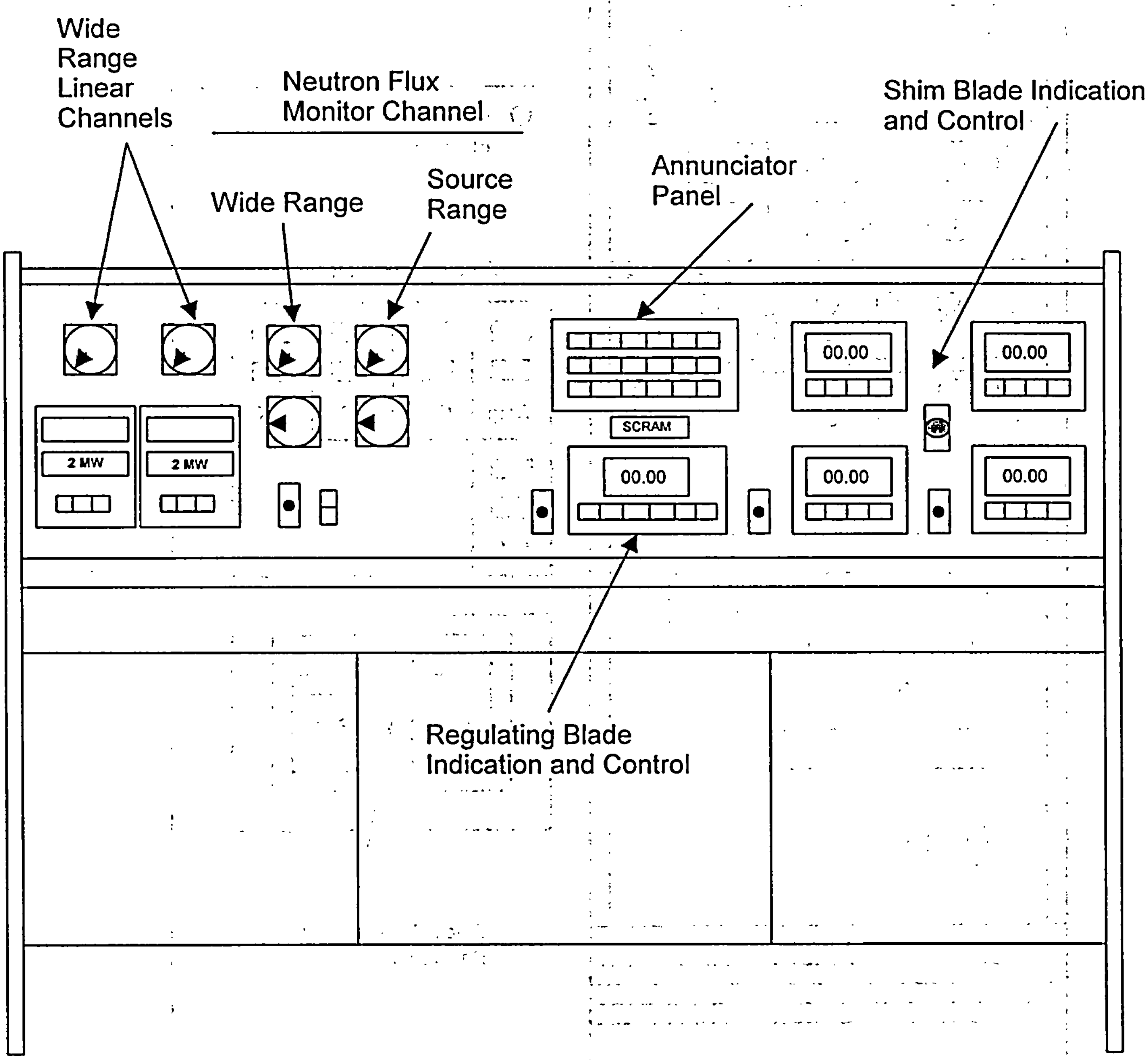
7.2.22 Flow Channel

A stainless-steel calibrated knife edged measuring orifice is installed in each primary coolant loop. The upstream pressure is directed to the high-pressure side of a differential pressure transmitter. The downstream pressure is directed to the low-pressure side of the transmitter. Any change in the differential pressure across the orifice produces a proportional output signal from the d/p cell. This output is connected to a signal conditioner then to a process monitor/control instrument. The signal is digitized and displayed in inches of water. Relay outputs interface with the alarm and scram circuits. The scram relays are setup in the fail-safe mode of operation, the relay trips when:

- The display value reaches or exceeds the set-point;
- The displayed value exceeds the high/low range of the meter.
- The meter cannot perform an A/D conversion;
- The power to the unit is off.

7.2.23 Pool Water Level Indicator

Equipment is furnished for monitoring the level of the reactor pool water. This includes float-actuated switches, immersed in the pool. For automatic filling of the pool, a decrease of one inch from the normal water level will energize a normally closed solenoid valve in the make up system to start filling the pool from the makeup water de-mineralizer. Two floats are arranged to ensure the valve de-energizes and closes when the pool is returned to normal level. A pool water level 2 inches below normal will cause a scram condition. There is no remote pool level indicator.



REACTOR CONTROL CONSOLE

Figure 7-1

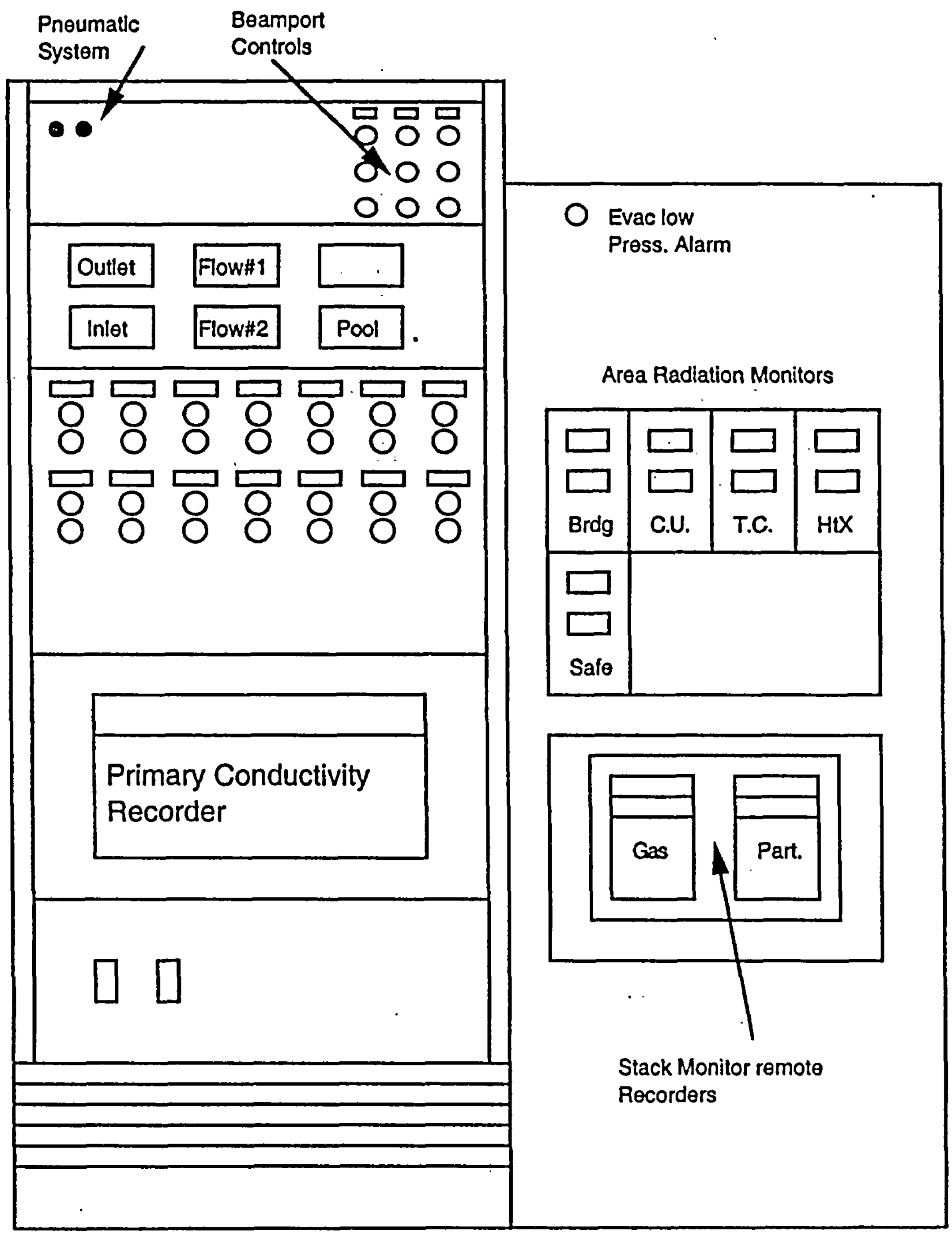

Pneumatic System
Beamport Controls
Outlet
Flow#1
Inlet
Flow#2
Pool
Evac low Press. Alarm
Area Radiation Monitors
Brdg
C.U.
T.C.
HtX
Safe
Primary Conductivity Recorder
Gas
Part.
Stack Monitor remote Recorders

Figure 7-2

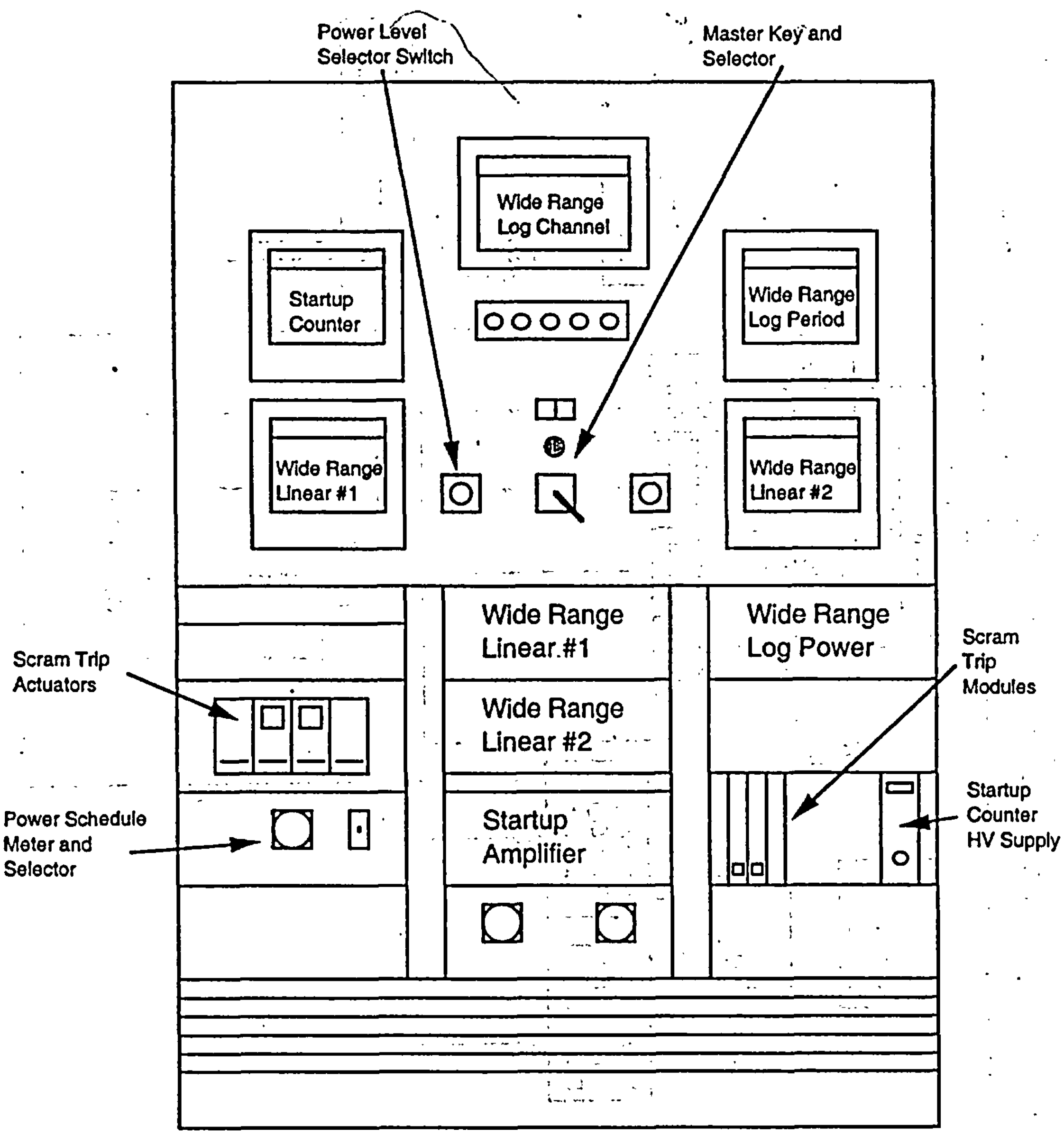

Power Level Selector Switch
Master Key and Selector
Wide Range Log Channel
Startup Counter
Wide Range Log Period
Wide Range Linear #1
Wide Range Linear #2
Wide Range Linear #1
Wide Range Log Power
Scram Trip Actuators
Scram Trip Modules
Wide Range Linear #2
Startup Counter HV Supply
Power Schedule Meter and Selector
Startup Amplifier

Figure 7-3

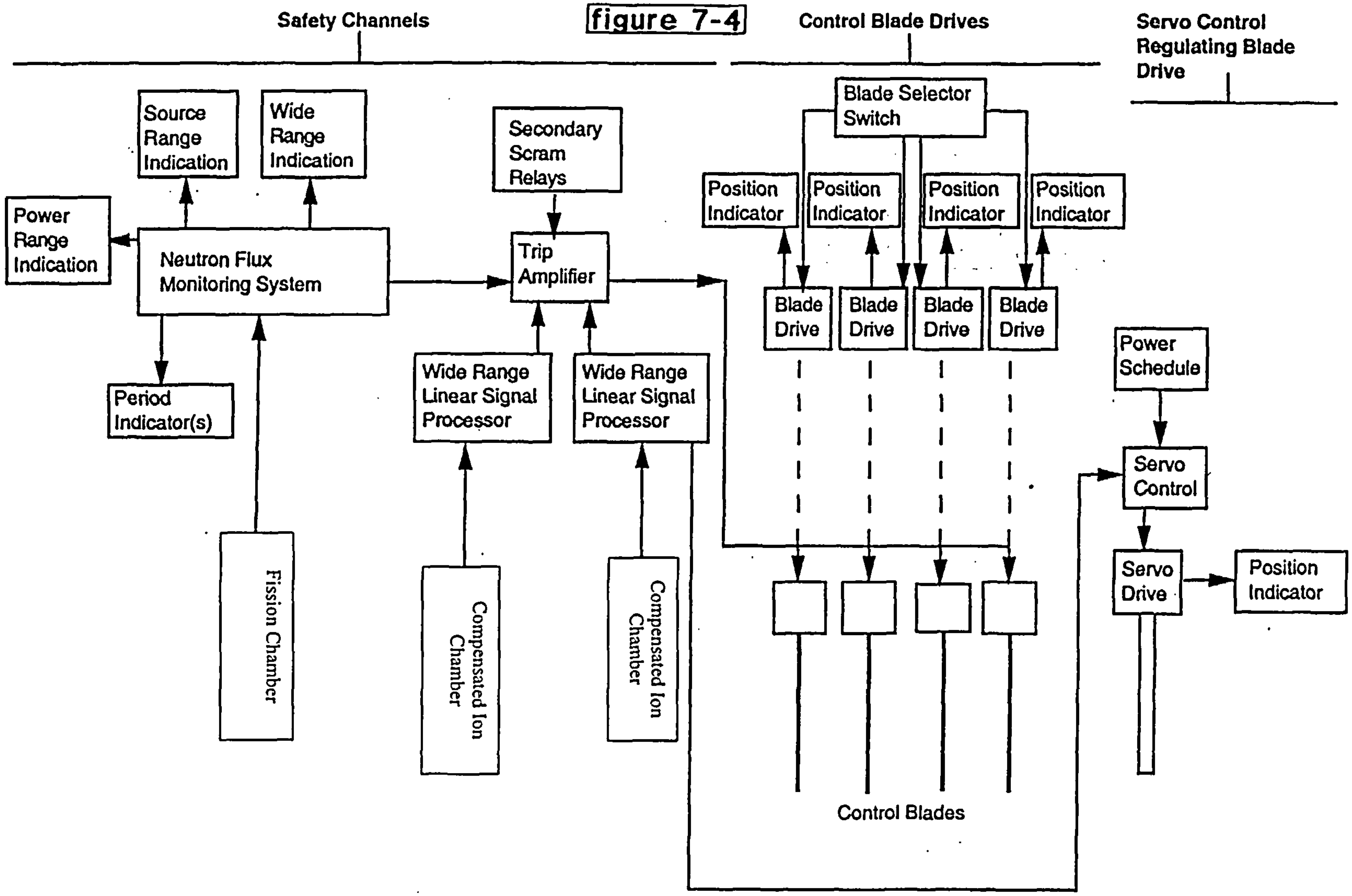


figure 7-4
Safety Channels
Control Blade Drives
Servo Control Regulating Blade Drive
Source Range Indication
Wide Range Indication
Power Range Indication
Neutron Flux Monitoring System
Period Indicator(s)
Fission Chamber
Secondary Scram Relays
Trip Amplifier
Wide Range Linear Signal Processor
Wide Range Linear Signal Processor
Compensated Ion Chamber
Compensated Ion Chamber
Blade Selector Switch
Position Indicator
Position Indicator
Position Indicator
Position Indicator
Blade Drive
Blade Drive
Blade Drive
Blade Drive
Control Blades
Power Schedule
Servo Control
Servo Drive
Position Indicator

Figure 7-4

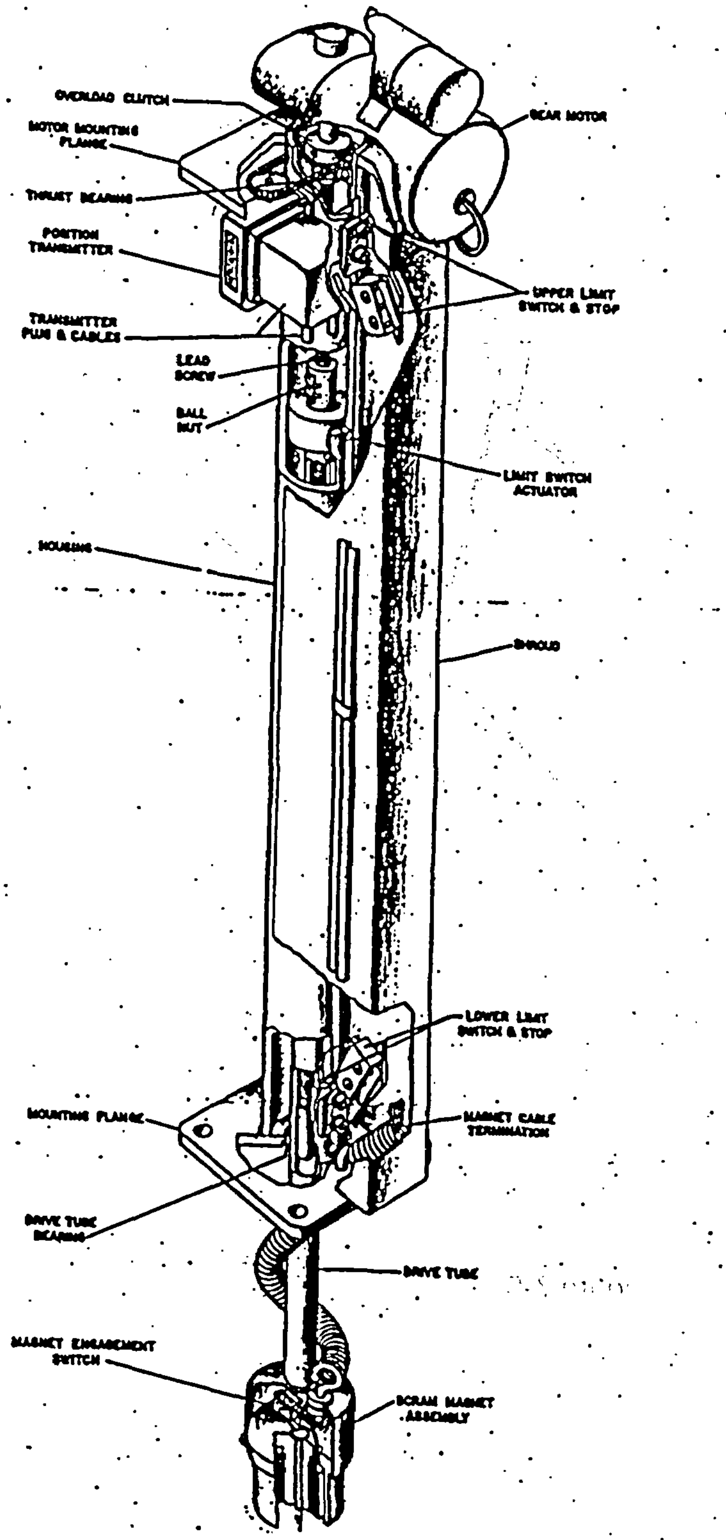

OVERLOAD CLUTCH
MOTOR MOUNTING FLANGE
THRUST BEARING
POSITION TRANSMITTER
TRANSMITTER PLUG & CABLES
LEAD SCREW
BALL NUT
GEAR MOTOR
UPPER LIMIT SWITCH & STOP
LIMIT SWITCH ACTUATOR
HOUSING
SHROUD
LOWER LIMIT SWITCH & STOP
MOUNTING FLANGE
MAGNET CABLE TERMINATION
DRIVE TUBE BEARING
DRIVE TUBE
MAGNET ENGAGEMENT SWITCH
SCRAM MAGNET ASSEMBLY

Figure 7-5

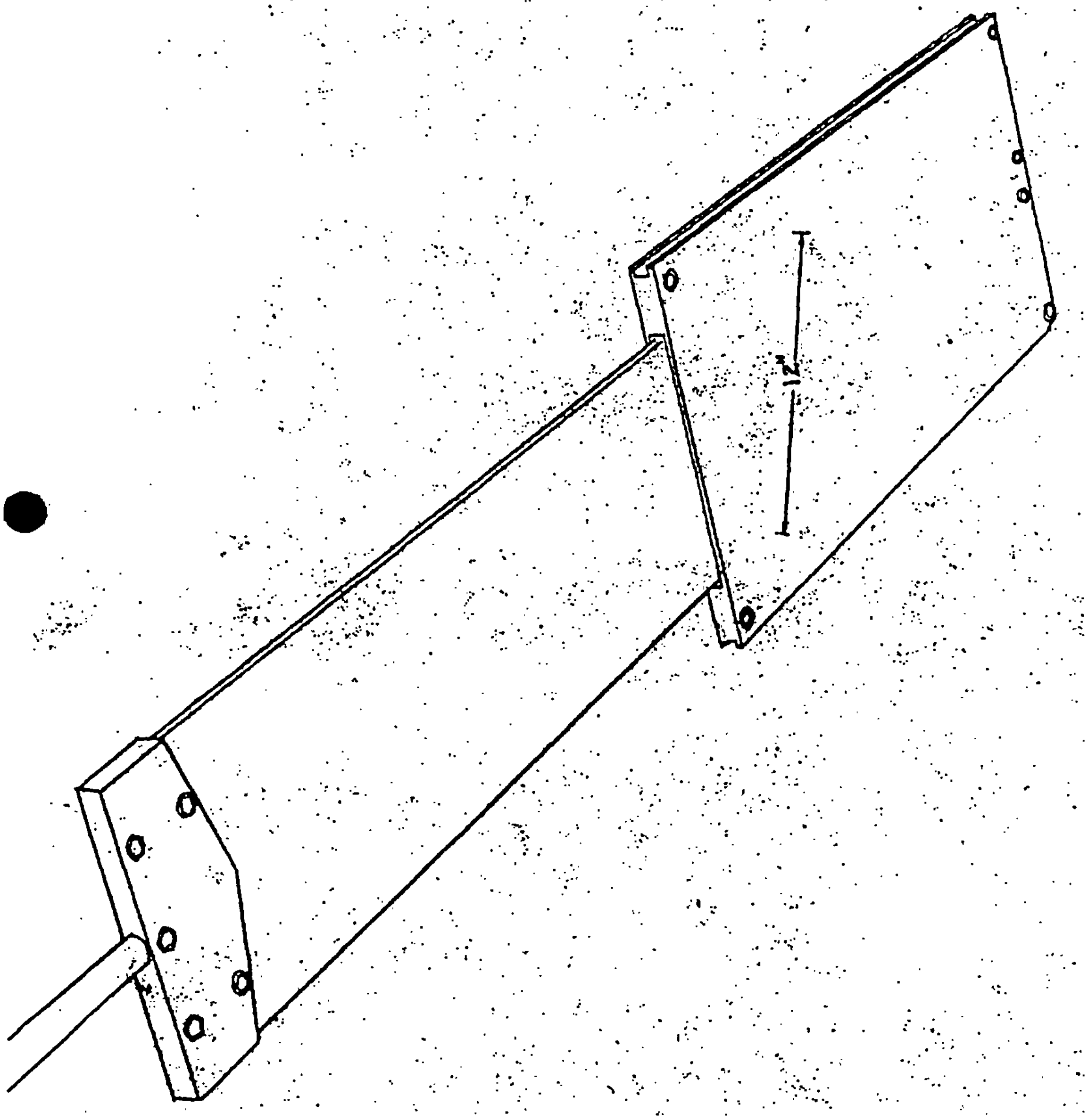


Figure 7-6

CHAPTER EIGHT

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Chapter 8

8.0 ELECTRICAL POWER SYSTEMS

This chapter discusses and describes the electrical power systems designed to support reactor operation at the facility. The information in this chapter is provided under two categories: normal and emergency electrical power systems.

8.1 Introduction

The normal electrical power for the RINSC facility is designed to supply continuous electrical power to all reactor components while the reactor is in operation. The interconnections between the main transformer and the RINSC are designed in accordance with the following codes and standards:

- National Electrical Code – NFPA-70;
- National Electrical Safety Code; and
- NEMA Standards.

The design of the RINSC reactor does not require electrical power to safely shut down the reactor, nor does it require electrical power to maintain acceptable shutdown conditions. However, the RINSC facility is equipped with an emergency generator that will supply vital equipment during abnormal losses of power (refer to Section 8.3 for details).

8.2 Normal Electrical Power Systems

Power is supplied from the local utility via a transformer located onsite. The transformer then feeds the RINSC facility main disconnect switch (600-VAC, 800-AMP). The main disconnect and associated wiring was upgraded in June 1998.

Figure 8.1 illustrates the RINSC electrical distribution system. Electrical power enters the RINSC via the main disconnect and is routed to various 480-VAC motor control centers and two individual step-down transformers. The transformers convert the 480 VAC to 220/110 VAC that is then supplied to various breakers and motor control centers. 480 VAC is supplied to the larger reactor loads such as the primary and secondary pumps. 220 VAC is supplied to the smaller loads, for example, the emergency exhaust blower and the reactor water cleanup pump. Finally, 110 VAC is supplied to the reactor control system and to area lighting, etc.

The reactor control system receives 110-VAC, 60-cycle unregulated power to a SOLA transformer which, in turn, supplies the console, instrument racks, blade drives, and the supply side of the 24-VDC power system. The 24-VDC system supplies the control system circuitry that includes scrams, alarms, and interlocks.

The electric and wiring cables from the control room to the various instruments and control blade drive mechanisms in and around the reactor pool are routed underneath the control room floor in cable trays that travel around the concrete reactor shield and up to a standpipe arrangement. The standpipe penetrates the walkway around the reactor pool and allows movement of the bridge/core throughout the length of the pool without wiring interference.

8.3 Emergency Electrical Power Systems

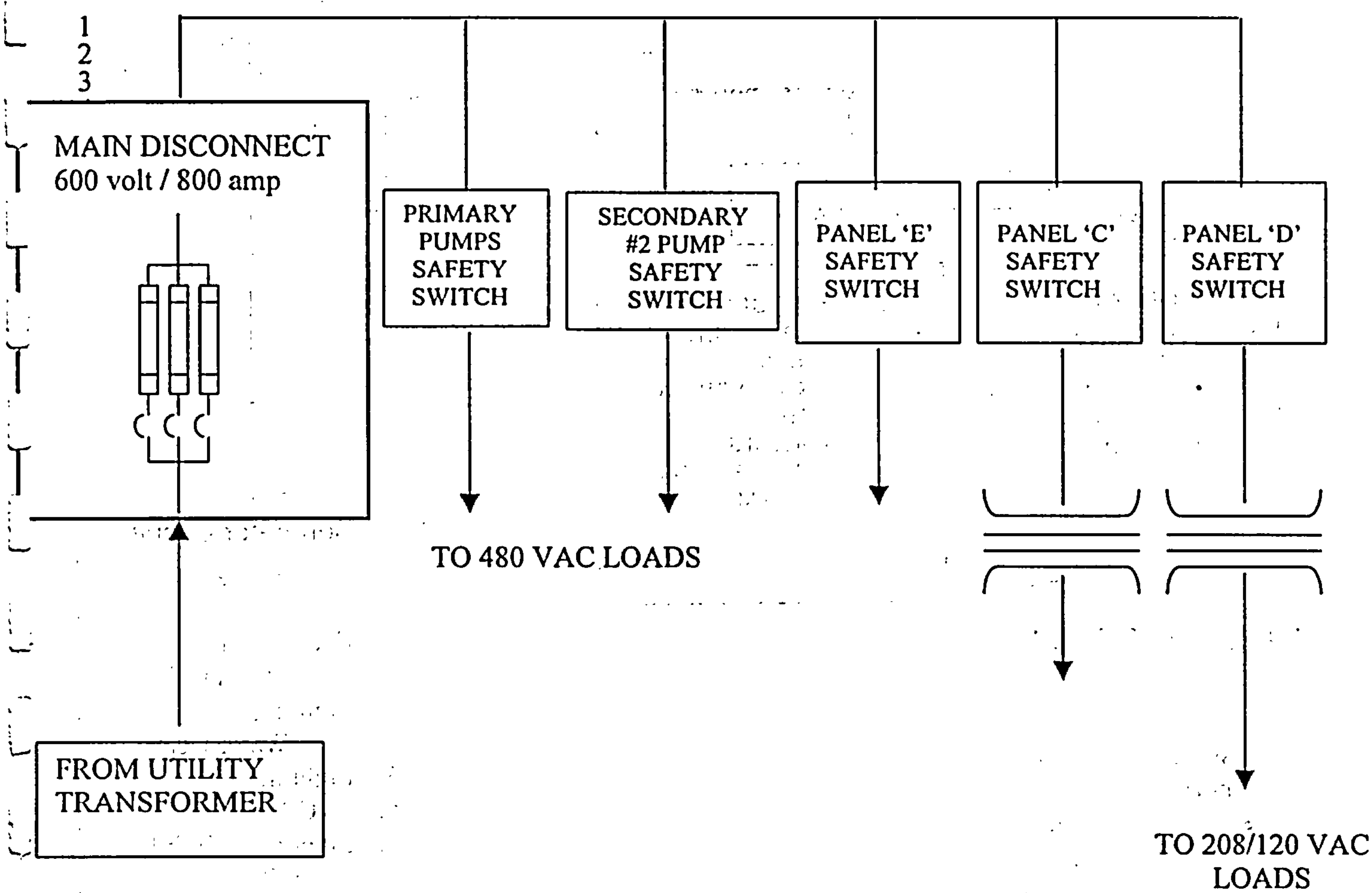
The safe shutdown of the reactor does not rely on continuous electrical power. If normal power is lost, the reactor is instantly shut down. The reactor control blade mechanisms (electro-magnets) are designed to release upon a loss of electrical power and subsequently insert all control blades instantly, thus making the reactor subcritical. Power would also be lost to the forced cooling systems (no flow), however, any residual heat from the core is dissipated into the pool water and eventually into the reactor room air space.

A lengthy interruption of electrical power does not pose any significant problems to the safety of the reactor. The reactor is supplied with an emergency generator that will automatically start (10-second time delay) after a loss of normal power. The main purpose of the generator is to supply power to the emergency exhaust blower. The exhaust blower draws a negative pressure on the reactor building, through charcoal and absolute filters, and discharges to the 115-foot tall stack. Routine and uncontrolled releases of radioactive material are mitigated by the exhaust blower and associated equipment. Monitoring of any release is via the stack monitor that can be powered by the emergency generator (requires operator action to manually shift the stack monitor from normal to emergency power).

Figure 8.2 illustrates the RINSC emergency power supply. The primary design and use of the emergency power source (15-kW generator) is to supply the emergency exhaust system/evacuation system in the unlikely event of a release of fission products or other airborne radioactivity. Additionally, the emergency generator supplies a dilution blower (Chem Lab blower), emergency lighting, communication equipment, and various electrical outlets that may be utilized for effluent sample retrieval and monitoring. The generator is capable of supplying power to the designated loads indefinitely if fuel (propane) is available. The RINSC facility maintains sufficient fuel inventory onsite for the performance of mitigation strategies specified in this SAR.

The emergency generator will automatically start and assume the loads, as necessary, upon a loss of normal electrical power. The automatic bus transfer switch will transfer the load to the generator and back again as normal electric power is restored. The generator, however, must be manually stopped.

The ability of the emergency power generator to assume the emergency load is tested and normal operability checks are performed. The operability checks indicate that the emergency power generator is available and provide reasonable assurance that the emergency power generator will remain available. A periodic load test provides reasonable assurance that the emergency power generator's electrical control and distribution system remains operable. Periodic maintenance is performed according to the manufacturer's recommendations.



RINSC ELECTRICAL DISTRIBUTION SYSTEM

Figure 8-1

[redacted]

EMERGENCY POWER SUPPLY

Figure 8-2

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9.1 Heating and Ventilation Systems

9.1.1 General Description

The heating system, described as a shared system with the URI/CACS building in Section 1.4, is a forced hot water circulating system. The reactor confinement building has unit heaters with thermostatically controlled fans to circulate warm air. The reactor control room has both a forced hot water unit and an electric heating unit. The reactor building air intake system, which has a butterfly damper as described in the confinement system, includes a hot water heating coil. The incoming air is warmed across the hot water coil and is thermostatically controlled with a by-pass air damper. There is no central air conditioning or humidity control for the reactor confinement building. The control room has a window air conditioner that does not penetrate the confinement building wall. It maintains temperature and humidity for proper operation of the control room electronic components.

The reactor building ventilation system consists of exhaust blowers which discharge into the exhaust stack. The stack also receives dilution air from a blower in the basement area outside of the reactor confinement building. Air from the pneumatic blower system is discharged into the stack via the off-gas suction line. (See Section 10.4) The off-gas blower removes gases from the thermal column, beam tubes and the pneumatic system and discharges into the suction line of the reactor room exhaust blower. In addition, the reactor room exhaust blower constantly exchanges the air from the reactor confinement building. The exhaust blower inlet plenum is located near the pool platform to essentially sweep air across the pool surface which helps remove airborne activity at the pool level. The discharge goes through the air exhaust duct, through the exhaust system butterfly damper, and into the stack. The stack is a 115-foot steel unit about 20" in diameter through which the total confinement building exhaust is released into the environment.

A constant air monitoring system monitors gaseous radioactivity that is exhausted to the stack. A continuous stream of gas is withdrawn from the stack through an isokinetic sampling tube and passed through the monitoring system. The stack monitoring system consists of a NaI crystal gaseous detector and a plastic scintillator particulate detector. A continuous graphical readout with alarm is located in the process control panel in the control room with remote readouts located in the Emergency Support Center. A constant air monitor detects airborne radioactivity in the reactor building. It is located on the main reactor floor and consists of an end window G-M detector with filter. The unit has an audible alarm with local and remote readouts.

The dilution air blower removes air from a basement lab and surrounding areas and discharges directly into the base of the reactor stack. The dilution air blower is required during normal operation since it dilutes the radioactive exhausts from other sources and also prevents radioactive exhausts from flowing downward in the stack into the laboratory and surrounding areas. This blower is part of the confinement and containment systems as described in Chapter 6.

9.1.2 Design Basis, Surveillance and Testing

The heating system is used for personnel comfort and also for maintaining proper operating environment for equipment. The heating system plant supply (hot water circulating system) is monitored by the University of Rhode Island / Graduate School of Oceanography maintenance service personnel. The Emergency Plan details the actions to be taken if heat is lost to the building. (Reference No. 9-1)

The ventilation system design basis is to insure safety of personnel and assure compliance with the 10 CFR Part 20 limits for the release of airborne radioactivity. The Technical Specifications describe the confinement and emergency exhaust systems and the applicability and specifications of the monitoring systems for airborne effluents to comply with the 10 CFR Part 20 limits. (See Chapter 14)

The operating procedures assure reliability of the stack monitor system through monthly testing. (Reference No. 9-2) The operating procedure limits reactor operation without the stack monitor.

9.2 Fuel Handling

9.2.1 General

The LEU core contains fuel elements that can be in use or stored in the underwater storage racks (See Section 9.2.3). Additional fresh fuel elements can be stored in the fuel safe.

Generally speaking, the types of fuel movements consist of (1) receiving fresh, unirradiated fuel elements and transferring them into the reactor pool storage racks and/or loading them into the reactor core grid; (2) unloading irradiated fuel elements from the reactor core into the storage racks; (3) relocating fuel elements within the core; (4) removing fuel elements from the reactor into storage racks and vice versa; (5) transferring spent fuel from the storage racks into the fuel cutoff assembly; and (6) transferring cut-off fuel into the cask for shipment.

9.2.2 Fuel Handling Equipment

The following is a general description of the handling devices used in routine fuel movements in the pool.

[redacted]

[redacted]

9.2.2.4 Viewing Box

The viewing box is a floating device that has a clear plastic viewing area in direct contact with the water, thereby reducing reflection and eliminating ripples that would otherwise interfere with underwater visibility. Lifting eyes on opposite sides of the viewing aid provide a means of lifting it from the water or securing it in a set position.

9.2.2.5 Cutoff Fuel Handling Tool

[redacted]

9.2.2.6 Overhead Crane

The reactor building has an electrically driven 15-ton overhead bridge crane. The crane is a 2-speed type with a hand held control box for controlling trolley and hook movements. The most recent modernization of the crane system was performed in 1988. The unit was upgraded to 15 tons and all work performed in accordance with OSHA 29 CFR 1910 and was also load tested (To 125% of Capacity) in accordance with OSHA. The use of the crane is specific in that it is limited in travel location when [redacted] It is also limited to path and wind load criteria. This information is the basis for the operational procedures for the use of the crane. (See Reference No. 9-2)

9.2.3 Pool Fuel Storage Racks

The single row racks (9 element capacity), Figure 9-1, top view and Figure 9-2, side view, (Ref. GE DWG 192C580) and the double row rack (18 element capacity), Figure 9-3, (GE DWG 192C581) are designed with a cadmium sandwich (poison) along the entire back of the units. (Part 1 of GE DWG 192C581 and Part 6 of 192C580) These are original design components of the facility and have been used satisfactorily for many years.

9.2.3.1 Storage Rack Design Bases

GE performed the original design of the single and double row racks as part of the original license. The racks are attached to the pool walls and/or bottom of the pool at the wall (see Figure 9-4). The racks were designed with sufficient spacing between fuel elements to insure that the array, when fully loaded, will be substantially subcritical.

The racks are located deep enough below the pool surface to assure adequate radiation shielding to personnel at the pool level. The racks are designed to permit proper handling during insertion, removal or interchange movements and to allow visual inspection for proper identification and seating in the bottom grid to prevent tilting or interfering with adjacent elements.

9.2.3.2 Rack Description

The racks are made of aluminum and have stainless steel hardware. The cadmium sandwich has aluminum plates on both sides and all screwed together. All materials of construction are compatible with reactor in-core components and also fit the graphite and beryllium reflectors, aluminum radiation baskets and, of course, fuel. The fuel racks have a $\Delta k/k < 0.8$ and are safe for use with the LEU fuel. (See Chapter 13) and (See Reference No. 9-9)

9.2.4 New Fuel Receipts and Storage

Cold, fresh fuel being received by the facility is handled in accordance with administrative controls and procedures. Vendor quality assurance and facility quality assurance, accounts for proper delivery and receipt at the reactor facility. Once the shipment has arrived, the facility implements a quality assurance procedure (See Reference No. 9-2). The procedure documents vendor delivery submittals, packaging and shipping, surface finish and defects and fuel element cleanliness and surface contamination. The procedure was developed in accordance with Reference No. 9-3. The elements are tested for core fit in accordance with the operating procedures. The elements are then stored in the fuel safe in accordance with Technical Specifications or stored in underwater storage racks. These storage devices assure that k_{eff} is less than 0.8% $\Delta k/k$. (Reference 9-9) documents previously analyzed acceptability for k_{eff} being less than 0.8%. The reactor room has a criticality monitor located in the vicinity of the fuel safe (meeting 10 CFR Part 70.24) which has audible alarm, setpoints and readout features.

9.2.5 Irradiated Fuel Handling

9.2.5.1 Normal in Pool Handling

[redacted]

All irradiated fuel movements are performed underwater and all are logged in the facility logbook. All core loading changes shall be performed under the immediate supervision of a licensed senior operator. At least one other technically qualified person shall act as an observer. All work and required systems are controlled by implementation of the facility operating procedures. (Reference No. 9-2)

Radiological safeguards systems (e.g., radiation monitors, etc.) are in use as described in the operating procedures. A licensed senior reactor operator determines the exact procedure for routine fuel movements, such as reloading the core. Operational procedures assure that the reactor core is at least 3% $\Delta k/k$ subcritical at all times. Technical Specifications (See Chapter 14) describes the surveillance of LEU fuel elements. The underwater storage racks have been used for over 38 years. Technical specification/administrative control section describes all necessary review requirements for all the operational procedures necessary for the safe handling of irradiated fuel.

9.2.5.2 Preparation for Shipment

Irradiated fuel, which has reached its useful lifetime, is stored in the underwater storage racks. The preparation and shipment of irradiated fuel is a task that is performed in accordance with the facility's operating procedures, an NRC approved quality assurance program for which ever fuel shipment method and container is approved and certified at the time of shipment for this particular LEU fuel type. The program must be approved for use through up-dated certificates of compliance specifically for the designated shipping package before spent fuel can be shipped.

[redacted]

[redacted]

9.3 Fire Protection System and Programs

9.3.1 Design Basis

The design basis for the Rhode Island Nuclear Science Center fire protection systems is to provide a detection and suppression capability which will mitigate any losses should a fire develop. It should be noted that fire protection is not required to accomplish a safe shutdown condition. If a loss of electrical power occurs the reactor automatically scrams.

9.3.2 Description

The Rhode Island Nuclear Science Center has fire protection from various sources. A wet pipe sprinkler system using water flow is the most effective system. Some spaces are monitored by heat sensors that detect a fire based on a rise in temperature. In addition a secondary Aerotherm fire detection system that monitors the reactor containment building is described below.
The fire protection systems are monitored by a private security company using a dedicated telephone modem.

a. Aerotherm System

This system consists of a small copper tube that surrounds the reactor biological shield within the confinement building. A rise in pressure within the tubing causes a local and remote alarm (alarms at "SC"). The system is tested in accordance with the RINSC operating procedures.

b. Fire Sprinkler System

A wet pipe (city water supply) system employing fire sprinkler heads. A locked standpipe post indicator valve outside the building assures proper valve positioning for water supply to the facility sprinkler system. The valve has a supervised tamper switch. There is a sprinkler flow alarm. A fire hydrant is located about 50 feet from the reactor building. (See Figure 9-6 for the fire protection system schematic.)

The reactor building has a fire detection system and also has portable carbon dioxide and dry chemical extinguishers deployed at strategic locations. The reactor control room has the Aerotherm system, a portable extinguisher, and an emergency escape door to a fire escape, and a series of sprinkler heads around the biological shield and building wall perimeter. The escape door is a metal fire door. The automatic fire detection systems are monitored 24 hours per day. The reactor room portal has fire doors at each end. The local fire department is the Narragansett Fire Department. It can be summoned by phone or by using a fire alarm box. In addition, there is a

[redacted]

The facility has offsite agreements with the local Fire Department, Police Department and other agencies, as described in the Emergency Plan, who will respond in case of fire. The facility Emergency Plan and the contracted alarm monitoring company have written procedures to follow in case of fire. The Facility Emergency Plan procedures (Reference No. 9-1) and the Facility Security Plan (Reference No. 9-4) define the fire response and appropriate notification procedure. The facility safety plan addresses the facility fire safety program (See Reference No. 9-8).

The facility minimizes the storage of combustibles and does not allow explosives into the reactor building in accordance with the facility safety program and operating procedures. Reactor staff response, by performing in accordance with its above referenced procedures, further minimizes the possibility or limits the consequences of a fire. The reactor operator, in accordance with operating procedure, shuts the reactor down if a fire occurs in the reactor room or control room. A fire in other parts of the building may also require a shutdown on notification by personnel. Since the building and biological shield is concrete, as well as other sections of the facility, the fire potential is lessened. Irradiated fuel is stored in the reactor pool and does not represent a fire hazard. New fuel is stored in a locked steel fire resistant vault. No other quantities of radioactive materials are stored in the reactor room that, if released due to a fire, represent a serious threat to personnel.

9.3.3 Evaluation

The fire protection program and systems have been designed to meet and exceed the design basis.

9.4 Communication System

The facility has an Uninterrupted Power Supply (UPS) system for the telephone/pager system. The facility also has a phone system equipped with local paging. Information regarding the status of reactor operations, normal reactor startup and shutdown, emergencies and other announcements are made using the general paging system accessed from any telephone. The telephone system can also be used, but the speakers are only at the local telephones, whereas the general paging can be heard throughout the facility. Direct communication using both the pager and telephone systems are shown in Figure 9-7.

[redacted]

The Emergency Plan Implementing Procedures (EPIP) discuss the actions to be taken upon loss of telephone service. In case of a required notification to the NRC, the EPIP specifies what type of communication is required (see Reference No. 9-1). Other specific emergencies are discussed separately in the EPIP. General verbal communication between operating personnel such as Senior Reactor and Radiation Safety Officer, experimenters etc. are discussed in various sections of the facility operating procedures.

Changes in shifts, core loading operations, critical experiments, bridge movements and other reactivity changes required direct communication between the reactor operator and other personnel. The operating procedures spell out how this is accomplished. Using the intercom at the two stations performs direct communication between the Reactor Operator and experimenters using the pneumatic system. Other experimenters as necessary use a phone in the reactor room main floor. Emergency testing of the emergency evacuation system is performed weekly and the paging system is used to notify all personnel in the building that a test is being conducted. The [redacted]

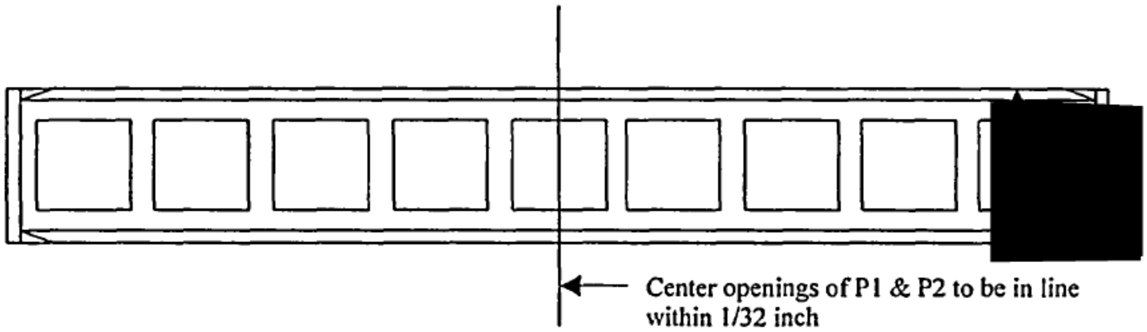
These systems have performed satisfactorily in the past and the reliability has improved greatly with the addition of the UPS and power supply back up systems a few years ago.

9.5 Building Water System

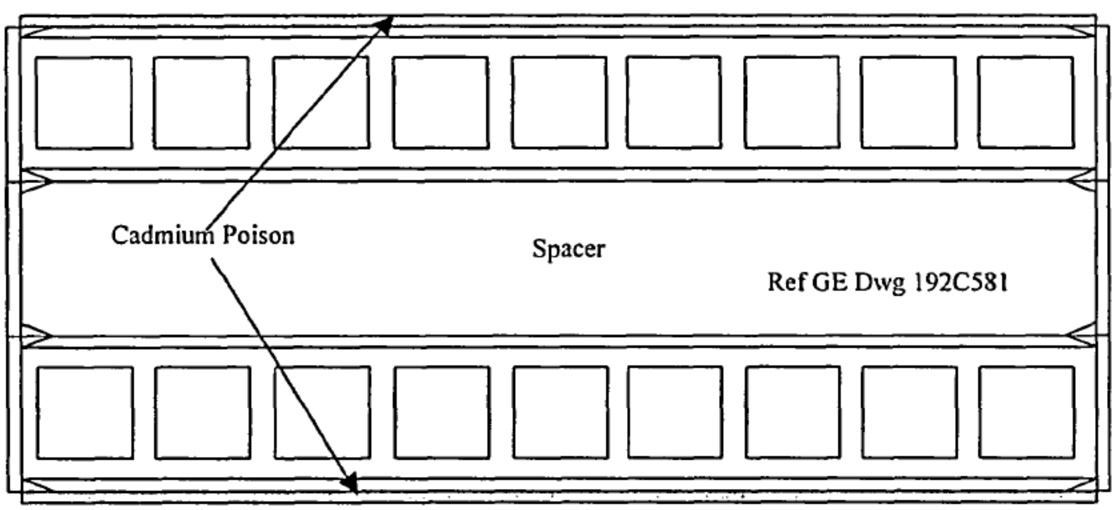
The facility water system consists of an 8" combination fire and domestic supply from a 12" University of Rhode Island Bay Campus supply system. That system is supplied via the Wakefield Water Company. The supply from the Wakefield Water Company goes to a 300,000 gallon storage tank. Water is stored and pumped to the Bay Campus through an array of three domestic water pumps and a fire pump. All pumps are equipped with emergency generator backup power. The supply to the Rhode Island Nuclear Science Center is a 6" fire sprinkler system, a 4" standpipe/fire hose in the reactor building and a 2" line supplying all other facility uses including cooling water. (See drawing "URI Bay Campus Water System", Figure 9-8). The facility ECCS supply is described in Chapter 6 and 13. The pool makeup system is described in Chapter 5.

CHAPTER 9 REFERENCES

9-1 RINSC Emergency Plan
9-2 RINSC Operating Procedures
9-3 TRTR-5 Fabrication Requirements
9-4 RINSC Security Plan
9-5 BMI-I Cask Letter of Compliance
9-6 RINSC Quality Assurance Program
9-7 NRC Approval Letter
9-8 RINSC Safety Plan
9-9 IAEA-TEC-643, April 1992, Appendix N-3.1 "Nuclear Criticality Assessment of LEU & HEU Fuel Storage" Argonne National Laboratory



Single Row Fuel Element Storage Rack (top view)



Double Row Fuel Element Storage Rack (top view)

Figure 9-1

RINSC Fuel Element Cut-off Saw

figure 9-3

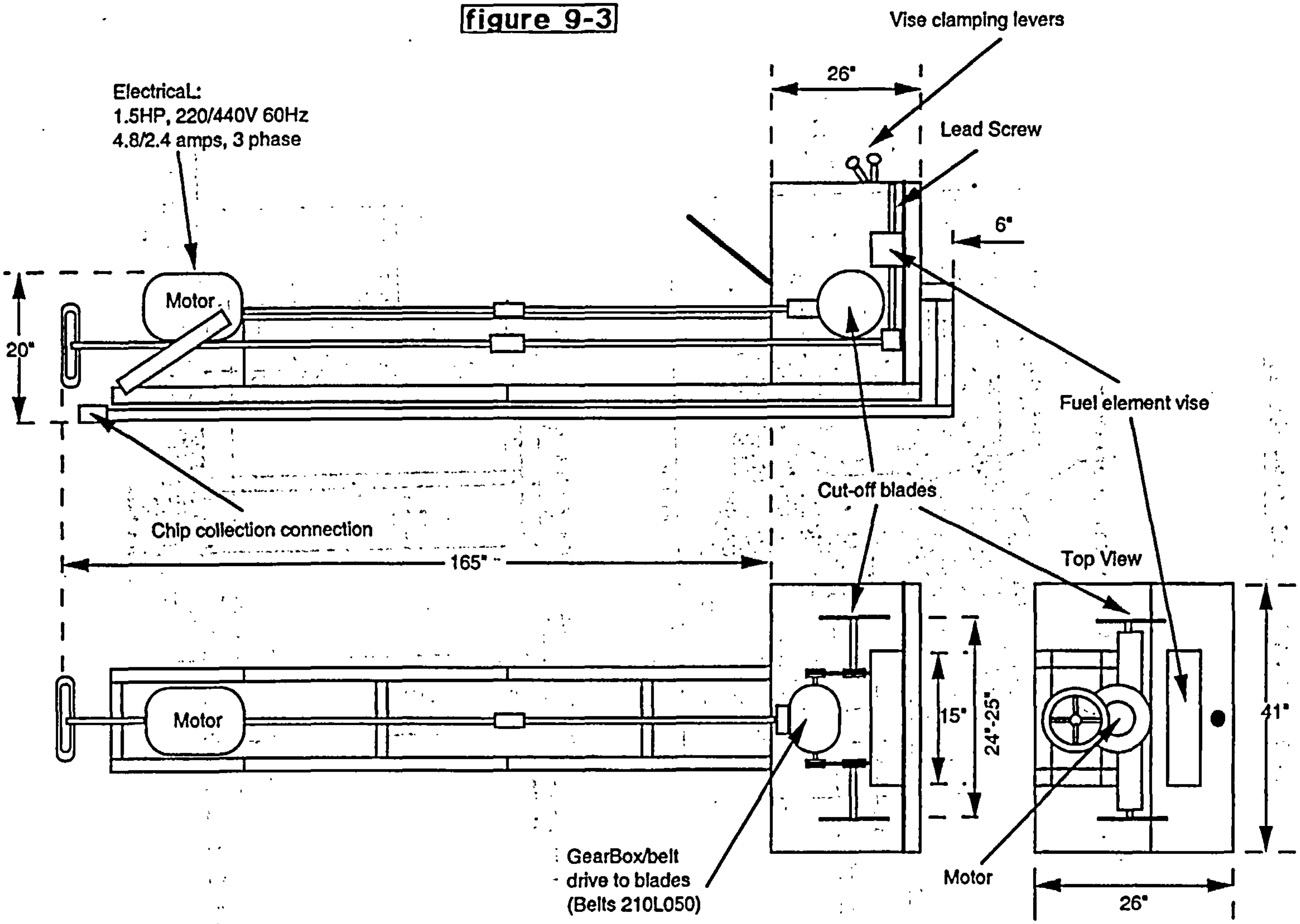


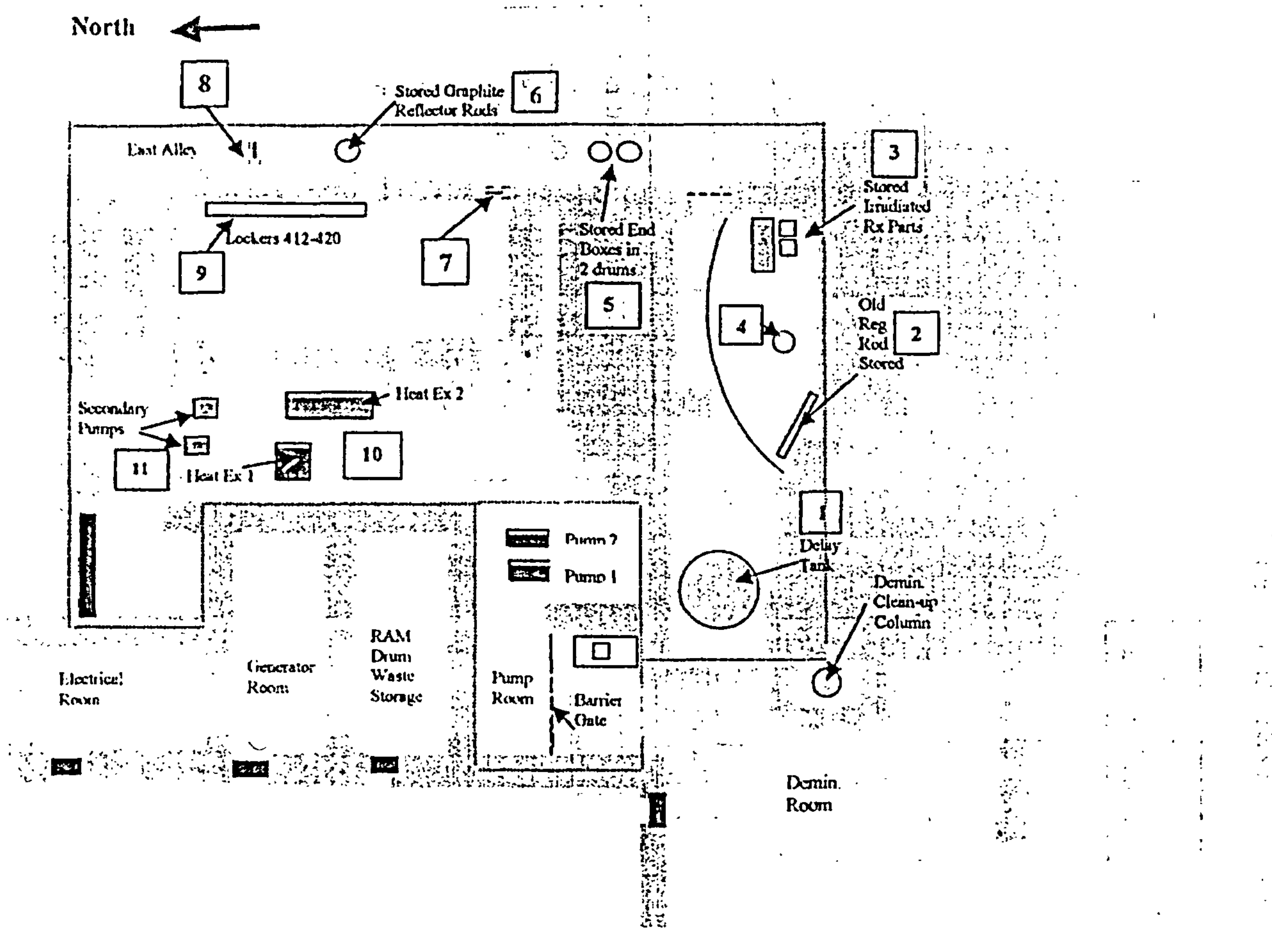
Figure 9-2

"LOCATION OF FUEL STORAGE RACKS"

[redacted]

[redacted]

Figure 9-3



18 April 2003 - **Survey of Heat Exchanger Room and Auxiliary Areas**
Time of Survey: 1430 Hours; Conditions: Reactor last run on Thursday, 17 April in AM for short time.
Surveyed using 740 Victoreen Cutie Pie; Note: Numbers below keyed to Map above.

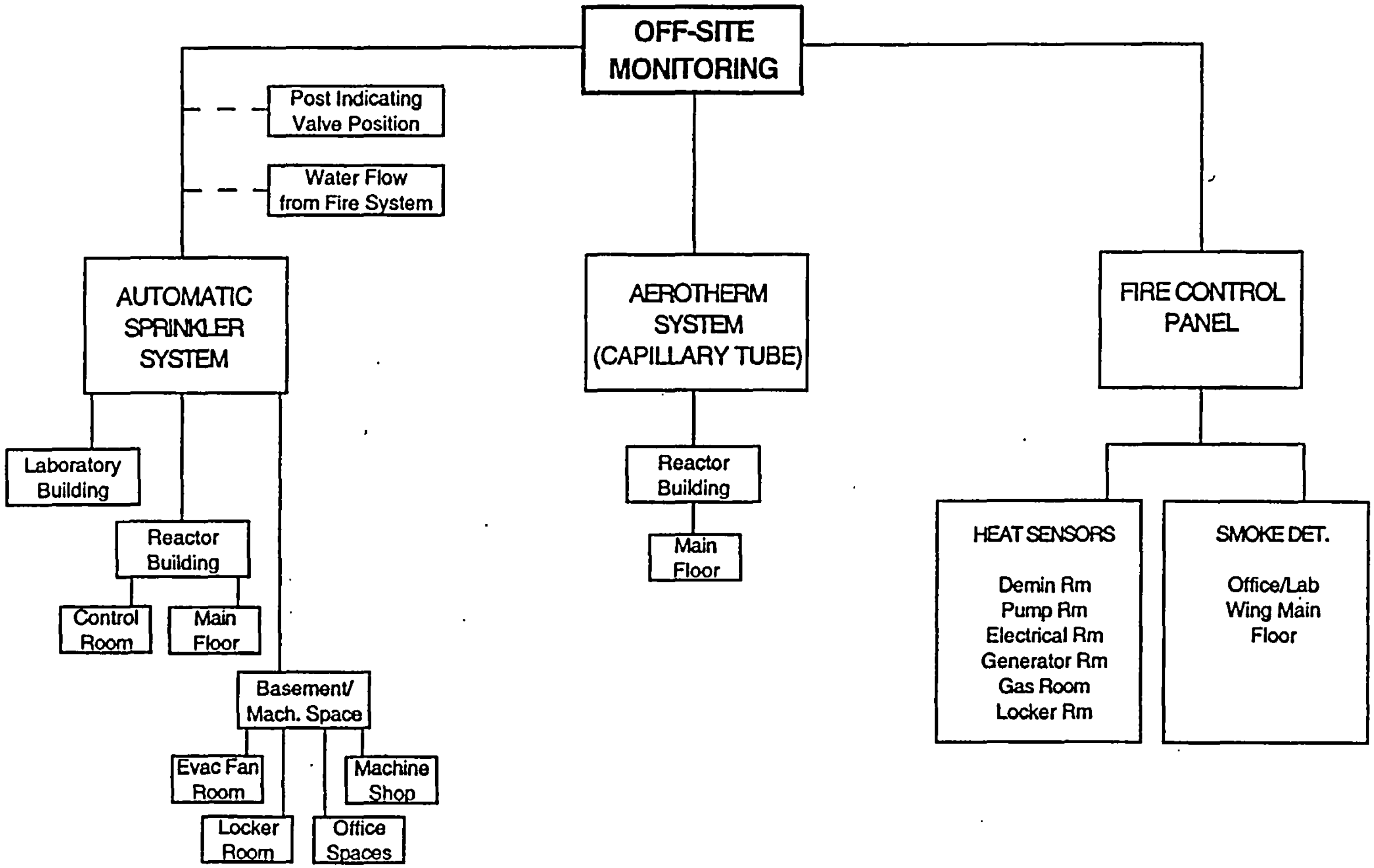
1) 2.5 mrem/hr @ contact with Delay Tank, FPT smear of floor: 32 dpm
2) 200 mrem/hr contact with concrete shielding blocks for stored old Rx regulator blade.
3) 6 mrem/hr in area of Stored irradiated Rx Parts (contact/rubber gasket)[Al baskets, graphite st ringers,etc.]
4) 6.5 mrem/hr @ contact with 55 gal. Drum containing bagged RAM
5) 250 – 500 mrem/hr @ contact of two unshielded 55 gallon drums containing stored End Boxes
6) 200 mrem/hr @ contact with unshielded 85 gallon drum containing Old Graphite Reflector Rods
7) 4.0 mrem/hr @ signed rope barrier leading to East Alley Way on south side
8) 3.0 mrem/hr @ signed rope barrier in East Alley Way, north side.

Surveyed using Model 3 Ludlum with Pancake Probe:

9) 1.5 mrem/hr @ contact with Locker No. 412
10) 1.5 mrem/hr @ contact with Primary Loop 2 piping
11) BKG in area of secondary pumps.

Figure 9-4

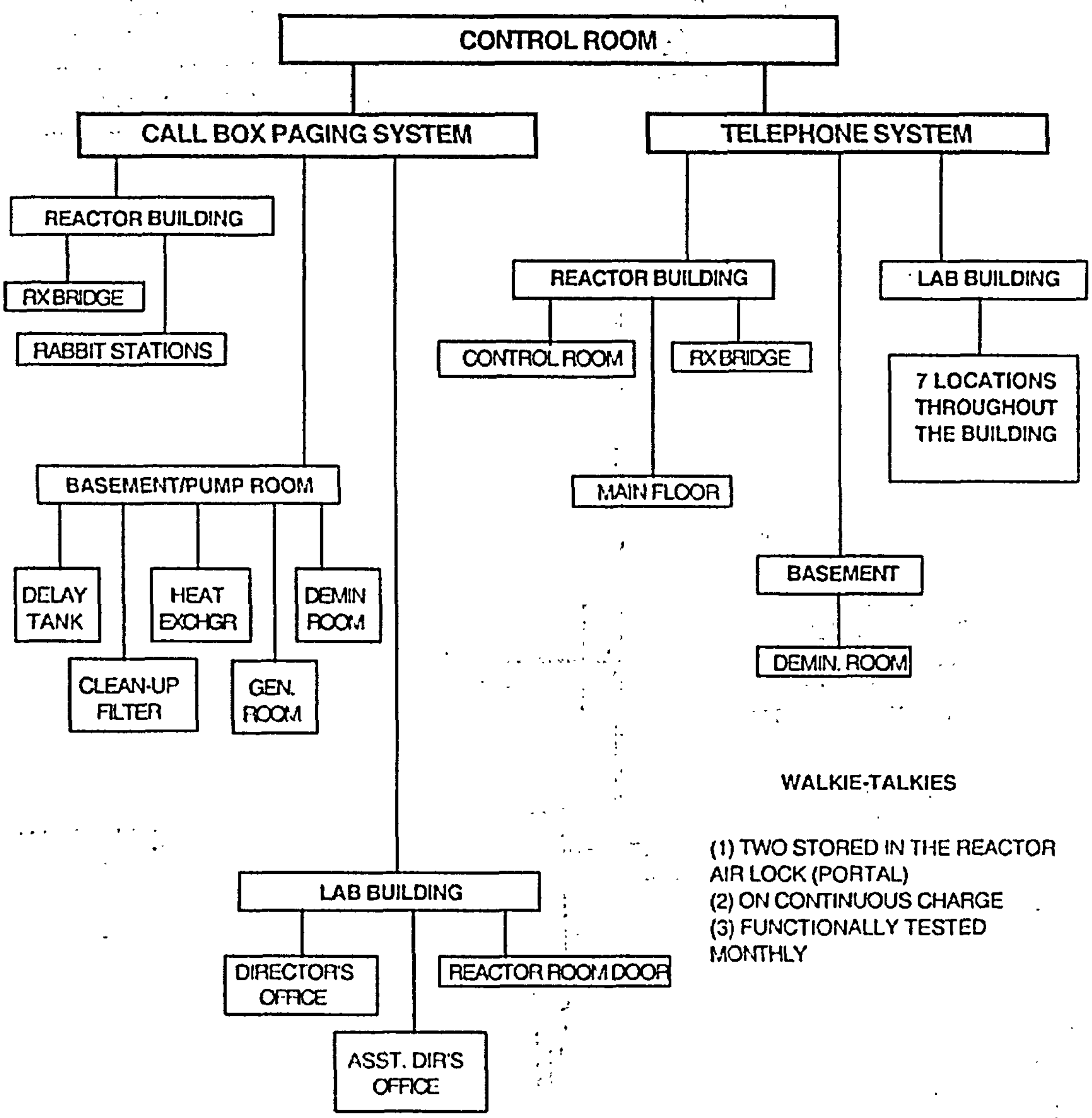
FIRE PROTECTION SYSTEM SCHEMATIC



NOTES:
1. All areas have portable fire extinguishers
2. Gas storage for emergency generator is outside

Figure 9-5

COMMUNICATIONS SYSTEM SCHEMATIC



NOTE: ALL LOCATIONS OF CALL BOXES AND TELEPHONES CAN CALL THE REACTOR CONTROL ROOM DIRECTLY.

Figure 9-6

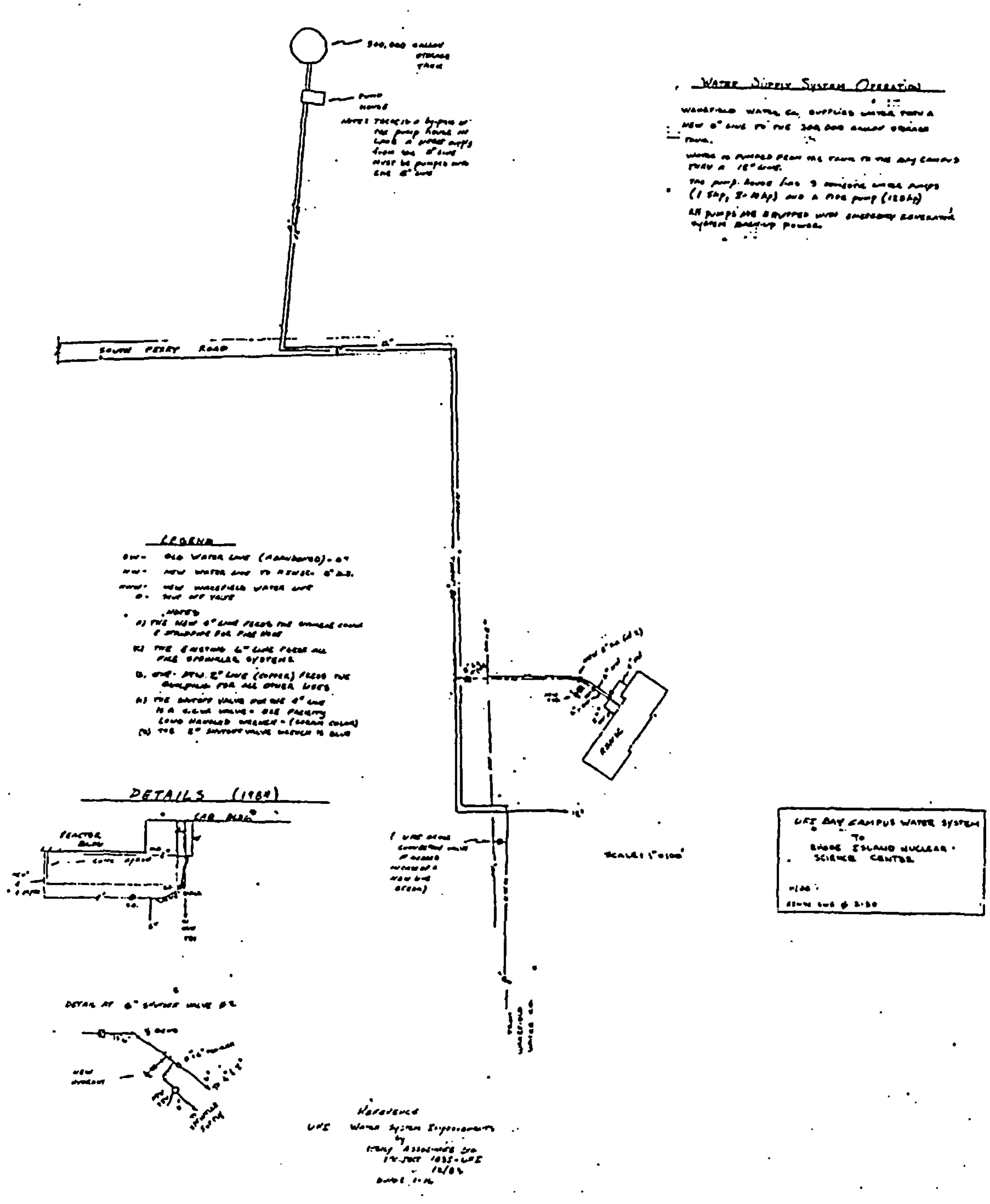


Figure 9-7

CHAPTER TEN

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10.0 EXPERIMENTAL FACILITIES AND UTILIZATION

This chapter describes and discusses the experiment facilities at the reactor facility, their intended uses, and the experiment program.

10.1 Summary Description

The experiment program at the Rhode Island Nuclear Science Center (RINSC) continues to provide a wide range of analytical, radiographic, and irradiation services to the educational, research, and commercial sectors. The RINSC is designed to provide these services through the use of the following experiment facilities:

- Six beam ports for neutron beam irradiation; a through-port, under the reactor core, is also available for use;
- Two pneumatic systems provide direct irradiation of small samples for neutron activation;
- A thermal column, composed of graphite, is available for various types of neutron radiography and other types of irradiations;
- A dry irradiation room, located behind the fuel storage end of the pool, can be used for gamma irradiations;
- A flux trap in a beryllium reflector element located in the center of the core for high flux irradiations; the radiation baskets on the perimeter of the core also are available for sample irradiations; and
- A fuel storage rack, fitted with a pipe (glory-tube), extends to the pool surface that is used for gamma irradiations; samples can be lowered down the tube and suspended next to fuel elements (stored in the rack) that provide decay gamma rays to the samples.

To meet the needs of the user groups, the reactor operates one shift per day. Radiation protection, surveillance and experiment review, over the operating history of the reactor, meets all current licensing requirements.

10.2 Experiment Facilities

10.2.1 Beam Ports

10.2.1.1 Description

The reactor is designed to accommodate two sets of three (total of six) horizontal beam ports. These hollow ports provide a controlled pathway through the biological shield for neutrons with minimum scattering from the reactor core to beam ports located outside of the biological shield. Also provided in the original facility design is a through-port; a structure similar to the beam ports. The beam ports can be utilized for long-term irradiations and neutron beam extraction experiments such as neutron scattering, neutron spectroscopy and radioisotope production. Thermal neutron flux at the core end of the beam ports varies between 1E12 n/cm^2/sec and 4E12 n/cm^2/sec.

The biological shield contains two sets of three beam ports; one set is located on the north side and the other is located on the south side of the reactor pool. Each set is composed of one 8-inch diameter beam port and two 6-inch diameter beam ports. The beam ports are positioned radial with respect to the reactor core and are spaced at 30-inches in the horizontal plane as shown in the pool outline drawing (GE198E273) and Figure 10-1. Figure 10-2 shows the location of the beam ports each of which penetrates the reactor pool wall and terminates on a face of the core. Each port is made up of four major sections: 1) an aluminum inner tube, 2) a lead shutter, 3) an aluminum shutter assembly, and 4) an outer steel tube. Figure 10-3 shows the port construction assembly. The end thimble extends into the pool. The inner tube extends to the shutter, a lead shield to which is attached a cable used for raising or lowering the shutter. The shutter sits in an aluminum housing which has a drain leading to the waste water retention facility. The shutter assembly has a vent leading to the off-gas removal system. The vent removes radioactive gases formed in the beam port when the reactor is in operation. The end of the beam port has two heavy concrete plugs fitted with an instrument lead tubes, skewed to limit radiation levels. An outer cover is bolted to the flange at the biological shield. It provides a positive seal and can withstand system pressures. The primary function of the plugs is to minimize radiation levels at the biological shield, even with the shutter open. Any experiments using the beam ports without the plugs in place require special design of the shielding. The radiation protection program provides proper control over this type of use. The possibility of loss of coolant, as analyzed in Chapter 13, is limited based on the design of the cover plate flange having a maximum of one-half inch diameter hole penetration.

10.2.1.2 Evaluation

The beam ports, by the use of the shutter and plug assembly in their design, provide the necessary shielding for personnel working on the main reactor floor in and around the beam port. The thimble end of the beam port is seal-welded to the outer tube and no leaks have occurred during the operation history of the reactor. Gas removal and liquid removal capability control the generation of any activated effluents created in the port. The beam port shutter provides the main attenuation of the neutron and gamma radiation beam. The shutters can be raised and lowered automatically by the experiment operator only after requesting and receiving permission from the reactor operator. Shutter operations are under supervision of the control room by permissive switches. The reactor operator in the control room activates the shutter control motor to raise and lower the unit and can override an up command from the experiment operator with a down drive command. In summary, reactor operator procedures are in place and are used to control the opening of each beam port shutter. The radiological safety program is used to set up and test new experiments and associated shielding configurations prior to initial shutter openings during reactor operation. New beam port experimental shield configurations are typically and routinely meter-surveyed at low power levels. Survey results are mapped and documented. Dose assessments are then projected to higher working-power levels and additional shielding is added, if necessary, and controlled areas are defined, marked and mapped before going to the targeted higher working power level(s). A final health physics survey of floor areas in and around the beam port area of the experiment is conducted to assess the previous projections. Periodic health physics re-surveys are conducted and documented to assure that on-file radiation levels remain the same.

10.2.2 Through-port

10.2.2.1 Description

The through-port location is shown in Figure 10.2. The six-inch diameter through-port construction is shown in GE drawing number 762D250. This is an original facility design structure, similar to the beam ports.

The six-inch horizontal through-port passes through the pool and each end terminates at the outside face of the biological shield. Figure 10-4 shows the details. Since it is located beneath the core and therefore could drain the pool below the bottom of the core, each end of the port is assured to be watertight by bolted blind flanges or a manual operated gate valve (closed position) during reactor operation. If the through-port is being used, administrative controls are established to assure that the through-port flanges are bolted water tight and/or the valves are closed prior to reactor operation. Radioactive gases are removed by the off-gas blower. For the experimental use of the facility, safeguards against loss of coolant, will be assured, by administrative control and surveillance, and also that the through-port be opened in accordance with written procedures and secured prior to reactor operation. Appendix C and D of Reference 3 address calculations relating to the decay heat aspects of a LOCA. In order to assure that no loss of coolant could occur through an open through-port prior to the decay heat not being low enough to cause fuel melting, the opening of the through-port should be delayed after a reactor shutdown. A conservative time based in the SAR is 12 hours. Since personnel would be working at the through-port during the removal and installation of the experiment, experiments in the through-port may require instrument leads to extend out the flanged ends. The total opening sizes should be restricted, (similar to the beam ports) to ½-inch diameter.

10.2.2.2 Evaluation

A recent 10 CFR 50.59(5) review was conducted on 6/19/96 for replacement of the through port cover flanges (steel) with a PVC plate type flange. Based on this evaluation, no technical specification changes are required and no changes to the current SAR were necessitated.

10.2.3 Pneumatic System

10.2.3.1 Description

The pneumatic tube system, shown in Figure 10-6, is designed to quickly transfer individual samples to and from positions adjacent to the reactor core for irradiation. The thermal neutron flux in the pneumatic tubes ranges from 3E12 n/cm^2/sec to 4E12 n/cm^2/sec.

Two pneumatic tubes are provided for rapid movement of small experimental specimens to and from the high flux region adjacent to the core in the high power section of the pool. Each system includes a sending and receiving station on the main reactor room floor (south side). The sample travels through the inner concentric aluminum tube. The tube extends from the station into the pool down to the reactor. A 2-inch diameter inner tube guides the sample. The 3-inch outer tube is connected through suitable solenoid air control valves to the exhauster (turbine blower). The system schematic is shown in Figure 10-6. The control air is maintained via the solenoid cabinet operation. The exhaust gases are sent through a HEPA filter to the reactor room exhaust system to the stack via off-gas system blower. Solenoid valves direct flow through the inner tube to either send or receive a sample. The sending unit positions the sample for transfer to the terminus at the end of the tube near the core.

The shielded receiving unit holds the sample after irradiation. The maximum speed of the sample is about 50 feet per second. Samples can be sent to the terminal in each system with a payload of about 2 pounds. Manual ball valves are located in the pneumatic air lines in case of a tube leak below the pool water elevation. The valves can be manually closed to prevent pool drainage or siphoning. The samples can receive irradiation for certain lengths of time by presetting the automatic mode system timer. The samples are automatically returned to the receiver when the time period ends. Manual operation is provided. The controls are located in each system control panel. The operator has overall control of the system. Radiation levels near the receiver are sensed by the local area monitors that provide an audible alarm and flashing light when "returned" samples exceed preset radiation levels.

10.2.3.2 Evaluation

The system has operated successfully from the initial license. The reactor facility has added a "sample" hood in the reactor room for experimenters to transfer and open samples from the receiver to allow transport of samples to the facility laboratories. The hood is vented through a HEPA filter exhausting near the building exhaust system intake, in case of sample spillage or gaseous release. The pneumatic tube system is the most used experimental device for neutron activation. All of the items referenced in 10.2.1.3 are applicable. The receiver shielding limit the personnel radiation levels. Use of the pneumatic system and the type of materials irradiated are controlled by operating procedures and the facility radiation safety program.

10.2.4 Thermal Column

10.2.4.1 Description

The design basis for the thermal column is to provide a thermal neutron flux for experiments such as neutron radiography. Thermal neutron flux in the thermal column is approximately 4E06 n/cm^2/sec.

The thermal column is a graphite-filled horizontal penetration through the biological shield that provides neutrons of the thermal energy range (about 0.025 ev) for experiments. The column is approximate 56 inches x 56 inches square and 8 foot long.

It extends from the reactor core to the outer face of the biological shield. A water-cooled lead shield adjacent to the core attenuates gamma radiation, thereby reducing heat generation within the column without seriously impeding the transmission of neutrons. Removable stringers in the graphite pack provide a 16-inch square opening for experiment samples, thus providing flexibility of radiation experiments by changing the speed and thermal energy range of the neutron flux through the column. A small experimental air chamber between the face of the graphite and the thermal column has conduits for service connections (air, water, electricity) to the biological shield face.

Personnel in the reactor building are protected against radiation by a heavy concrete door that closes the thermal column at the biological shield. Each of the 16 graphite stringers consists of two interlocked sections of equal length. Except for a 1-inch diameter hole in the end of the outer sections, the 2 sections of each stringer are identical. The hole is tapped to accommodate a customer-supplied handling tool. Each stringer in the central array weights approximately 94 pounds. When the reactor is operating at licensed power, cooling air is required to prevent heat generated by gamma radiation from raising the graphite temperature above 107°C (225°F) and to remove radioactive gases and airborne corrosion products from the thermal column. Reactor room air is drawn into the experimental chamber through the clearance gap between the thermal column door and casing. An aluminum cooling plate in the graphite pack allows the air to flow downward through the graphite and out the ventilation tube on the floor of the chamber. A pipe manifold to the ventilation tube leads the air away to the off-gas blower from which the air is exhausted to the ventilation stack.

The blower runs continuously while the reactor is operating, and also when the thermal column door is open during reactor shutdown in accordance with the RINSC operating procedures. The thermal column door is approximately 57.5 inches, square by 21 inches thick. The door is mounted on four flanged, railroad-type wheels running on rails set into the concrete floor, and can be moved perpendicular to the shield face by means of a motor-operated chain drive mounted on the door face. Figure 10-2 shows the location of thermal column on the reactor room main floor. The construction of the thermal column is shown in Figure 10-5. The thermal column door is shown in GE drawing 762D410.

10.2.5 Dry Gamma Room

10.2.5.1 Description

A room adjacent to the spent fuel pool that can be used by moving the reactor bridge over to the low-power section of the pool. It has a leak proof aluminum window (part of pool liner) that is available to provide bulk irradiation of experiments in air by moving the core adjacent to the window.

The dry gamma room, which was built into the reactor biological shield, is located as best shown in Figures 1-1 or 10-1,The room floor is about 1 foot above the bottom of the low-power pool section. The window into the gamma room is made of double-thick aluminum window designed to withstand the static water pressure and any foreseeable abuse. The dry gamma room is presently un-vented (the vent line is closed off). The personnel access opening on the north side of the reactor room main floor, has been sealed shut. Concrete blocks, with a sealing plate inside the frame, were installed years ago to provide a structurally sound secondary containment barrier in the event of a pool leak into the gamma room.

10.2.5.2 Evaluation

This original design experimental facility is not in use at this time. The design was such that, if in use, the reactor can only be operated at 100 kW adjacent to the room. The production of small amounts of Argon-41 could be vented. Any future use by personnel would require the proper safety evaluations by the facility radiation protection program.

10.2.6 Gamma Tube

10.2.6.1 Description

The gamma tube is a three-inch diameter aluminum tube extending from the low-power section fuel storage rack (north side) to above the pool surface that provides gamma rays from stored fuel elements to samples lowered into the tube.

The gamma tube is usually positioned into the north rack into a fuel element storage space. The tube has an anti-flotation collar to overcome buoyancy and can be moved into different rack positions. The samples can be monitored during irradiation to provide the proper total dose.

10.2.6.2 Evaluation

Monitoring at the top of tube reveals dose rates below those at the pool surface level during normal reactor operation thereby making them acceptable. There are no radioactive gases produced in the tube that require special venting. Normal surveillance utilizes the reactor room monitoring equipment (e.g. area monitors, vamps etc.). Prior to using the gamma tube, appropriate analysis and experiment review must be performed to satisfy the requirements of the Technical Specifications, operating procedures.

10.2.7 Radiation Baskets

10.2.7.1 Description

Aluminum baskets that fit the grid box and have an open top that provides for in-core irradiations.

There are five baskets, interchangeable with fuel elements and reflector elements, placed into the reactor grid box. Although the baskets can be placed at other locations in the core, the design of the LEU core allows baskets to be located in core positions 9B, 9C, 9D, 9E or 9F. The aluminum baskets provide a chamber of about 2 5/8-inch square by 33 inches long for experiments. The baskets, including experimental contents, are cooled by pool water flowing through the primary coolant system. An aluminum orifice plate in the bottom of each basket permits the water to pass through the experiment chamber. A lifting bail at the top of the basket is provided by use of the remote handling tools.

10.2.7.2 Evaluation

The radiation baskets have successfully been used for many experiments over the years. Prior to the use of any of the baskets, appropriate analysis and an experiment review must be performed to satisfy the requirements of the Technical Specifications, operating procedures, 10 CFR 50.59 and other applicable regulations. The experiment review process defines the insertion, removal and radioactive handling procedures, including radiation monitoring, for the specific in-core experiment (See Chapter 11).

10.2.8 Flux Trap

10.2.8.1 Description

A flux trap in a special beryllium reflector element designed with a removable plug provides for high flux experiments. The thermal neutron flux at the center of the core in the flux trap is approximately 5E13 n/cm^2/sec.

The flux trap in a special beryllium reflector is called a "plug type" reflector and is capable of fitting into a standard grid box opening. The reflector has a removable center plug of about 1½ inches in diameter and 29 inches long which can be remotely removed by use of the facility fuel handling tool. Flux measurements were made showing that samples irradiated could receive peak thermal fluxes of up to 5E13 n/cm^{2}sec.

10.2.8.2 Evaluation

Since the use of the experiment device is considered an in-core experiment, the same evaluation applies to experiments as in 10.8.3. This device was approved for use in the 1992 Safety Analysis Report Part A, VIII.

10.3 Experiment Review

The Nuclear & Radiation Safety Committee is responsible for the review of all experiments prior to initial performance at the RINSC. New types of experiments or experiments of a type significantly different from those previously performed shall be described and documented for the study of the Nuclear & Radiation Safety Committee. The documentation includes the purpose of the experiment, a description of the experiment, and an analysis of the possible hazards associated with the performance of the experiment.

All uses of experimental facilities shall be approved by the Director to ensure that:

- the absolute value of the reactivity worth of any single independent experiment or the combination of connected experiments that can be added to the core simultaneously does not exceed 0.6% Δk/k;
- the calculated reactivity worth of any single independent experiment not rigidly fixed in place or the combination of connected or related experiments that can be added to the core simultaneously will not exceed 0.08% Δk/k;
- experiments installed in the reactor do not shadow the nuclear instrumentation such that erroneous or unreliable information is given to the control system safety circuits;
- experiments installed in the reactor can not fail in a manner that interferes with the insertion of a reactor control element;
- experiments do not involve materials that might credibly result in an explosion;
- experiments do not involve materials that could contaminate the reactor pool with corrosive agents;
- experiments are not performed that could credibly result in fuel element damage; and,
- experiments are not performed that result in more than one vacant fuel element position within the periphery of the active section of the core.

[redacted]

Figure 10-1

[redacted]

Figure 10-2

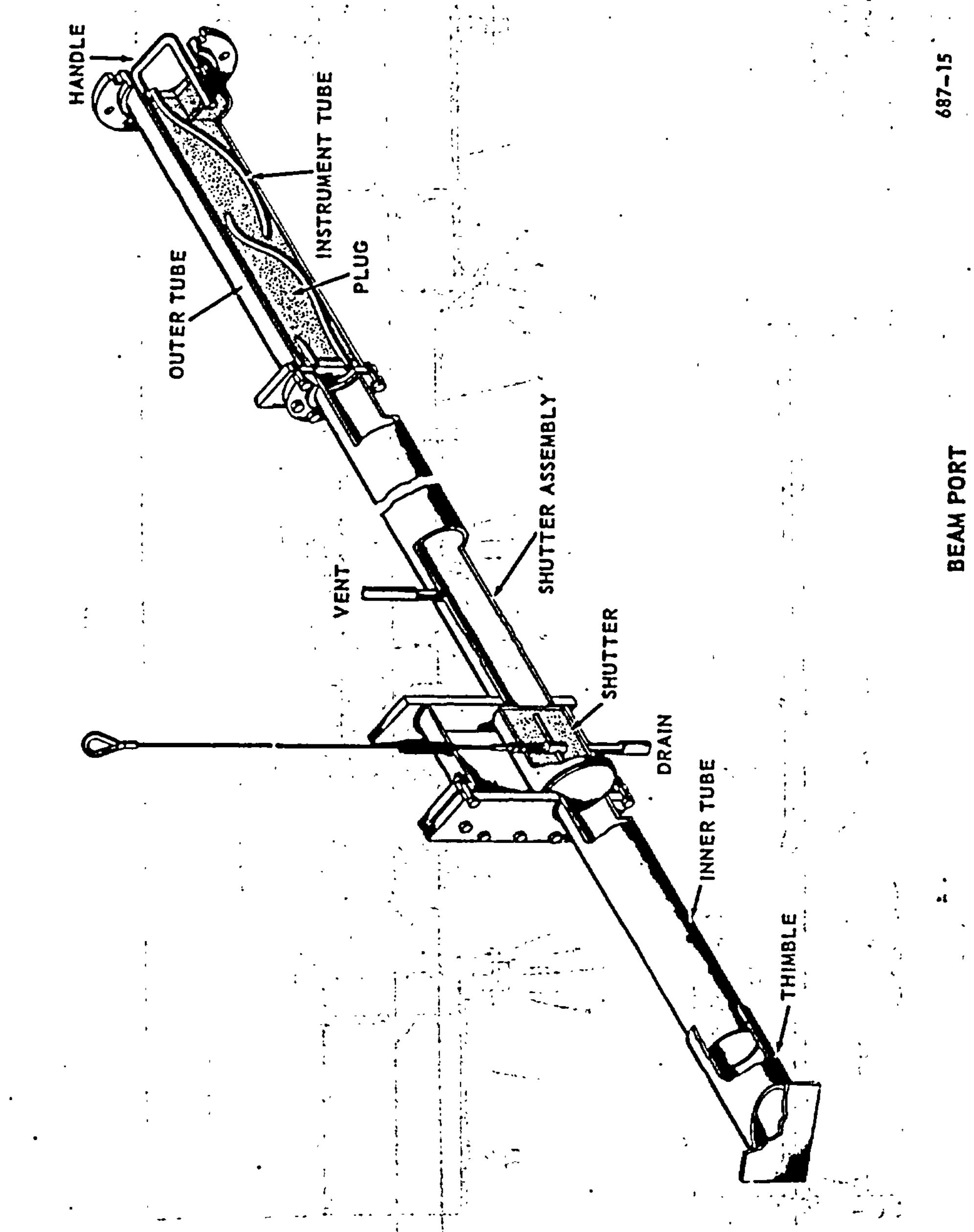
HANDLE
OUTER TUBE
INSTRUMENT TUBE
PLUG
VENT
SHUTTER ASSEMBLY
SHUTTER
DRAIN
INNER TUBE
THIMBLE
BEAM PORT
687-15

Figure 10-3

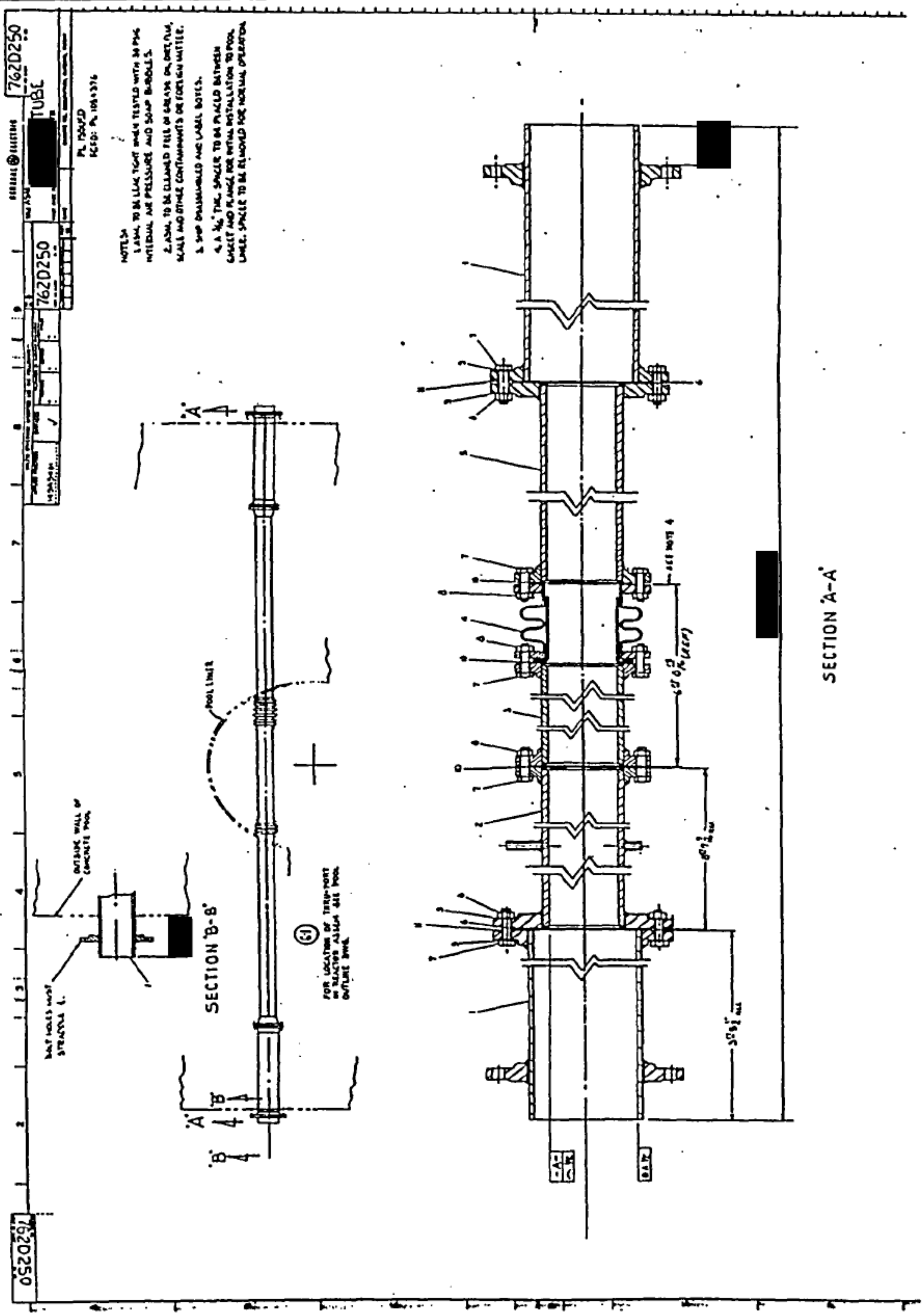
762D250
TUBE
NOTES:
1. ASM. TO BE LEAK TIGHT WHEN TESTED WITH 30 PSIG INTERNAL AIR PRESSURE AND SOAP BUBBLES.
2. ASM. TO BE CLEANED FREE OF GREASE, OIL, DIRT, FLUX, SCALE AND OTHER CONTAMINANTS OR FOREIGN MATTER.
3. SHIP DISASSEMBLED AND LABEL BOXES.
4. A 1/16" THK. SPACER TO BE PLACED BETWEEN GASKET AND FLANGE FOR INITIAL INSTALLATION TO POOL LINER. SPACER TO BE REMOVED FOR NORMAL OPERATION.
BOLT HOLES MUST STRADDLE ℄
OUTSIDE WALL OF CONCRETE POOL
SECTION 'B-B'
POOL LINER
FOR LOCATION OF THRU-PORT IN REACTOR ASSEM SEE POOL OUTLINE DWG.
SEE NOTE 4
SECTION 'A-A'

Figure 10-4

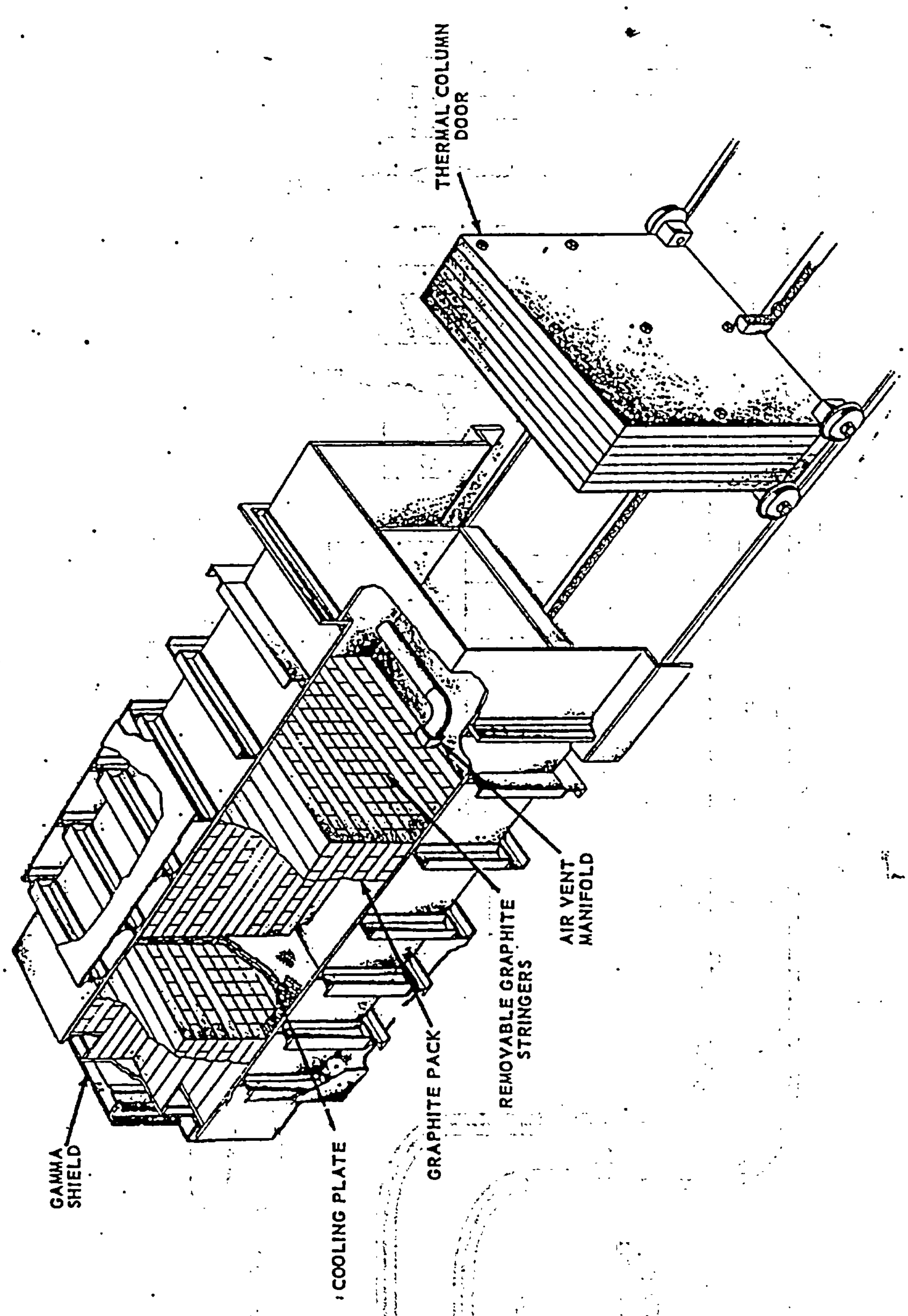
THERMAL COLUMN DOOR
GAMMA SHIELD
COOLING PLATE
GRAPHITE PACK
REMOVABLE GRAPHITE STRINGERS
AIR VENT MANIFOLD

Figure 10-5

$2\frac{1}{4}$" AIR VACUUM PNEUMATIC SAMPLE IRRADIATION SYSTEM

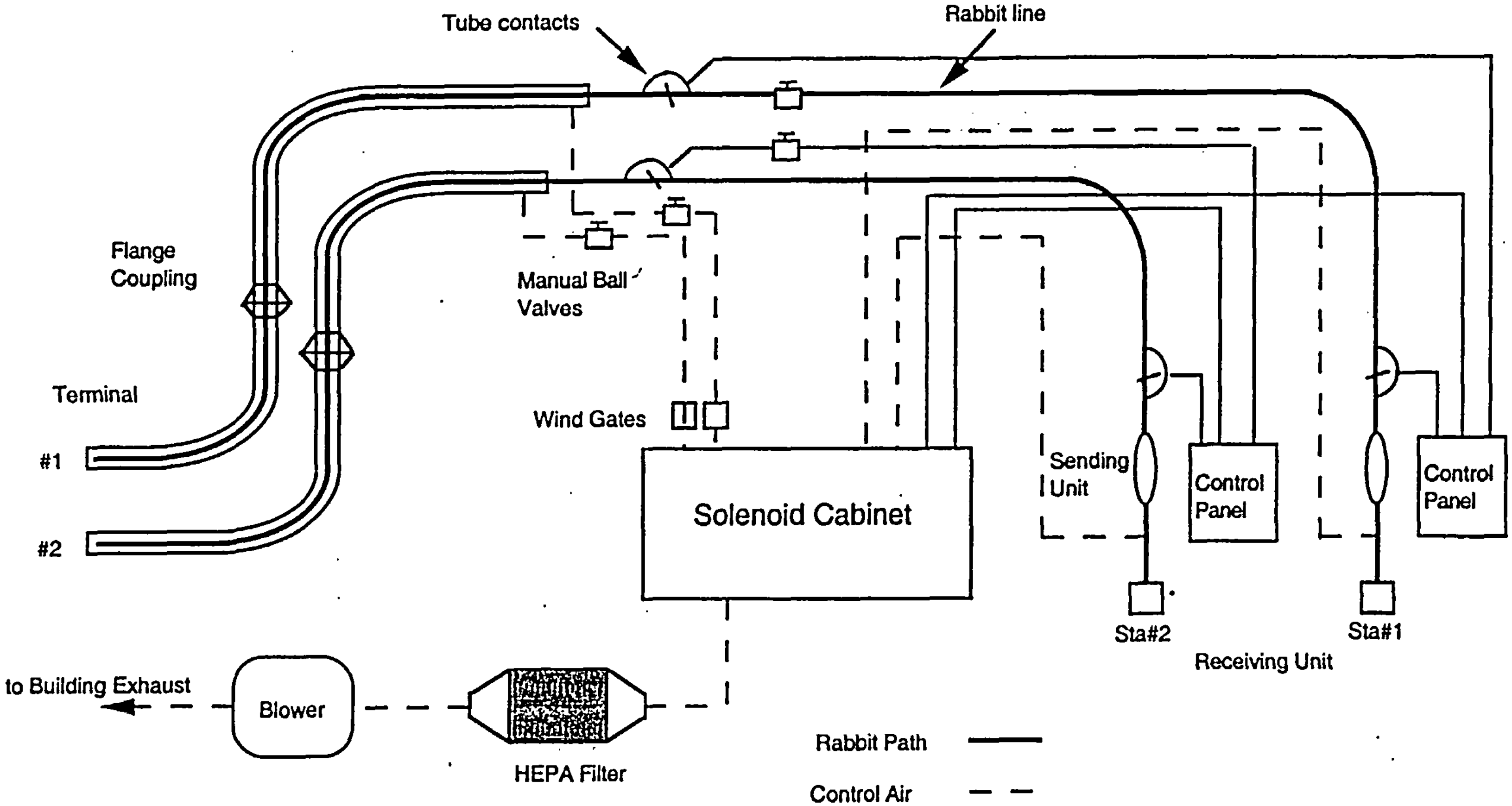

Figure 10-6

1

CHAPTER ELEVEN

TABLE OF CONTENTS

Chapter 11

11.0 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

This chapter discusses and analyzes all radiological consequences related to normal operation of the reactor. The principal discussions of the facility program to control radiation and expected exposures due to operation, maintenance, and use of the reactor are included. This chapter outlines the methods for quantitative assessment of radiation doses in the restricted and unrestricted areas, application of these methods to all applicable radiation sources related to the full range of operation, and the program and provisions for protecting the health and safety of the public (including workers) and the environment.

11.1 Radiation Protection

The purpose of the RINSC Radiation Protection Program (RPP) is to regulate the activities of, and protect the health and safety of, the RINSC staff, the research associates, the students, the general public, and the environment. In accordance with 10 CFR 20.1101, this program has been developed, documented, and implemented to a level commensurate with the scope and extent of licensed activities at the RINSC, and is sufficient to ensure compliance with regulations in 10 CFR 20 as well as the State of Rhode Island Rules and Regulations. A primary component of this program is the fundamental principle of maintaining individual exposures and radioactive effluents as low as is reasonably achievable (ALARA). Responsibility for maintaining the RINSC ALARA Program extends to all individuals who are granted unescorted access to the reactor facility.

All personnel using radioactive materials or radiation sources shall become familiar with the requirements of the RINSC RPP and shall conduct their operations in accordance with said program. However, the Health Physics Staff has the authority to interdict or terminate the use of radioactive materials or radiation sources if adequate health physics support is not available or if significant deviations from established procedures have occurred or are likely to occur.

11.1.1 Radiation Sources

The radiation sources present at the RINSC can be categorized as airborne, solid, or liquid. While each of these categories will be discussed individually in Sections 11.1.1.1 through 11.1.1.3, the major contributors to each category can be summarized as follows: Airborne sources consist mainly of Argon-41 (Ar-41, half-life 1.8 hrs), due largely to neutron activation of air in reactor beam ports and experiment sample positions along with air dissolved in the reactor's primary coolant, and Nitrogen-16 (N-16, half-life 7.1 sec.), due to neutron interactions with oxygen in the primary coolant. Solid sources for the most part are very typical of a research reactor facility. Such sources include the fuel in use, spent fuel, and fresh un-irradiated fuel. In addition, other solid sources are present such as the neutron startup source, small fission chambers for use with nuclear instrumentation, irradiated samples, irradiated components subjected to neutron radiography, other items irradiated as part of normal reactor use, and small instrument check and calibration sources.

11.1.1.1 Airborne

During normal operation of the RINSC reactor, there are two sources of airborne radioactivity, namely Argon-41 and Nitrogen-16.

- Argon-41 produced by activation of natural argon (a normal trace element in atmospheric air). (Argon-41 decays to Potassium-41 with a half-life of 100 minutes by the emission of a 1.2-MeV beta particle and a 1.3-MeV gamma ray.)
- Nitrogen-16 produced by the activation of Oxygen-16. (Nitrogen-16 decays to Oxygen-16 by beta decay accompanied by the emission of 6-7 MeV gamma rays with a half-life of approximately seven seconds).

The assumptions and calculations used to assess the production and radiological impact of these airborne sources during normal operations are detailed in Appendix A. Therefore, that information will only be summarized in this section.

Argon-41 is not a serious problem. Comply Code calculations show a projected dose equivalent of 0.021 mrem per curie of Argon-41 released. Using the 10 mrem per annum limitation of projected dose equivalent to the maximally exposed individual, the facility can release up to approximately 476 curies of Argon-41. Historic generation rates of Argon-41 show 0.14± 0.03 Curies per MW-hour permitting operation up to approximately 3400 MW-hours or 1700 hours of operation at full licensed power.

Although producing very energetic gamma photons, Nitrogen-16 is not a serious problem. At low power levels during convection cooling, production is limited and diffusion through twenty-three feet of water allows time for decay. During forced convection cooling, the reactor coolant pumps circulate water at about 1950 gpm [redacted] core, the water passes through a delay tank (allowing the Nitrogen-16 to decay for at least 90 seconds) and a heat exchanger and re-enters the core. The circulation is designed so that water re-entering the core is essentially Nitrogen-16 free.

Fuel element failure, although not expected, could occur while the reactor is operating normally. Such a failure would usually occur due to a manufacturing defect or corrosion of the cladding and would result in a small penetration of the cladding through which fission products would be slowly released into the reactor coolant. Some of these fission products, primarily the noble gases, would migrate from the cooling water into the air of the reactor room. Although this type of failure could occur during normal operation, its occurrence is <u>not</u> normal and no normal operation would take place after such an event until the situation had been eliminated (i.e., the failed element located and removed from the core.) As a result, the failure of a single element (for any reason) is evaluated as an abnormal situation or an accident, and is discussed in Chapter 13

11.1.1.2 Solid

Because the actual inventory of reactor fuel and other radioactive sources continuously changes as part of the normal operation, the information in this section should be considered representative rather than an exact inventory.

The principal solid radioactive sources associated with the RINSC reactor operation are the mixed fission products produced and retained within the fuel during the normal operation of the reactor. At shutdown, following a seven-hour run at 2 MW, the fission product inventory is approximately [redacted] curies. Following an hour of decay, the remaining fission product inventory is approximately [redacted] curies. Following sixteen hours of decay, the remaining fission product inventory is approximately [redacted] curies.

Other solid radioactive sources are summarized in the following table.

Source Description	Radionuclides	Nominal Activity (Ci)	Physical Characteristics	wt% Uranium	Approximate Original Total Grams	
					U-235	Total U
14 Fuel Elements	U-235, U-238 and their natural decay products.	[redacted] (excluding the activity from the mixed fission products)	Uranium silicide-aluminum dispersion fuel elements containing 22 Standard plates per fuel element	Less than 20	[redacted]	[redacted]
# Fission Chambers						
Reactor Startup Source	Sb-125	[redacted]	Aluminum-encapsulated Antimony-beryllium	NA	NA	NA
Instrument Calibration Sources	A variety of sealed sources are possessed under Rhode Island Department of Health License Number 3K-063-01. The sources are used for survey and other instrument calibrations.					

Historically, solid waste has been limited in volume and activity. RINSC routinely produces low-level dry active waste including paper towels, plastic bags, rubber gloves, and other materials used for contamination control or decontamination. The activity of this material is normally in the microcurie range, and historically about one to two regular 55-gallon drums of this material are generated each year.

11.1.1.3 Liquid

No liquid radioactive material is routinely produced by the normal operation of the RINSC except for miscellaneous neutron activation product impurities in the reactor coolant, most of which is deposited in the mechanical filter and the demineralizer resins. Therefore, these materials are dealt with as solid waste (Section 11.1.1.2). Non-routine liquid radioactive waste could result from decontamination or maintenance activities (i.e., filter or resin changes). The amount of this type of liquid waste is expected to remain small, especially based on past experience.

11.1.1.3.1 Radioactivity in the Reactor Coolant

As mentioned above, the only significant liquid radioactive source at the RINSC is the reactor coolant. Radioactivity in this liquid source occurs due to neutron activation of Argon-40 in entrained air (creating Ar-41); neutron interactions with oxygen in the water molecule (creating N-16); and neutron interactions with tank and structural components with subsequent transfer of the radioactivity into the primary coolant. Radionuclides such as Manganese-56 and Sodium-24 are common examples of waterborne radioactivity created in this manner. Tritium is also present in the coolant due to activation of D_2O and other mechanisms.

As noted, other sources of liquid radioactivity are not currently projected for the RINSC reactor system, and no radioactive liquid effluents and no liquid wastes have been generated as part of the current 2 MW operation.

Radionuclides and their concentrations in the primary coolant vary depending on reactor power, reactor operating time and time since reactor shutdown, assuming that other variables (e.g., the effectiveness of the water purification system) remain constant. Historically, measured concentrations of radionuclides in the primary coolant have remained very low. The principal radionuclide routinely noted in the primary coolant is sodium-24 (with a half-life of 14.96 hours) at a concentration of approximately [redacted] μCi/ml. In addition, small quantities of Tritium are produced [redacted]μCi/ml).

As mentioned, it is RINSC policy not to release liquid radioactivity as an effluent or as liquid waste. Therefore, the reactor coolant does not represent a source of exposure to the general public during normal operations. Furthermore, occupational exposure from liquid sources is also limited because there are few operations that require contact with the primary coolant. In cases where contact is a potential, such as in certain maintenance operations, the reactor coolant is allowed to decay for several days or more to significantly reduce radioactivity concentrations. Because of the short half-life of the predominant radionuclide in the primary coolant, after 48 hours, sodium-24 would be reduced by about a factor of 10, and experience at other research reactors has shown that Tritium is not a source of significant occupational dose.

11.1.3.2 N-16 Radiation Dose Rates from Reactor Cooling System Components

N-16 has been addressed previously in Section 11.1.3.1. However, the potential for N-16 radiation dose rates from primary coolant system components and from the heat exchanger were not included in that discussion. Measurements of gamma dose rates at contact with those cooling system components after extended operation at 2 MW show typical contact dose rates in the range of 5 to 6 rem/hour at the Delay Tank, 1 to 2 rem/hr at the heat exchanger, and less than 1 rem/hr at the secondary pumps. These radiation levels are not considered abnormal and do not represent a radiation protection problem since they are expected and they occur inside the posted radiation area in the reactor building.

11.1.2 Radiation Protection Program

The rules, instructions and procedures for procurement, disposal and safe handling of radionuclides are contained in the Radiation Safety Guide.

11.1.2.1 Radiological Safety Organization

The Radiological Safety Organization at the RINSC is comprised of the following:

- Rhode Island Atomic Energy Commission (RIAEC),
- Nuclear and Radiation Safety Committee (NRSC),
- Nuclear and Radiation Safety Subcommittee (NRSSC),
- Radiation Safety Officer (RSO),
- Radiation Safety Staff,
- Health Physicist, and
- Health Physics Technician/Reactor Operator (Health Physics Specialty).

The RIAEC is an agency of the government of the State of Rhode Island. The RIAEC operates the RINSC. The legal basis of the RIAEC is Title 42, Chapter 27, General Laws and Chapter 142, Laws of 1958.

The RIAEC consists of five (5) individuals appointed by the Governor, one of whom he/she appoints as Chairperson. The RIAEC employs a Director who is responsible for implementing and coordinating all decisions of the Commission. For the operation of the RINSC, the Commission employs an Assistant Director for Reactor Operations who is responsible for the day-by-day operations of the facility, including the use of radioactive materials.

11.1.2.2 Nuclear and Radiation Safety Committee and Subcommittee

The Nuclear and Radiation Safety Committee is appointed by and reports to the Rhode Island Atomic Energy Commission (RIAEC). The Nuclear and Radiation Safety Committee (NRSC) has the following membership, functions, method of convening, and reporting duties.

The NRSC is appointed by the Rhode Island Atomic Energy Commission and has membership as follows:

- Director, RIAEC;
- Assistant Director for Reactor Operations;
- Assistant Director for Radiation and Reactor Safety (Radiation Safety Officer);
- A representative from the University of Rhode Island;
- A representative from Brown University;
- A representative from Providence College; and,
- A representative from the Rhode Island academic community at large.

Members serve until replaced. The NRSC and RIAEC review representation every 5 years. The NRSC may request funds from the RIAEC to employ consultants as required to assist in its determinations.

The function of the NRSC is to insure compliance with all federal, state, and RINSC regulations as follows:

1. Oversight and review of reactor operations.
2. Review of any tests, experiments or modifications involving the reactor under 10 CFR 50.59.
3. Administering a broad RINSC radioactive material license, source and special nuclear material licenses, and specific licenses.
4. Considering in advance and approving (or disapproving) the production, procurement, use, and ultimate disposal of radioactive materials at the RINSC from the standpoint of radiological and nuclear safety.
5. Prescribing special conditions and regulations such as physical examinations, additional training, designation of location of use, disposal methods, accountability, etc. as necessary for use and control of radioisotopes.
6. Reviewing the qualifications of persons applying for permission to use radioactive materials and approving or denying applications for experiments.
7. Overseeing the maintenance of records of actions, approving or disapproving the use of radioisotopes and of other transactions, communications, and reports.
8. Receiving reports from the Radiation Safety Officer (RSO) and auditing RSO records and performance.
9. Overseeing the maintenance of records, and performance of physical inventories and radiation surveys of radioactive materials at the RINSC.
10. Recommending appropriate action to the RIAEC for an individual who fails to observe safety recommendations, rules, or regulations when using radioisotopes.
11. Conducting an annual review of the entire radiation safety program, including records required to be maintained, possible improvements to the safety program, and changes to the Radiation Safety Guide.

As a general rule the committee grants authorizations only to fully qualified individuals. The committee evaluates the applicant's relevant training and experience prior to granting authorization. When the committee judges an applicant has insufficient training and experience for the proposed use, the applicant may be advised to work under supervision of an authorized user to gain experience. Acceptable training and experience will include, but is not limited to:

1. Principles and practices of radiation protection as they apply to the requested radionuclide, activity, and procedure.
2. Radioactivity measurement techniques and instruments.
3. Mathematics basic to measurements of radioactivity.
4. Radiobiological effects.

The NRSC authorizes all radioactive materials produced or utilized in the experiment programs involving the reactor (e.g., neutron activation analysis, neutron spectroscopy, etc.). All use of radioactive materials in the facility is authorized by the NRSC based on requests submitted by experimenters. The authorizations are for specific isotopes in specific quantities. The NRSC will not issue authorization for materials not contained in the license and will not issue authorization for materials for which the cumulative quantity, both individually and in the aggregate, exceeds the license. All radioactive materials, whether purchased from the outside or locally produced in the reactor, are ordered through the Radiation Safety Officer. All materials received are processed through the RSO. All disposals are also accomplished through the RSO. For radioactive materials produced as part of the neutron activation analysis process, the irradiation record will serve as the inventory record. For radioactive materials produced as part of the neutron spectroscopy process, the experiment approval will serve as the inventory record. For all other radioactive materials, the RSO maintains a decay-corrected inventory broken down by users. At its quarterly meeting, the NRSSC (subcommittee) will review the inventory to verify compliance with license requirements.

The Radiation Safety Office provides an application for possession and use of radioactive materials. Applicants are requested to provide considerable detail because the committee evaluates the qualifications, facilities, proposed uses, and radiation safety measures primarily on the information provided on the form. The application must demonstrate sufficient knowledge of the RINSC policies and procedures and radiation safety practices so as to keep doses ALARA and maintain compliance for the specific radionuclides, activities, sources of radiation, and procedures requested. The Radiation Safety Officer reviews the application and forwards it to each member of the committee. Authorization requires unanimous approval of the voting members. The Radiation Safety Office issues an authorization that may include special conditions imposed by the committee.

Investigators wishing to add a new radionuclide to their authorization complete a new application for possession and use of radioactive material form for committee review. Investigators wishing to increase an authorized limit for a radionuclide complete an application for increase in quantity of radioactive material form. The Radiation Safety Officer is authorized to approve the increase temporarily provided the applicant has demonstrated that any additional radiation safety precautions have been addressed. The Radiation Safety Officer submits applications for increase to the committee for final approval. The Radiation Safety Officer or committee may impose new conditions.

The NRSC has appointed a subcommittee consisting of the Chairperson, the Radiation

Safety Officer, and one other Committee member selected by the committee. The NRSC has delegated to this subcommittee the responsibility for the review of the qualifications of applicants and the review of the radiation hazards associated with a proposed experiment. The subcommittee possesses the authority to approve applications without convening the committee if (a) an application of similar nature has already been approved by the committee, (b) the application falls within approved guidelines set down by the committee, or (c) the radioisotope quantities requested by the applicant are less than or equal to those given in 10 CFR 20, Appendix C and the Rhode Island "Rules and Regulations for the Control of Radiation", Part A, Appendix C. The full committee must review all other proposals and all disapprovals.

The NRSC holds meetings as often as necessary to discharge its responsibilities. The Radiation Safety Officer or Chairperson will call a meeting to process pending applications or review reports. The full committee will meet at least annually. Committee deliberations will follow the Roberts Rules of Order if so requested by a member. The Chairperson of the Committee reports to the Rhode Island Atomic Energy Commission.

When the Committee members are replaced, the technical competence and character of the Nuclear and Radiation Safety Committee and its Subcommittee will remain similar to that submitted.

The NRSC voted unanimously to appoint the Subcommittee to periodically review the detailed records required by the licenses. This review will be accomplished during the quarterly meeting of the Subcommittee. In addition, the Subcommittee reviews the list of authorized experiments and determines the status of each authorized experiment. The Subcommittee transmits the results of this review to the remaining committee members in oral and/or written form. At its annual meeting, the NRSC receives and reviews the reports of the Subcommittee and the RSO. At its discretion, the NRSC will select one or more of the detailed records to compare with the reports presented by the Subcommittee or the RSO. Based on the reports, the NRSC will determine what changes, if any, may be necessary in the safety program.

11.1.2.3 Radiation Safety Officer

The Radiation Safety Officer (RSO) is the chief administrative officer of the radiation safety program. The RSO:

1. Prepares and maintains a Radiation Safety Guide for the Center;
2. Administers the Center's ALARA Program;
3. Establishes and administers a radiation dosimetry program, including a record-keeping system, issuing and processing of personnel dosimeters, notifying individuals of exposures at least annually and more frequently when warranted;
4. Establishes procedures for, and causes to be made, periodic surveys of areas where radioactive materials are used or stored;
5. Submits necessary documents (with the cooperation of affected individuals) for licensing of radioactive materials and registration of radiation-producing devices;
6. Develops and administers a calibration program for survey instruments;
7. Prepares and presents briefings and training sessions for RINSC personnel and others who might use or be exposed to radiation;

8. Maintains an emergency call list providing 24 hours per day coverage in the event of accident or other abnormal occurrence;
9. Reviews and approves procedures and documents with radiation safety implications prior to implementation by the RINSC users;
10. Encourages compliance with procedures outlined in this Safety Analysis Report and with regulations of appropriate government agencies; and
11. Audits, as appropriate, records of possession and use of radioactive materials and radiation-producing devices. Such audits include a review of departmental procedures, manuals, logs, emergency plans, and other documents relevant to compliance with applicable standards.

The Radiation Safety Officer reports to the Director. The Health Physicist and the Health Physics Technician report to the Radiation Safety Officer.

11.1.2.4 Health Physics Staff

The health physics activities at the RINSC are performed by a staff consisting of the RSO, a Health Physicist and a Reactor Operator (Health Physics Specialty). Two of the staff members are shared by the University of Rhode Island's Radiation Protection Program. The Health Physicist and the Health Physics Technician report to the Radiation Safety Officer.

11.1.2.5 Radiation Protection Training

The Radiation Safety Office provides radiation safety training for individuals who work with or around radioactive materials or other generators of ionizing radiation. Authorized Users and workers using radioactive materials must have initial training before using radioactive materials. Annual retraining is required for the continued use of and authorization to possess radioactive materials and/or use radiation-producing equipment. Non-radiation workers (for example, custodial and security personnel) are retrained annually. Ancillary training is tailored to specific job functions.

11.1.2.6 Health Physics Procedures

The Radiation Safety Office describes its health physics activities in standard operating procedures. An administrative procedure describes the development, review and approval steps for those standard operating procedures.

11.1.2.7 Health Physics Audits

Audits of the radiation protection program are performed to verify compliance with applicable federal and state regulations and to determine the effectiveness of the program.

11.1.2.8 Health Physics Records

Health Physics records related to personnel exposures, routine and special surveillance activities, environmental releases, etc. are maintained as permanent facility records. Health Physics records are retained electronically and as hard copies.

11.1.3 ALARA Program

The ALARA program at the Rhode Island Nuclear Science Center consists of the following elements:

- A training program for individuals using radiation sources so that they can recognize and protect themselves for sources of ionizing radiation.

- A comprehensive program of dosimeter services including badge monitoring and bioassays. Using the results of these services, the Radiation Safety Officer will investigate any exposures that are not ALARA.

- A survey program to check each area where radiation sources are used for radiation and contamination levels. Laboratories are checked to ensure proper techniques are used during procedures involving radiation sources. The Radiation Safety Officer also provides advice and assistance to personnel using radiation sources.

- The Radiation Safety Officer reviews effluent releases and investigates releases over 10% of the limits.

- A review process whereby the Radiation Safety Officer and the NRSC must approve the use of radioactive material and audit the Radiation Safety Program.

The RINSC is committed to ALARA. All radiation workers shall use procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and public doses that are ALARA. The radiological safety organization works to achieve that goal.

11.1.4 Radiation Monitoring and Surveying

11.1.4.1 Radiation and Contamination Surveys

A survey is an evaluation of the radiation hazards associated with the presence of radioactive materials and/or radiation sources under a given set of circumstances. The Radiation Safety Office conducts routine radiation and contamination surveys described in standard procedures to evaluate basic radiological conditions at the RINSC.

11.1.4.2 Radiation Monitoring Equipment

Radiation monitoring and survey equipment is available for reactor operation fuel movement and handling of radioactive material in the reactor building. Some are required by Technical Specifications. When the reactor is operating, gaseous and particulate sampling of the stack effluent is monitored by a stack monitor with a remote readout in the control room. A constant air monitoring unit is located in the confinement building. Area monitors are installed throughout the reactor building as part of the reactor system. These instruments are equipped with a remote readout in the control room. Area monitors are also installed in the vicinity of the pneumatic systems to monitor samples coming out of the reactor. Radiation monitors are installed near the door from the reactor room. This monitor is intended for use as a hand and foot monitor and to provide an alarm for the inadvertent removal of a "hot" sample from the reactor room. Other portable survey meters are available for contamination monitoring.

The following table shows representative radiation monitoring equipment used in the RINSC:

Type of Instrument	Radiation Detected	Use
Continuous Air Monitors	Beta-gamma	Particulate, Iodine, Noble Gas monitoring in gaseous effluents
Area Monitors	Gamma	Area dose rate monitoring
Area Monitor	Neutrons	Area dose rate monitoring
Gas Flow Proportional Counter	Alpha, Beta	Activity measurement
Gamma Spectroscopy Analyzer System	Gamma	Activity measurement
Neutron Survey Meter	Neutrons	Dose rate measurement
Radiation/Contamination Monitors	Beta, Gamma	Contamination surveys
Air-equivalent Ionization Chamber Survey Meters	Alpha, Beta & Gamma	Dose rate measurement
Survey Meters equipped with Geiger-Mueller Probes	Alpha, Beta, & Gamma	Dose rate measurement

11.1.4.3 Instrument Calibration

The RINSC calibrates all survey meters used in its routine and special surveillance programs. The RINSC possesses a calibrated [redacted] Cs-137 source, a [redacted] Cs-137 source and a Co-60 source, all are traceable to NIST, capable of calibrating instruments with gamma exposure rates of 100 R/hr to <0.1 mR/hr. At present this range is sufficient for all survey instruments at the RINSC. If for some reason the RINSC is unable to calibrate the instrument, the Radiation Safety Officer will return the meter to its manufacturer or to a reputable company for repairs and/or calibration. Calibration procedures are attached for each source. Results of each instrument calibration are recorded on Form NSC-17. The formulae used for each source is given on the attached sheets. The dose rates are decay corrected to the day of calibration at which time the distances are calculated. Calculations necessary for calibrating instruments in an integrating dose mode require obtaining the correct distance and calculating an appropriate exposure time. Survey meters that are used only for the purpose of detecting contamination will be electronically calibrated and checked with an appropriate source of radiation to determine if the probe is functional.

Technically qualified personnel under the supervision of its Radiation Safety Officer perform all survey meter calibrations at the RINSC. Survey instruments, if required, (i.e., GM meters, ionization survey meters) will be calibrated every 6 months during the time of use of hard beta and gamma emitters. Instruments that are used for dose rate measurements and for detecting contamination will be calibrated at the RINSC. A reference check source should be read at the time the instrument is calibrated. A reading should be taken with the check source placed in a specific geometry relative to the detector. A reading of this reference check source should be taken as follows:

- Before each use and also after each survey to ensure that the instrument was operational during the survey, and

- After each maintenance and/or battery change.

11.1.5 Radiation Exposure Control and Dosimetry

Rules and regulations for protection against radiation hazards have been established by the Rhode Island Department of Health. A copy of these regulations is available from the Radiation Safety Officer. These regulations specify limits for exposure in restricted and in unrestricted areas. These limits will be discussed briefly in the following sections.

11.1.5.1 Radiation Shielding

Shielding design considered the seven principal sources of radiation produced by operation of the RINSC reactor:

- Prompt fission neutrons
- Delayed fission neutrons
- Prompt fission gamma rays
- Fission product decay gamma rays
- Inelastic gamma rays
- Capture gamma rays
- Activation gamma rays

The reactor is shielded to maintain radiation levels from those sources at all points above and outside the pool below one millirem per hour. The reactor is kept beneath twenty-four feet of water. The sides of the core are shielded by at leas[redacted] feet of water in addition to [redacted] feet of concrete or equivalent shielding in the low-power section of the pool. Dense concrete plugs shield the beam ports. Steel doors shield the thermal column. During routine operation, radiation surveys are performed to ensure that radiation levels do not exceed design requirements. If the radiation levels ever exceed design requirements, the reactor would be shut down and/or additional shielding would be provided to reduce the radiation levels to design requirements.

11.1.5.2 Ventilation System

The reactor building ventilation system consists of exhaust blowers which discharge into the exhaust stack. The stack also receives dilution air from a blower in the basement area outside of the reactor confinement building. Air from the pneumatic blower system is discharged into the stack via the off-gas suction line. (See Section 10.4) The off-gas blower removes gases from the thermal column, beam tubes and the pneumatic system and discharges into the suction line of the reactor room exhaust blower. In addition, the reactor room exhaust blower constantly exchanges the air from the reactor confinement building. The exhaust blower inlet plenum is located near the pool platform to essentially sweep air across the pool surface which helps remove airborne activity at the pool level. The discharge goes through the air exhaust duct, through the exhaust system butterfly damper, and into the stack. The stack is a 115-foot steel unit about 20 inches in diameter through which the total confinement building exhaust is released into the environment.

A constant air monitoring system monitors gaseous radioactivity that is discharged from the confinement building. A continuous stream of gas is withdrawn through an isokinetic sampling tube and passed through the monitoring system. The monitoring system consists of a NaI crystal gaseous detector and a plastic scintillator particulate detector. A continuous graphical readout with alarm is located in the process control panel in the control room with remote readouts located in the Emergency Support Center. A constant air monitor detects airborne radioactivity in the reactor room. It is located on the main reactor floor and consists of an open window G-M detector. The unit has an audible alarm and local graphical readout.

11.1.5.3 Confinement

The confinement building (reactor building) is a concrete building normally maintained under a negative pressure. Normal operation results from reactor room exhaust air being removed through the reactor room blower, through a butterfly valve into the reactor exhaust stack. Air intake comes through a butterfly valve into the building heated air duct. The valves close air tight on activation of the emergency exhaust system.

Additionally, off-gas is removed from experimental facilities through an off-gas blower and filter system. The blower removes gases from the thermal column, beam tube drain and vents etc. and discharges into the reactor room exhaust piping before the reactor room discharge butterfly valve. A pneumatic system blower discharges its gas removal into the off-gas discharge line. The off-gas and pneumatic systems and operation are more fully described in Chapters 9 and 10. A dilution air blower provides additional air flow and discharges into the stack at its base. Normal building differential pressure is measured at two locations, one in the control room, and one in the adjacent lab building.

11.1.5.4 Entry Control

Only trained personnel are granted unescorted access to radiologically controlled areas within the facility. Radiation Worker Training provides the training required to obtain unescorted access to the reactor facility. The Radiation Safety Office provides radiation safety training to all individuals who in the course of their employment are likely to receive an occupational dose in excess of 1 mSv (100 mrem) in a year. The training is provided, as part of the facility's As Low As Reasonably Achievable (ALARA) Program and to meet regulatory requirements.

In determining candidates for the training, the Radiation Safety Office considers assigned activities during normal and abnormal situations involving exposure to radiation and/or radioactive material that can reasonably be expected to occur during the life of the licensed facilities. The extent of the training is commensurate with potential radiological health protection problems present in the work place.

The radiation safety training keeps those individuals informed of the storage, transfer, or use of radiation and/or radioactive material. The training also instructs them in the health protection problems associated with exposure to radiation and/or radioactive material, in precautions or procedures to minimize exposure, and in the purposes and functions of protective devices employed. The training also instructs the individuals in, and requires them to observe, to the extent within the workers control, the applicable provisions of the federal and state regulations for the protection of personnel from exposure to radiation and/or radioactive material and the reactor and radioactive materials license conditions. They are instructed of their responsibility to report promptly to RINSC management any condition that may lead to or cause a violation of federal or state regulations and/or the reactor or radioactive materials license or lead to an unnecessary exposure to radiation and/or radioactive material. They are instructed in the appropriate response to warnings made in the event of any unusual occurrence or malfunction that may involve exposure to radiation and/or radioactive material and advised as to the radiation exposure reports which they may request.

11.1.5.5 Protective Equipment

Radiation workers wear lab coats, disposable gloves, and protective eyewear when handling unsealed radioactive materials. Ventilation controls are used to minimize exposures to airborne radioactive materials.

11.1.5.6 Annual Radiation Dose Equivalents

Personnel at the RINSC have been monitored for radiation exposures for nearly forty years. Measured external dose equivalents have consistently met regulatory standards in effect at the time of the dose determination.

External and internal dose equivalent determinations were made separately until changes in the regulatory requirements became effective in 1994. The monitoring program was adjusted in response to those regulatory changes to include an annual determination of the total effective dose equivalent (TEDE).

Since 1994, total effective dose equivalents have shown a log-normal distribution with an average deep dose equivalent less than 100 mrem and maximum committed dose equivalents less than 1,000 mrem.

11.1.6 Contamination Control

Radioactive contamination is controlled at the RINSC by using written procedures for radioactive material handling, by using trained personnel, and by operating a monitoring program designed to detect contamination in a timely manner. The program for routine monitoring to detect and identify fixed and loose contamination is described in Section 11.1.4.1. In addition to this monitoring program, the following items are also part of the program for contamination control at the RINSC:

The RINSC standard operating procedures contain specific procedures for working with radioactive material and for working with experiments that originate from in-pool or from the pneumatic transfer system hood. For other work where contamination is considered likely, a detailed written procedure provides the necessary contamination controls. All work requires oversight by a qualified health physics technician and all material which must be removed from a contaminated area with suspected loose contamination is appropriately monitored and contained to minimize potential spread, or is decontaminated.

After working in contaminated areas, personnel are required to perform surveys to ensure that no contamination is present on clothing, shoes, etc., before leaving the work location. Additionally, personnel exiting controlled areas surrounding a contaminated area are required to frisk themselves using an appropriate survey meter located at an exit. The RINSC personnel are not exposed to sources of radioactivity likely to result in internal exposure.

Anti-contamination (Anti-C) clothing designed to protect personnel against contamination is used as appropriate. Normally, Anti-C clothing is specified in the written procedure governing the work. Anti-C clothing is monitored after each use.

The RINSC standard operating procedures contain procedures for monitoring and handling contaminated equipment and components.

Procedures for classifying contaminated material, equipment and working areas and managing, controlling, storing, and disposing of identified contaminated material are contained in the RINSC Health Physics Procedures.

Staff and users are trained on the risks of contamination and on the techniques for avoiding, limiting, and controlling contamination as specified in the RINSC Health Physics Procedures.

Contamination events are documented in a radiological incident report. These reports help avoid repeating events which caused unplanned contamination. The reports are maintained by the Radiation Safety Office and are retained for the life of the facility.

Encapsulation requirements for items likely to cause contamination during or after irradiation are included in routine irradiation procedures for those items.

11.1.7 Environmental Monitoring

A stack monitor monitors airborne effluents during reactor operation. Limits are established in the Technical Specifications.
The concentration of radioactive materials in the effluent released from the facility exhaust stack shall not exceed the concentration specified in 10 CFR 20, Appendix B, Table II where the plume first touches the ground following its release. A suitable dilution factor is used to calculate the maximum ground concentration down wind for noble gases. Offsite projected doses are calculated annually using generally accepted models and conversion factors. Projected doses from the Comply Code show annual doses consistently less than 10 mrem for operations between 1997 and 2001.

Liquid Effluents - The liquid waste retention tank is batch sampled and the gross activity per unit volume determined before release. The concentrations are within the limits of Appendix B, Table II of Subpart A, "Rhode Island Rules and Regulations for the Control of Radiation."

Environmental Doses - Doses outside the facility are monitored at certain locations using quarterly Optically-Stimulated Luminescent Dosimeters. These doses are within the limits of 10 CFR 20 for the general public taking into account the occupancy factor.

11.2 Radioactive Waste Management

11.2.1 Radioactive Waste Management Program

A waste is a material that is no longer useful. A radioactive waste is a waste containing radioactive materials in quantities sufficient to require control in the judgment of regulatory authorities. The RINSC may generate a variety of gaseous, liquid and solid wastes including wastes that could contain biohazardous, hazardous and/or radioactive materials. Since there are stringent regulatory requirements for wastes containing any of these materials, the RINSC has established a radioactive waste management program to meet those regulatory requirements.

The reactor generates short-lived isotopes. Short-lived isotopes can be decayed, surveyed and their wastes released from restrictions due to their radioactivity. Users are required to maintain adequate records to support disposal. To minimize the generation of possible mixed wastes all experimental studies are reviewed to avoid hazardous chemicals.[1]
In general, the Center's radioactive wastes are collected from the generator, processed and stored in a secure area within the facility and transferred to a licensed broker, processor or burial site operator. Wastes containing radioisotopes with half-lives of less than one hundred and twenty (120) days are held for at least ten (10) half-lives

[1] Chemicals subject to the Resource Conservation and Recovery Act and/or the Toxic Substances Control Act.

(decaying to near background levels as confirmed by surveys), then disposed as other types of wastes after obliterating radioactive markings. Wastes contaminated with hazardous chemicals or infectious agents and short-lived radioactive materials may be disposed as hazardous or infectious waste after decaying to background levels.

11.2.2 Radioactive Waste Controls

Operations are routinely reviewed to minimize radioactive waste generation. We separate our radioactive wastes by radioisotope and type. Separation allows proper handling and helps reduce overall disposal costs. Short-lived radioactive wastes are held for decay, surveyed and released to other waste disposal methods. Long-lived radioactive wastes are retained for transfer to a licensed broker, processor and/or waste disposal site.

11.2.2.1 Solid Waste

The lowest cost disposal option for dry solid waste that is currently available is incineration, which may reduce the volume of the radioactive waste by a factor of up to 100 but is subject to stringent limitations. Isotopes forming volatile compounds in the incinerator, such as tritium and carbon-14, and which would be released to the environment cannot be incinerated in large quantities. High activity tritium and carbon-14 waste cannot be incinerated. Halogenated plastics, (e.g., polyvinyl chloride or PVC and Teflon), form volatile acids when burned which attack and damage the incinerator's lining. Glass and metals form slag. Slag can attack the incinerator's firebrick and cannot be incinerated. Incombustible liquids such as water may not be present in more than incidental amounts. We visually inspect bags of dry active waste to determine if the waste is a candidate for incineration. We remove any bags that fail to meet the incinerator's operator criteria. Candidate materials for incineration are packaged in fiberboard containers acceptable to the incineration waste processing facility. Dry active wastes that fail to qualify for incineration are packaged for super-compaction.

The second option for disposal of long-lived dry active waste is super-compaction. Super-compaction crushes waste to remove essentially all voids and often achieves volume reduction factors of between 3 and 6 depending on the original density of the waste and potential spring-back. Super-compaction does not produce any off gases and may be applied to higher activity wastes. Materials which are essentially incompressible, manufacturer's stock vials, EPA classified hazardous waste, toxic, pyrophoric, explosive, or biohazards must be excluded from waste that is to be super-compacted. Additionally, the waste may not contain more than incidental amounts of any liquids. Super-compaction candidate wastes are packaged in DOT-approved 17H drums, (either 30-gallon or 55-gallon sizes).

11.2.2.2 Liquid Waste

Liquid radioactive waste sources are quite limited at the RINSC and include mainly the reactor primary coolant. Any effluent or liquid waste is discharged in accordance with 10 CFR 20.2003 requirements.

The principal liquid wastes generated at the facility are aqueous liquid wastes. An aqueous liquid waste is a radioactive liquid waste in which the waste materials are either dissolved in water or evenly distributed in a liquid that is mainly composed of water. Standard operating procedures require aqueous liquid wastes to have a pH between 7 and 9 and not be contaminated with toxic, flammable, poisonous or reactive materials. Aqueous liquid wastes are sampled and counted to determine the types and activities of the radionuclides present. If aqueous liquid wastes meet criteria in 10 CFR 20, they are discharged to the sanitary sewer.

11.2.2.3 Gaseous Waste

The principal gaseous waste produced by facility operations is Argon-41. Gaseous effluent concentrations are documented on the Monthly Information Sheets (Form NSC-78) or equivalent. The gaseous effluents, primarily Argon-41, are typically less than 5% of the 10 CFR 20, Appendix B, Table 2, Column 1 effluent limits and result in offsite doses in unrestricted areas less than 10 mrem per annum.

11.2.3 Release of Radioactive Waste

The final disposal method for Low Level Radioactive Waste (LLRW) generated at the RINSC is dependent on the waste form. The following is a list of waste forms and the disposal method employed.

DRY SOLID - LLRW - Two forms of dry solid LLRW materials are generated, namely, radioactive ion exchange resins and laboratory waste materials. Radioactive, wet, reactor-clean-up ion-exchange resins are ambient air dried for two (2) months in an access controlled area prior to final packaging in DOT approved 30-gallon drums. This drying time also doubles as a decay time for sodium-24 (half life = 15 hours), one of the major radioisotopes found in the resin. A member of the health physics staff collects laboratory waste materials from all radioactive use area locations on a weekly basis. These collected wastes, (irradiated plastics, contaminated tools, toweling, etc.) are added to a DOT approved 30 or 55-gallon steel drum. The contents of these drums are compacted as much as possible to reduce void space. After each drum is full, the drums containing gamma emitters are counted using a Ge(Li) detector to determine the isotope contents. The dose rates on the outside of the drums are measured and total drum activities are calculated. In the case of beta emitters (e.g. C-14 and H-3), a separate record of the contents of the drum is maintained.

AQUEOUS LIQUID WASTES - Highly radioactive aqueous liquid wastes are absorbed on materials (such as speedi-dri) according to the vendor's specific instructions and the requirements of Chapter A.4 of the Rhode Island State Rules and Regulations. The absorption process is accomplished under the direct supervision of the health physics staff. The drums are packaged according to Department of Transportation regulations and the appropriate waste manifests are completed. Low activity aqueous liquid waste is sink-disposed to the regional sewer at concentrations below the concentration limits for discharge to an unrestricted area (Appendix A, Table 1, Column 2).

ORGANIC LIQUID WASTES - Liquid scintillation vials that contain C-14 and/or H-3 in concentrations less than 0.05 μCi/gm are packaged in LSV Deregulated Drums. All other liquid scintillation vials are packaged in LSV Regulated Drums according to the vendor's specifications.

SHORT LIVED RADIOISOTOPES (HALF LIFE ≤ 120 DAYS) - Samples from short irradiations are labeled and stored for ten half-lives until the short-lived isotopes decay away. Samples that are not needed are surveyed with a GM meter and, if readings are background, labels are removed and the samples are disposed in normal trash.

CHAPTER TWELVE

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CHAPTER 12

12.0 CONDUCT OF OPERATIONS

This chapter describes and discusses the Conduct of Operations at the Rhode Island Nuclear Science Center (RINSC). The Conduct of Operations involves the administrative aspects of facility operations, the facility emergency plan, the security plan, the quality assurance plan, the reactor operator selection and re-qualification plan, the startup plan, and environmental reports. This chapter of the Safety Analysis Report (SAR) forms the basis of Section 6 of the Technical Specifications (See Chapter 14).

12.1 Organization

The RINSC Director reports directly to the Rhode Island Atomic Energy Commission (RIAEC). The RINSC is organized and administratively controlled as shown in Figure 12.1.

12.1.1 Structure

The organizational structure in Figure 12.1 shows the RINSC licensee as the RIAEC. The RINSC is under the direct control of the Director. The Director reports to the RIAEC for all nuclear safety and licensing issues.

Both the reactor operations, radiation and reactor safety branches report to the RINSC Director. Both the reactor operations supervisor and the radiation and reactor safety supervisor can go directly to the Nuclear and Radiation Safety Committee (NRSC) with nuclear or radiation safety concerns if they cannot resolve the issue with the RINSC Director.

The RINSC license to operate is issued by the United States Nuclear Regulatory Commission (USNRC). Licensing and reporting information goes from the RINSC Director through the RIAEC to the USNRC.

The RINSC Director has a Nuclear and Radiation Safety Committee (NRSC) that meets at least annually. This committee performs the review and audit of nuclear operations for the RINSC Director. The committee issues an annual audit report to the RINSC Director concerning the regulatory compliance and operation of the RINSC.

12.1.2 Responsibility

RINSC Director

a. The Director shall have responsibility for all activities in the reactor facility which may affect reactor operations or involve radiation hazards, including controlling the admission of personnel to the building. This responsibility shall encompass administrative control of all experiments being performed in the facility including those of outside agencies.

b. It shall be the responsibility of the Director to insure that all proposed experiments, design modifications, or changes in operating and emergency procedures are performed in accordance with the license. Where uncertainty exists, the Director shall refer the decision to the NRSC.

Senior Reactor Operators

a. A licensed senior reactor operator pursuant to 10CFR55 shall be assigned each shift and be responsible for all activities during his shift which may affect reactor operation or involve radiation hazards. The reactor operators on duty shall be responsible directly to the senior operator.

b. The identity of and method for rapidly contacting the on-call senior reactor operator shall be known to the reactor operator on duty. The on-call senior reactor operator must be capable of being contacted by the duty reactor operator within ten minutes. The senior reactor operator shall be present at the facility during initial startup and approach to power, recovery from an unplanned or unscheduled shutdown or significant reduction in power, and refueling. The name of the person serving as senior reactor operator as well as the time he/she assumes the duty shall be entered in the reactor log. When the senior operator is relieved, he/she shall turn the operation duties over to another licensed senior reactor operator.

In such instances, the change of duty shall be logged and shall be definite, clear, and explicit. The senior reactor operator being relieved of his duty shall insure that all pertinent information is logged. The senior reactor operator assuming duty shall check the log for information or instructions.

Reactor Operators

a. The responsible senior reactor operator shall pursuant to 10CFR55 designate for his shift a licensed operator (hereafter called "operator") who shall have primary responsibility under the senior reactor operator for the operation of the reactor and all associated control and safety devices, the proper functioning of which is essential to the safety of the reactor or personnel in the facility. The operator shall be responsible directly to the senior reactor operator.

b. Only one operator shall have the above duty at any given time. Each operator shall enter in the reactor log the date and time he/she assumed duty.

c. When operations are performed which may affect core reactivity, a licensed operator shall be stationed in the control room. When it is necessary for him/her to leave the control room during such an operation, he/she shall turn the reactor and the reactor controls over to a designated relief, who shall also be a licensed operator. In such instances, the change of duty shall be definite, clear, and explicit. The relief shall acknowledge his entry on duty by proper notation in the reactor log.

d. The operator, under the senior reactor operator on duty, shall be responsible for the operation of the reactor according to the approved operating procedures.

e. The operator shall be authorized at any time to reduce the power of the reactor or to scram the reactor without reference to higher authority, when in his judgment such action appears advisable or necessary for the safety of the reactor, related equipment, or personnel. Any person working on the reactor bridge shall be similarly authorized to scram the reactor by pressing a scram button located on the bridge.

Radiation Safety Officer

The Radiation Safety Officer shall be responsible for assuring that adequate radiation monitoring and control are in effect to prevent undue exposure of individuals to radiation.

12.1.3 Staffing

1. A list of reactor facility personnel by name and number is available in the reactor control room for use by the Reactor Operator whenever needed. The call list shall include:

    a. Management personnel;

    b. Radiation Safety personnel; and

    c. Reactor Operations personnel.

2. Reactor operator trainees shall be permitted to manipulate the controls of the reactor under the direct supervision of Licensed Reactor Operators.

3. The following staffing requirements shall be satisfied as a part of reactor startup, operation and shutdown:

    - An operator or senior operator licensed pursuant to 10CFR55 shall be present in the control room unless the reactor is secured;
    - A second person in the facility that can perform prescribed instructions;
    - A Senior Reactor Operator shall be readily available. The available Senior Reactor Operator must be capable of being contacted by the duty Reactor Operator within 10 minutes; and
    - A Senior Reactor Operator shall be present at the facility during initial startup and approach to power, recovery from an unplanned or unscheduled shutdown or significant reduction in power, and refueling.

12.1.4 Selection and Training of Personnel

The RINSC Selection and Training Program contains the detailed information concerning the selection, training, licensing and re-qualification of reactor personnel. This plan addresses the qualifications, initial training, licensee responsibilities, and re-qualification of RINSC reactor operations personnel.

The RINSC training program complies with ANSI/ANS 15.4, Selection and training of personnel for Research Reactors. The program's objective is to train, qualify, and re-qualify individuals for operation and maintenance of the reactor. The content of the training program covers the as-built and existing facility, significant facility modifications, current procedures, and administrative rules and regulations.

In addition to actual personnel training, Reactor Operators and Senior Reactor Operators are required to meet specific medical qualifications. The physical condition and the general health of RINSC reactor operations personnel shall be such that they are capable of properly operating under normal, abnormal and emergency conditions. The primary responsibility for assuring that medically qualified personnel are on duty rests with the RINSC Director.

In addition to the selection and training of reactor operations personnel, the RINSC provides formal training for all facility personnel in radiation protection topics, in items required by 10CFR19, in the As Low as Reasonably Achievable (ALARA) concept and in other related areas.

12.1.5 Radiation Safety

The purpose of the Radiation Safety Program is to allow the maximum beneficial use of radiation sources with minimum radiation exposure to personnel. Requirements and procedures set forth in this program are designed to meet the following fundamental principles of radiation protection:

- Justification – No practice shall be adopted unless its introduction produces a net positive benefit;

- Optimization – All exposures shall be kept as low as reasonably achievable, economic and social factors being taken into account; and

- Limitation – the dose equivalent to individuals shall not exceed limits established by appropriate state and federal agencies. These limits shall include, but not be limited to, those set forth in the Code of Federal Regulations (CFR).

All personnel using radiation sources shall become familiar with the requirements of the Radiation Safety Program and conduct their operations in accordance with them.

The Radiation Safety Program uses ANSI/ANS 15.11, Radiation Protection at research reactors, as a guide.

The details of the Radiation Safety Program can be found in Chapter 11.

12.2 Review and Audit Activities

General Policy: It is the policy that nuclear facilities shall be designed, constructed, operated, and maintained in such a manner that facility personnel, the general public, and property are not exposed to undue risk. These activities shall be conducted in accordance with applicable government regulatory requirements.

The Rhode Island Atomic Energy Commission (RIAEC) as the facility licensee has ultimate responsibility for assuring that the above policy is followed. The Nuclear and Radiation Safety Committee (NRSC) has been chartered to assist in meeting this responsibility by providing timely, objective, and independent reviews, audits, recommendations and approvals on matters affecting nuclear safety. The NRSC is established in accordance with the guidance of ANSI/ANS 15.1; The Development of Technical Specifications for research reactors. The following describes the procedures that govern the composition and conduct of the NRSC.

12.2.1 Composition and Qualifications

The RIAEC shall appoint a Nuclear and Radiation Safety Committee (NRSC) consisting of a minimum of seven members as follows:

a. The Director

b. The Assistant Director for Reactor Operations

c. The Radiation Safety Officer

d. A qualified representative from the faculty of Brown University

e. A qualified representative from the faculty of Providence College

f. A representative from the University of Rhode Island.

g. A representative from the Rhode Island academic community at large

A qualified alternate may serve in lieu of one of the above. The Director, Assistant Director and Radiation Safety Officer are not eligible for Chairpersonship of the Committee.

12.2.2 Direction and Rules

1. The NRSC shall review reactor operations to assure that the facility is operated in a manner consistent with public safety and within the terms of the facility license.

2. The NRSC shall have written direction defining such matters as the authority of the Committee, the subjects within its purview, and other such administrative provisions as are required for effective functioning of the committee. Minutes of all meetings of the Committee shall be kept. All minutes of the previous Reactor Utilization Committee shall be retained for the life of the facility.

3. A quorum of the NRSC shall consist of not less than four (4) members and shall include the Radiation Safety Officer or designee, the Director or the Assistant Director for Operations and the Chairperson or designee.

4. The NRSC shall meet at least annually.

12.2.3 Review Function

The responsibilities of the NRSC shall include but are not limited to the following:

1. Review of proposed tests and experiments utilizing the reactor facilities.

2. Review of proposed changes to the facility systems or equipment, procedures, and operations.

3. Determination of whether a proposed change, test, or experiment would constitute an unreviewed safety question or which may require a change to the Technical Specifications or facility license.

4. Review of all violations of the Technical Specifications and Nuclear Regulatory Commission Regulations, and significant violations of internal rules or procedures, with recommendations for corrective action to prevent recurrence.

5. Review of the qualifications and competency of the operating organization to assure retention of staff quality.

6. Review changes to the NRSC charter.

7. Review, at least annually, the radiation safety aspects of the facility.

12.2.4 Audit Function

The responsibilities of the NRSC shall include but are not limited to the following:

1. Audit of operating and emergency procedures and records.

2. Audit of proposed tests and experiments utilizing the reactor facilities.

3. Audit of proposed changes to the facility systems or equipment, procedures, and operations.

12.3 Procedures

Written procedures, reviewed and approved by the NRSC, shall be used for items 1-9 listed below. The procedures shall be adequate to assure the safe operation of the reactor, but should not preclude the use of independent judgment and action should the situation require such.

1. Startup, operation and shutdown of the reactor;

2. Installation and removal of fuel elements, control blades and incore devices where necessary;

3. Maintenance procedures that could have an effect on reactor safety;

4. Periodic surveillance of reactor instrumentation and safety systems, area monitors, and continuous air monitors;

5. Implementation of the physical Security Plan and Emergency Plan;

6. Radiation control procedures;

7. Receipt, inspection, and storage of new fuel elements;

8. Storage and shipment of irradiated fuel elements; and

9. Experiment review on a case-by-case basis assuring that section 3.8.3(2) of ANSI/ANS 15.1 is satisfied. Operational approval shall be by written approval by a licensed senior operator. Written procedures should be established and supervision of the installation of such experiments shall be defined and exercised.

Substantive changes to the above procedures shall be made only with the approval of the NRSC. A Senior Operator may make temporary changes to the procedures that do not change their original intent. Temporary changes to procedures shall be documented and subsequently reviewed by the NRSC Committee.

12.4 Required Actions

12.4.1 Reportable Occurrence

In the event of a reportable occurrence:

1. The Senior Reactor Operator shall be notified promptly and corrective action shall be taken immediately to place the facility in a safe condition until the cause of the reportable occurrence is determined and corrected.

2. The Director shall report the occurrence to the NRSC and RIAEC. The report shall include an analysis of the cause of the occurrence, corrective actions taken,

and recommendations for appropriate action to prevent or reduce the probability of a repetition of the occurrence.

3. The NRSC shall review the report and the corrective actions taken.

4. Notification shall be made to the NRC in accordance with Section 12.5.

12.4.2 Exceeding Safety Limit

In the event a Safety Limit has been exceeded:

1. The reactor will be shut down and reactor operations will not be resumed until authorization is obtained from the NRC.

2. Immediate notification shall be made to the NRC in accordance with Section 12.5 and to the Director.

3. A prompt report shall be prepared by the Senior Reactor Operator. The report shall include a complete analysis of the causes of the event and the extent of possible damage together with recommendations to prevent or reduce the probability of recurrence. This report shall be submitted to the NRSC for review and appropriate action, and a suitable similar report shall be submitted to the NRC in accordance with Section 12.5 in support of a request for authorization for resumption of operations.

12.5 Reports

Reports include the following:

1. Within 24 hours, a report by telephone through the NRC Operations Center, Washington, DC, the NRC Region 1 and the RIAEC:

   a. Any accidental release of radioactivity to unrestricted areas above permissible limits, whether or not the release resulted in property damage, personal injury or exposure.

   b. Any significant variation of measured values from a corresponding predicted or previously measured value of safety related operating characteristics occurring during operation of the reactor.

   c. Any reportable occurrences as follows:

      i. A safety system setting less conservative than the limiting setting established in the Technical Specifications;

      ii. Operation in violation of a limiting condition for operation established in the Technical Specifications;

iii. A safety system component malfunction or other component or system malfunction which could, or threaten to, render the safety system incapable of performing its intended safety functions;

iv. Release of fission products from a failed fuel element;

v. An uncontrolled or unplanned release of radioactive material which results in concentrations of radioactive materials inside or outside the restricted area in excess of the limits specified in Appendix B of 10CFR20;

vi. An uncontrolled or unanticipated change in reactivity in excess of 0.5 $\%\Delta K/K$;

vii. Conditions arising from natural or man-made events that affect or threaten to affect the safe operation of the facility; and

viii. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or threatens to cause the existence or development of an unsafe condition in connection with the operation of the facility.

d. Any violation of a Safety Limit.

e. Discovery of any substantial variance from performance specifications contained in the technical specifications and safety analysis.

2. The written report shall be sent within 14 days. The report shall:

   a. Describe, analyze, and evaluate safety implications;

   b. Outline the measures taken to assure that the cause of the condition is determined;

   c. Indicate the corrective action taken, including any changes made to the procedures and to the quality assurance program, to prevent repetition of the occurrence and of similar occurrences involving similar components or systems; and

   d. Evaluate the safety implication of the incident in light of the cumulative experience obtained from the record of previous failure and malfunctions of similar systems and components.

3. Unusual Events. A written report shall be forwarded within thirty (30) days in the event of:

   a. Discovery of any substantial errors in the transient or accident analysis or in the methods used for such analyses as described in the safety analysis or in the bases for the technical specifications;

b. Discovery of any condition involving a possible single failure which, for a system designed against assumed failure, could result in a loss of the capability of the system to perform its safety function; and

c. Permanent changes in the facility organization involving the Director or Assistant Directors.

4. An annual report shall be submitted in writing within 60 days following the 30th of June of each year. The report shall include the following information:

a. Tabulation showing the energy generated by the reactor (in megawatt days), the number of hours the reactor was critical, and the cumulative total energy output since initial criticality.

b. The number of emergency shutdowns and inadvertent scrams, including the reasons.

c. Discussion of the major maintenance operations performed during the period, including the effect, if any, on the safe operation of the reactor, and the reasons for any corrective maintenance required.

d. A description of each change to the facility or procedures, tests, and experiments carried out under the conditions of Section 50.59 of 10CFR50 including a summary of the safety evaluation of each.

e. A description of any environmental surveys performed outside the facility.

f. A summary of annual radiation exposures in excess of 500 mrem received by facility personnel, including the dates and times of significant exposures.

g. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge.

12.6 Records

In addition to the requirements of applicable regulations and in no way substituting therefore, records and logs of the following items, as minimum, shall be kept in a manner convenient for review and shall be retained as indicated.

12.6.1 Five Year Records

Records to be retained for a period of at least five years:

a. Reactor operations:

b. Principal maintenance activities;

c. Experiments performed including aspects of the experiments that could affect the safety of reactor operation or have radiological safety implications;

d. Reportable occurrences;

e. Equipment and component surveillance activities;

f. Facility radiation monitoring surveys;

g. Fuel inventories and transfers; and

h. Changes to procedures systems, components, and equipment.

12.6.2 Lifetime Records

Records to be retained for the life of the facility:

a. Gaseous and liquid radioactive effluents released to the environs;

b. Off-site environmental monitoring surveys;

c. Personnel radiation exposures;

d. Updated, "as-built" drawings of the facility; and

e. Minutes of the Nuclear and Radiation Safety Committee (NRSC) (and previous Reactor Utilization Committee) meetings.

12.7 Emergency Planning

The RINSC Emergency Plan contains detailed information concerning the RINSC response to emergency situations. The RINSC Emergency Plan is written to be in accordance with ANSI/ANS 15.16, Emergency Planning for Research Reactors. The information below will give a general overview of the emergency plan.

The RINSC Emergency Plan is designed to provide response capabilities to emergency situations involving the RINSC. The plan deals with the RINSC facility, the spectrum of emergency situations and accident conditions that could arise within the facility, and the associated emergency responses that are required due to the unique nature of the reactor facility. Detailed emergency implementing procedures are referenced in this plan. This approach provides the RINSC facility emergency staff the flexibility to cope with a wide range of emergency situations without requiring frequent revisions to the plan.

The responsibility for the plan rests with the RINSC Director who is also responsible for response to and recovery from emergencies. Implementation of the RINSC Emergency

Plan on a day-to-day basis is the responsibility of the Senior Reactor Operator on duty. Provisions for reviewing, modifying, and approving emergency implementation procedures are defined in the RINSC Emergency Plan to assure that adequate measures to protect the staff and the general public are in effect at all times.

12.8 Security Planning

The RINSC Physical Security Plan contains detailed information concerning the RINSC security measures. The information below will give a general overview of this plan.

The RINSC Physical Security Plan provides the criteria and actions for protecting the facility from acts of intrusion, theft, civil disorder and bomb threats.

Overall responsibility for facility security rests with the RINSC Director, who is responsible for implementation of the plan. Implementation of the security plan on a day-to-day basis during hours of operation is the responsibility of the Senior Reactor Operator (SRO) on duty.

12.9 Quality Assurance

The RINSC Quality Assurance (QA) Program contains detailed information concerning the RINSC QA Program elements and their implementation.

The RINSC QA Program provides criteria for design, construction, operation, and decommissioning of the RINSC reactor facility. The level of QA applied to RINSC reactor activities is consistent with the importance of these activities to safety. The activities included in the UCD/MNRC QA Program are those related to reactor safety and applicable radiation monitoring systems. The specific elements of the RINSC QA Program are the same as those listed in ANSI/ANS 15.8,Quality Assurance program Requirements for research Reactors.

12.10 Operator Training and Requalification

The RINSC has an established program for selection and training of reactor personnel. This program has been established to train, qualify, and re-qualify individuals for operation and maintenance of the reactor. The content of the training shall cover the as-built and existing facility, significant facility modifications, current procedures, and administrative rules and regulations as delineated in ANSI/ANS 15.4, Selection and Training of personnel for research reactors.

The program shall carry the trainee through documented stages of classroom and on-the-job training. The intended results shall be a candidate who anticipates conditions, who communicates well and who can accomplish required tasks during normal and abnormal operational situations. Licensing of a candidate is achieved after successful completion of the training and subsequent testing by the NRC.

The objective of the re-qualification program is to refresh reactor operator's knowledge in areas of infrequent operation, to review facility and procedural changes, to address subject matter not reinforced by direct use, and to improve performance weaknesses. The program shall be designed to evaluate an operator's knowledge and proficiency for his duties. The program shall take into account the specialized nature and mode of operation of the RINSC reactor, and the background, skill, degree of responsibility, and participation of RINSC reactor operations personnel in activities related to reactor operations.

12.11 Startup Plan

The RINSC has been in operation since initial criticality was achieved in 1964. Conversion to an LEU core was accomplished in 1993-94. For any activities that might require a written startup plan, it will be supplied under a separate cover.

12.12 Environmental Reports

The RINSC has retained the services of an outside contractor to produce an accurate un-biased environmental report. The report details a historic review of data and records that document the fact that no adverse effects to the surrounding environment have been caused by the operation of the facility. This report is on file and available for review.

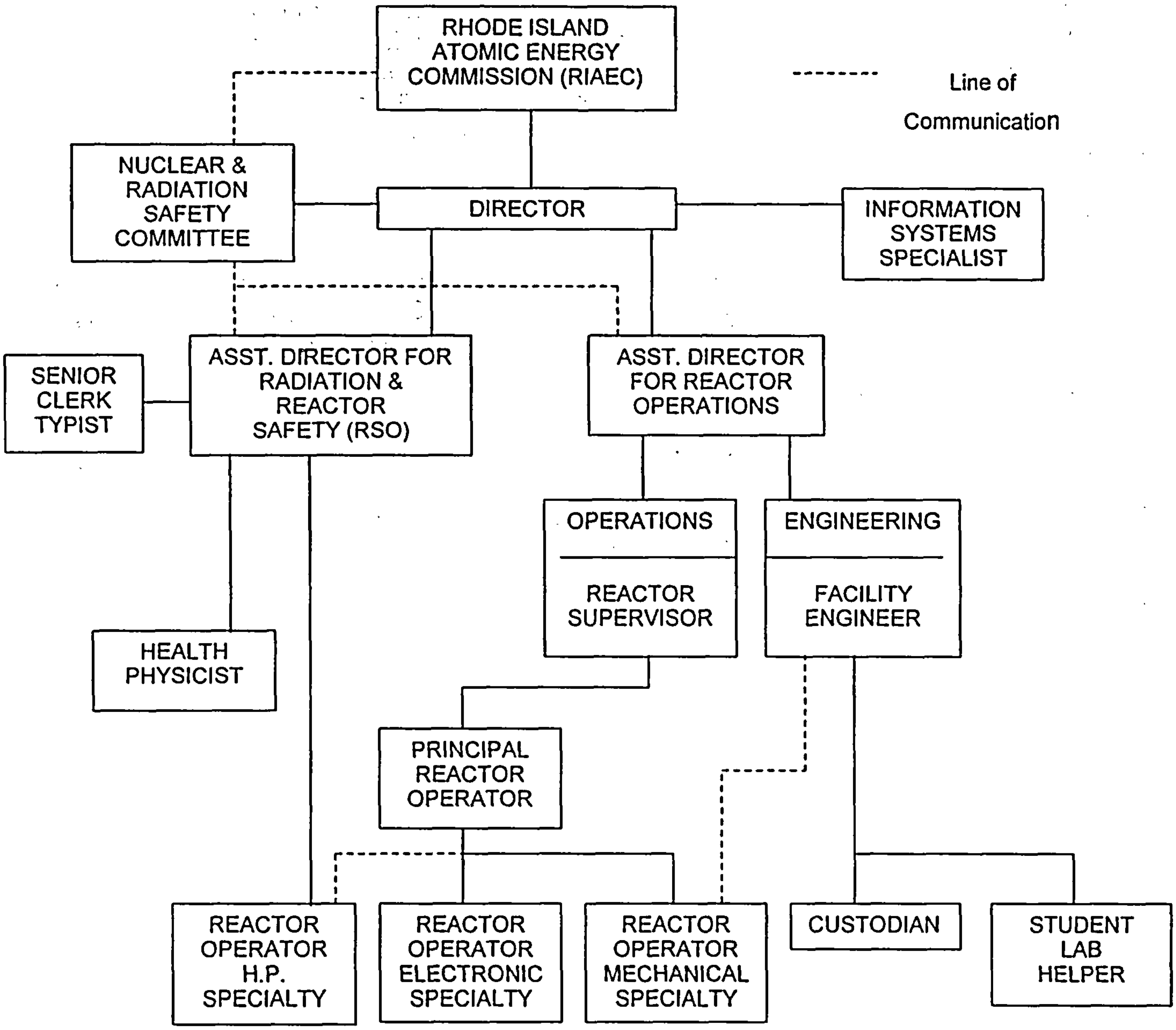
RHODE ISLAND ATOMIC ENERGY COMMISSION (RIAEC)
Line of Communication
NUCLEAR & RADIATION SAFETY COMMITTEE
DIRECTOR
INFORMATION SYSTEMS SPECIALIST
SENIOR CLERK TYPIST
ASST. DIRECTOR FOR RADIATION & REACTOR SAFETY (RSO)
ASST. DIRECTOR FOR REACTOR OPERATIONS
OPERATIONS
REACTOR SUPERVISOR
ENGINEERING
FACILITY ENGINEER
HEALTH PHYSICIST
PRINCIPAL REACTOR OPERATOR
REACTOR OPERATOR H.P. SPECIALTY
REACTOR OPERATOR ELECTRONIC SPECIALTY
REACTOR OPERATOR MECHANICAL SPECIALTY
CUSTODIAN
STUDENT LAB HELPER

Figure 12-1

CHAPTER THIRTEEN

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13.0 ACCIDENT ANALYSIS

13.1 Introduction

Nine credible accidents for research reactors were identified in NUREG-1537 (Reference 13.1) as follows:

- the maximum hypothetical accident (MHA);
- insertion of excess reactivity;
- loss of coolant analysis (LOCA);
- loss of coolant flow;
- mishandling or malfunction of fuel;
- experiment malfunction;
- loss of normal electrical power;
- external events; and
- mishandling or malfunctioning of equipment.

For the RINSC, the Design Basis Accident is considered to be a loss of coolant accident with the water draining through a beam port containing no plugs. The LOCA assumes a guillotine severance of the end of a beam port in the pool with water leaving an experimental beam port end and also through a drain. The MHA, is a hypothetical accident that involves the highly improbable release of fission products although no mechanism is identified to produce this release. Other accidents evaluated for the RINSC are:

- insertion of excess reactivity;
- loss of coolant flow;
- start-up accident;
- loss of normal electrical power;
- external events; and
- mishandling or malfunctioning of equipment.

13.2 Accident Initiating Events and Scenarios

13.2.1 Maximum Hypothetical Accident (MHA)

For the RINSC, the Maximum Hypothetical Accident (MHA) is considered to be an accident involving the release of fission products. The probability of an accident occurring that might result in a significant release of fission product from the RINSC core is extremely remote. Multiple barriers are present to minimize the release of fission products to the environment should the fuel become damaged:

- The fuel is composed of uranium-silicide (U_3Si_2) dispersed in an aluminum matrix and clad in aluminum. This design retains most mixed fission products and retards the leakage of halogens and noble gas.
- Fission products escaping the fuel element are delayed and retained by the pool water.

- The emergency exhaust system provides two stages of roughing filters and two stages of high efficiency particulate air (HEPA) filters to remove particulates and a charcoal filter to remove halogens and delay noble gases.

For the plate type fuel utilized in the RINSC core, physical core damage to an element is possible via pitting of plate cladding or inadvertent mechanical shock; for instance, scratching of a plate surface in handling; such damage could lead to the escape of fission products. While the real magnitude of such damage is indeterminate, it is difficult to conceive of any situations or processes within the scope of reactor operations that could result in the release of a large fraction of the core fission product inventory. Experience at the University of Virginia with an earlier generation fuel design supports the conclusion that a release of a large fraction of the core fission product inventory is unlikely.

In a fission gas release from a fission plate at the University of Virginia reactor on May 3, 1968, isotopic concentrations of noble gases were measured in the reactor room, amounting to ~0.1% of the total fission plate inventory. Iodine isotope concentrations were too low to be detected. However, the University of Virginia concluded from instrument readings that the iodine isotopes were released from the pool in amounts less than 10% of the concentrations of noble gases.

In order for the RINSC to evaluate the consequences of fission product releases in the event of some damage to an element occurring, the hypothesis is made that one plate in an element (from the aspect of vulnerability to mechanical damage, an outside plate would be most likely involved) is damaged to such an extent that total cladding integrity is lost and that volatile fission products are completely available for release to the primary coolant. (This situation is difficult to conceive because even if the cladding were completely ineffective, the metallic fuel matrix provides a substantial barrier against diffusion of fission products within the element.)

The consequences of release of gaseous fission products to the primary coolant and from there to the total pool volume from which escape to the confinement air occurs are reviewed below. Release of air from confinement is assumed to occur through the emergency exhaust system.

<u>Off-site Dose Consequences of Release of Radioactive Iodine Isotopes from a Single Fuel Plate</u>

According to Regulatory Guide 1.4 (Reference 13.4), only 25% of the halogens are released in a fuel assembly cladding failure. For purposes of this assessment, the following assumes release of 25% of the fission product radioactive iodine isotopes in the form of elemental iodine that, in the amounts present, would readily dissolve in the primary coolant water and be transferred by diffusion to the pool volume from which they would escape to the confinement air. In the first section of the following calculations, complete mixing of the iodine in the pool volume is assumed, although no correction is applied for radioactive decay during mixing. Operation at 2 megawatts with 14 elements and 22 plates per element is assumed, and iodine activities are assumed present in saturation quantities. The radioiodine production and decay rate for the i^{th} radioactive iodine isotope in the whole core is given by $\lambda_i N_i$ where:

$$\lambda_i N_i = (3.1 \times 10^{11} f\ sec^{-1})\ (2 \times 10^{6}\ watts)\ (\gamma_i)$$

and where:

λ_i = radioactive decay constant for the i^{th} radioiodine
N_i = saturation number of atoms of the i^{th} radioiodine present
γ_i = fractional fission yield for the i^{th} radioiodine
f = fissions

Table 13-1 gives values of fission yield γ_i and the activity of the appropriate radioactive iodine isotopes. The total number of iodine atoms present is [redacted] whence the number of molecules of I_2 present is [redacted] The activity of the i^{th} radioactive iodine isotope from a single fuel plate would be total core activity of that isotope divided by 14 fuel elements with 22 plates each.

TABLE 13-1 RADIOIODINE ACTIVITY PRESENT AT SATURATION IN THE PLATE

Iodine Isotope	γ_i fission yield	Activity (μCi)
131	0.029	[redacted]
132	0.043	[redacted]
133	0.065	[redacted]
134	0.080	[redacted]
135	0.064	[redacted]

The pool is assumed to release 1 % of the dissolved iodines to the confinement air. Based on the experience of the University of Virginia, this assumption is conservative by a factor of 100. Fifty percent of the iodines released to the confinement air are assumed to plate out and be unavailable for release to the environment outside the building through the emergency exhaust system. The charcoal filter within the emergency exhaust system is assumed to remove 99% of iodines. Source data, consistent with the above assumptions, are presented in Table 13-2.

TABLE 13-2 ACTIVITIES OF RADIOIODINE ISOTOPES IN CONFINEMENT

Iodine Isotope	Concentration μCi/cm^3
131	[redacted]
132	
133	
134	
135	

The release is assumed to originate from the stack and dilution is assumed to occur under Pasquill Type F condition (inversion) and a wind speed of one meter per second. Wind direction is assumed to be constant over the entire duration of the release. These assumptions are conservative given the influence of Narragansett Bay and the Atlantic Ocean on local meteorological conditions. The minimum radius defining the area under the immediate control of RINSC is 48 meters and this is the distance at which $(\chi/Q)_{eff}$ has been calculated. χ/Q is calculated according to

$$\chi/Q = \frac{1}{\Pi \sigma_y \sigma_z u}$$

Values of σ_y = 2.0 m and σ_z = 1.2 m for 48 meters have been estimated by extrapolation of the curves given in Reference 13.3 for Pasquill Type F condition:

$$x = \frac{1}{\Pi(2.0\text{m})(1.2\text{m})(1\text{m sec}^{-1})} = 0.133 \text{ sec m}^{-3} \text{ or } 1.33 \times 10^{-7} \text{ sec cm}^{-3}$$

The release rate in microcuries per second for each of the iodine isotopes is given in Table 13-3. The breathing rate is taken as 2 liters per minute. Values for conversion factors in millirems/microcurie are taken from Reference 13.2.

Although administrative control of the 48-meter radius around the reactor would be established much sooner under the RINSC Emergency Plan, a member of the public is assumed to be present at that distance for two hours before being discovered and removed to a greater safe distance.

TABLE 13 – 3 THYROID DOSE CALCULATION

Iodine Isotope	Release Rate (µCi/sec)	*Two-Hour Intake (µCi)*	*Dose Equivalent* (mrem/µCi)	Thyroid (mrem)
131	0.229	0.0592	1080	82.19
132	0.340	0.0877	6.44	0.73
133	0.514	0.1326	180	30.71
134	0.632	0.1632	1.06	0.22
135	0.507	0.1306	31.3	5.27
			Total:	119.12

The total committed dose equivalent to thyroid is a small fraction of the value of 5 rems for an individual member of the general public commonly assumed as the limiting thyroid dose for research reactors (10CFR20). It should also be noted that the numbers shown in Table 13-3 represent conservative estimates of committed dose equivalent based on the assumptions and restrictive meteorological considerations.

<u>Estimation of Whole Body Gamma Ray Exposure Consequences of Release of Gaseous Radioactivity from a Single Fuel Plate</u>

In the following calculations gamma ray doses are calculated on the basis of the submersion dose principle wherein the concentration of radioactivity at the 48 meter location is compared to the occupational MPC for each nuclide of interest. The assumptions of release and atmospheric dispersion are as indicated above. No adjustments have been made to account for the fact that the cloud of radioactivity is not semi-infinite in extent and the doses calculated are significantly higher than would be obtained under spatial equilibrium conditions. Table 13-5 shows source data for the released gases under the earlier release assumptions; A_i^0 represents the activity (µCi) of species i in the confinement air at the instant of release. MPCs for infinite cloud gamma submersion dose are calculated according to ICRP recommendations;

$$MPC_i = \frac{2.6\times10^{-6}}{\sum E_i}\left(\mu Ci\ cm^{-3}\right)$$

where $(MPC)_i$ is for an occupational situation (i.e., exposure to $(MPC)_i$ results in 2.5 mrem h^{-1} whole body dose) and where ΣE_i (E_i* in table) is taken as the total gamma energy emitted per disintegration of radionuclide, i. Table 13-5 contains pertinent data for the nuclides of major concern used in the calculations. Activities, $\lambda_i N_i$, are for release of the noble gases from a single plate as in the previous paragraph (i.e. A_i^0 x 0.5). Values of A_i^0 for radioiodines are as given in Table 13-2. (i.e. A_i^0 x 0.5 x 0.1)

TABLE 13 - 5

DATA USED IN ESTIMATION OF SUBMERSION CLOUD GAMMA DOSES

Nuclide	λ_i (/ hr)	γ_i (fis. yield)	E_i* (MeV/dis)	$(MPC)_i$ (µCi/ml)	$\lambda_i N_i$	A_i (µCi)
I-131	0.0035	0.029	0.400	6.50E-06	[redacted]	[redacted]
I-132	0.2888	0.043	2.120	1.23E-06	[redacted]	[redacted]
I-133	0.0333	0.065	0.550	4.73E-06	[redacted]	[redacted]
I-134	0.7920	0.080	1.250	2.08E-06	[redacted]	[redacted]
I-135	0.1037	0.064	1.500	1.73E-06	[redacted]	[redacted]
Kr-85m	0.1589	0.013	0.190	1.37E-05	[redacted]	[redacted]
Kr-87	0.5331	0.025	0.630	4.13E-06	[redacted]	[redacted]
Kr-88	0.2502	0.036	2.180	1.19E-06	[redacted]	[redacted]
Xe-131m	0.0024	0.029	0.002	1.30E-03	[redacted]	[redacted]
Xe-133m	0.0126	0.065	0.006	4.33E-04	[redacted]	[redacted]
Xe-133	0.0055	0.065	0.080	3.25E-05	[redacted]	[redacted]
Xe-135m	2.6637	0.064	0.150	1.73E-05	[redacted]	[redacted]
Xe-135	0.0760	0.064	0.240	1.08E-05	[redacted]	[redacted]

Assume a reduction of 10% by pool and 50% by plateout of iodines and 50% for noble gases.
* It must be noted that E_i* is the total gamma energy emitted per disintegration of the radionuclide as distinct from the average gamma energy per disintegration. Values of E_i* were obtained by dividing the core strength at 1 MW (MeV/sec) as given in Table IV of TID-14844 (Ref. 13-5) by the disintegration rate of the particular radionuclide at saturation activity in the core.

The instantaneous release rate, Q_i, of nuclide i at time t post the accident is given by:

$$Q_i = k\ A_i^0 e^{-(\lambda_i+k)t}$$

and the instantaneous concentration at 48 meters is given by x_i:

$$x_i = \left(4.4\times10^{-8}\right)k_i^0\ e^{-(\lambda_i+k)t}$$

whence the instantaneous dose rate at time t, R_i, is

$$R_i = \frac{\left(4.4\times10^{8}\right)k\,A_i^0\ e^{-(\lambda_i+k)T}\left(2.5\times10^{-3}\right)}{MPC_i}\left(rem\ hr^{-1}\right)$$

and the integral dose from t = 0 to t = T due to nuclide I is D_i:

$$D_i = \frac{\left(4.4\times10^{8}\right)k\,A_i^0\left(2.5\times10^{-3}\right)}{MPC_i\left(\lambda_i+k\right)}\left(1-e^{-(\lambda_i+k)T}\right)$$

where k (numerator) has units of sec^{-1} and (λ_i+k) has units of hr^{-1}.

Whole body gamma doses uncorrected for finite cloud size, for two hour and infinite times are given in Table 13-6. (0.1 rem for 2 hours and 0.2 rem for an infinite time)

TABLE 13 - 6
INTEGRAL WHOLE BODY GAMMA RAY DOSES AT 48 METERS (REMS)

Nuclide	2h	24h	48h	168h	Infinite
I-131	8.37E-04	4.10E-03	4.45E-03	4.48E-03	4.48E-03
I-132	5.06E-03	9.37E-03	9.37E-03	9.37E-03	9.37E-03
I-133	2.51E-03	1.03E-02	1.07E-02	1.07E-02	1.07E-02
I-134	3.73E-03	4.48E-03	4.48E-03	4.48E-03	4.48E-03
I-135	6.30E-03	1.87E-02	1.88E-02	1.88E-02	1.88E-02
Kr-85m	1.54E-03	3.80E-03	3.81E-03	3.81E-03	3.81E-03
Kr-87	7.14E-03	9.94E-03	9.94E-03	9.94E-03	9.94E-03
Kr-88	4.51E-02	8.95E-02	8.95E-02	8.95E-02	8.95E-02
Xe-131m	4.19E-05	2.07E-04	2.25E-04	2.26E-04	2.26E-04
Xe-133m	2.79E-04	1.29E-03	1.38E-03	1.38E-03	1.38E-03
Xe-133	3.75E-03	1.81E-02	1.96E-02	1.97E-02	1.97E-02
Xe-135m	1.38E-03	1.39E-03	1.39E-03	1.39E-03	1.39E-03
Xe-135	1.03E-02	3.44E-02	3.49E-02	3.49E-02	3.49E-02
Totals:	**0.0880**	**0.2055**	**0.2085**	**0.2087**	**0.2087**

For a distance of 100 meters, the above uncorrected whole body dose would be reduced to approximately 0.025 and 0.05 rem, respectively.
The above numbers compare to 500 mrem whole body dose to an individual member of the general public considered limiting for research reactors (10 CFR 20). An estimate of the actual potential gamma dose, considering the correction for the fact that the cloud is, indeed, finite in size may be made. If it is assumed that at the 48 meter distance the cloud is of the same effective dimension (width and height) as the reactor building we can more realistically estimate the gamma dose. Values of σ_y and σ_z are very small for the 48 meter distance and indeed for a point source release the y and z dimension of the cloud would be less than 5 meters). While the cloud may be considered infinite in extent in the x direction beyond the 48 meter point, it is considered to extend 48 meters in the x direction toward the reactor from the dose point, 24 meters in the z direction, and 12 meters in either y direction. The estimated fraction of the semispherical infinite cloud dose based on these dimensions is less than 0.10. The calculated 2 hours and infinite gamma dose at the 48 meter distance would then be reduced to approximately 0.01 rem and 0.02 rem, respectively.

<u>LEU Fuel and Doses with Plutonium Buildup</u>

With the previous use of HEU fuel, the content of ^{238}U was small and the production of plutonium and other actinides negligible. The LEU fuel will have a content of about 80% ^{238}U and a greater potential for the buildup of plutonium with burn-up. Studies have shown that though there may be a slight increase in dose due to plutonium buildup, the consequences are insignificant. (Reference 13.7)

13.2.2 Insertion of Excess Reactivity

Technical specification 1.25.7 sets an uncontrolled or unanticipated change in reactivity in excess of 0.5%Δk/k as a reportable occurrence. Section 3.1.3 restricts the total reactivity worth of all fixed experiments to a maximum of 0.6%Δk/k and moveable experiments to 0.08%Δk/k. In order to assure that these criteria are adequate, ANL ran a PARET Analysis using a 200 millisecond delay to beginning of control blade insertion (the blades actually begin to insert within 100 milliseconds of a scram).
Figure 13.1 shows the response of core power and peak clad temperature with time. The onset of nuclear boiling (ONB) is approached but does not occur. In fact, core power and the clad temperature reach a peak and begin to decrease prior to control blade insertion.

13.2.3 Loss of Coolant Accident (LOCA)

The loss of coolant analysis is the Design Basis Accident for the RINSC and is based upon an assumption that the beam port end in the pool is severed and the pool drains into the beam port and leaks out on reactor main floor and also to the beam port vent and drain lines.
There are four (4) six inch diameter and two (2) eight inch diameter aluminum beam ports which extend from the outside concrete wall through the pool wall liner at the mid-level of the reactor core. If the pool water drained to this level, active fuel would still remain immersed in about eight (8) inches of water. A typical beam port is shown in Figure 13-2. Figure 13-3 shows the vent and drain location and interconnection. The one inch line then runs to the outside of the concrete shield and has a one inch manual gate valve fitted with a one-half inch orifice. From there, all the beam port drain lines interconnect and run to the basement area below the reactor where the vent to an off-gas blower system which draws air through the experimental systems and discharges the activated air to the suction side of the reactor room exhaust fan, which discharges to the stack. It should be noted here that manual closing of the beam port 1 inch gate valves will stop a beam port vent line or drain line leak from continuing.
The beam port itself has four (4) basic barriers that can prevent leakage. The beam port end in the pool is a welded cap assembly (thimble). The beam port shutter, which can be lowered to close off the beam port, is another barrier between the pool and the reactor room end. A third barrier, concrete shielding plugs with a one inch spiral hole for instrument leads, sit at the reactor room end of the beam port. During experiments these plugs may be in place or may be replaced with an experimental facility of similar design. The final barrier is a cover flange that bolts to the end of the beam port. This flange is in place while the beam port is not in use for experiments. A fixed experiment which will be placed in the beam port end will be limited to a design which allows no more than an equivalent ½" diameter hole extending to the reactor room. All experiments must be designed to withstand a backpressure equivalent of the hydraulic head of the pool, or about 25 feet of water pressure.

Figure 13-4 shows the schematic diagram for the postulated loss of coolant calculations. The assumptions used to calculate the drainage time are shown below.

- A guillotine cutoff occurs at the beam port and floods a beam port. A 2 inch drop in pool level scrams the reactor,
- The shutter is fully open,
- The drain line manual valve is open,
- An experiment is in place with an aperture of ½ inch diameter maximum opening,
- The pool fill valve does not open automatically, and
- No operator action to reduce or stop flow.

It is important to note that the elevation of the bottom of the core box is lower than the bottom elevation of the eight (8) inch beam port. In the event of the water of the pool dropping to this elevation, the bottom 8 inches of the fuel element plates are still submerged in the core box water. Fuel plate cooling can take place due to the ½ inch diameter hole in the bottom of the core box, since the cooler surrounding pool water is available for heat removal.

SURFACE AREAS (FREE FLOW AREA)

Area of entire pool surface (A)	150 ft^2
Area of two 1/2 inch diameter holes (a)	0.00273 ft^2

A calculation was performed to determine the consequences of gravity draining of the pool from the pool surfaces to the middle of the core box out of the 8 inch beam port which has a plug in place and the shutter in the up position. There is no cover flange. The data elevation of 114.13 is used due to the assumption that water will not drain below this elevation in the event of shear of the 8 inch beam port.

Leak rate (maximum)

Datum is el. 114.1 feet (invert of the bottom of an eight-inch beam port)
el. 139.4 feet (normal water level of pool)

head of water (h) = 139.4 – 114.1 = 25.3 feet
area of leak (a) = two ½ inch diameter holes = 0.00273 ft^2
standard orifice discharge coefficient = 0.61

Flow through the standard orifice: $v = 0.61a\sqrt{2gh}$

$$v = 30.13gpm$$

Drain time of pool with two ½ inch diameter holes:

$$t = \frac{2A(\sqrt{h1} - \sqrt{h2}}{Ca\sqrt{2g}}$$

$$t = \frac{1508.6}{0.01335} = 31.4 hours$$

These calculations indicate that the pool would drain to the top of the core box in about 31.4 hours.

POOL MAKE-UP WATER

The Rhode Island reactor has a pool fill make-up system consisting of a 2 inch water line from a make-up demineralizer system which provides a normal flow of 5 gallons per minute. An automatic fill is initiated with a 1 inch drop in pool water level. A 2 inch drop in pool water level scrams the reactor. Manual filling of the pool provided 20 gallons per minute. The reliability of the water supply system has been described in Section 9.5 of this SAR. The Bay Campus can provide 5 gallons per minute or more, even with a power failure.

DECAY HEAT CALCULATIONS

In determining the decay heat generation following a LOCA, the following assumptions are made:

- The reactor has been operating for 40 hours continuously, therefore Table 13.8 can be used in determining the power ratio $P_{(ts)}/P_0$,
- The reactor scrams at time zero,
- The reactor pool water level and corresponding drainage time are as previously determined, and
- Following the LOCA, the decay heat is conducted away to water in the core box until the fuel either reaches its melting point or reaches a value that is lower.

The time to drain the pool to the mid section of the fuel elements is 31.4 hours.

2 MW Operation Heat Generation:

$$\mathbf{Q} = \frac{\mathbf{6.86 \times 10^6\ Btu/hr}}{\mathbf{14 elements \times 22 plates/element}} \times \frac{\mathbf{1}}{\mathbf{3{,}600 sec/hr}}$$

= 6.187 Btu/sec per plate

From Table 13.8 Time = 1.13×10^5 seconds; $\frac{P}{P_0} = 0.00465$

$$\frac{P}{P_0} = .00465 \times 6.187 = .02876\ Btu/\sec$$

This heat generation now must be removed by conduction and convection to assure that fuel plate melting does not occur. To calculate what heat generation would be required to melt the fuel, the following calculation was performed:

HEAT CONDUCTION DOWN THE PLATE (FUEL SECTION)

It is assumed that the heat generation sine distribution

The volumetric heat rate Q^{111} is defined a

$$Q^{111} = \frac{Q_{max}}{volume} = \frac{Q_{max}}{l \cdot w \cdot t} Btu/ft^3 \qquad (1)$$

where Q_{max} = Btu/sec
and l = fuel plate length
w = fuel plate width
t = fuel plate thickness

For an average sine

$$Q_{max} = Q_{ave} \times \Pi / 2 \qquad (2)$$

Substituting equation (2) into equation (1) we obtain:

$$Q^{111} = Q_{ave} \cdot \Pi / 2 \qquad (3)$$

For conduction downwards (-x direction) and using the heat conduction equation from **reference 13-X.**

$$\frac{d^2t}{dx^2} = \frac{-Q_{111}}{kf} \qquad (4)$$

where
kf = fuel plate thermal conductivity

for a sinusoidal heat generation

$$Q^{111} = Q_{max} \times \sin \Pi \times / l \qquad (5)$$

substituting equation (3) into equation (5) we obtain

$$Q^{111} = \frac{Q_{ave}}{l \cdot w \cdot t} \cdot \Pi / 2 \cdot \sin \Pi \times / l \qquad (6)$$

substituting (6) into (4)

$$\frac{d^2t}{dx^2} = \frac{-Q_{ave}}{l \cdot w \cdot t} \cdot \Pi / 2 \cdot l / kf \cdot \sin \Pi \times / 2 \qquad (7)$$

integrating (7)

$$\frac{dx}{dt} = -\Pi/2 \cdot Q_{ave}/l \cdot w \cdot t \cdot kf(-\cos \Pi x/l) \cdot l/\Pi + C_1$$

evaluating C_1 dt/dx = 0 AT x = 1 (1 = top of fuel)

$$C_1 = Q_{ave}/2\ w \cdot t \cdot kf$$

then

$$\frac{dt}{dx} = \frac{Q_{ave}}{w \cdot t \cdot kf} \cdot \cos \Pi x/l - Q_{ave}/2w \cdot kf$$

then integrating with the limits

T = 1200 °F at top of fuel plate x = 2.0'
T = 212 °F at surface of water in core box x = 0.7'

then

$$Q_{ave} = 0.013 \text{ Btu/sec}$$

HEAT CONDUCTION TO THE WATER IN CORE BOX FROM THE NON FUEL ALUMINUM IN THE ELEMENT

Calculation Basis – Per Plate

A. Non Fuel Plate Cross Section
Plate Cross Section = 0.05 ins x 2.79 ins = 0.1395 sq-ins
Max Fuel Cross Section = 0.02 ins x 2.47 ins = 0.0494 sq-ins
Non Fuel Plate Cross Section = 0.1395 ins – 0.0494 ins = 0.0901 sq-ins
22 plates x 0.0901 = 1.9822 sq-ins

B. Side Plates of the Element
Average width = 0.187 ins
22 grooves (0.187 – 0.088) x 0.058 ins
Cross Section
Two Side Plates x [(0.187 x 3.045) – 22 x (0.099 x 0.058)] = 0.8862 sq-ins

C. Total Area for the Element
Area = 1.9822 ins + 0.8862 ins = 2.8684 sq-ins
Per Plate Basis
Area = 2.8684 / 22 = 0.13038 sq-ins / 144 = 0.0009054 sq-ins
Heat Conducted from the Aluminum to the Water
$Q = k_{ae} A\ dt/dx$
$Q = k_{a1}A\ (T_{max} - T_{sat} / 1)$

TABLE 13.8

The Ratio, $P(t_s) / P_0$, of the Fission Product Decay Power to Reactor Operating Power as a Function of Time, t_s, After Shutdown (ANS, 1968)

Time After Shutdown, t_s (seconds)	Power Ratio $P(t_s) / P_0$	Time After Shutdown, t_s (seconds)	Power Ratio $P(t_s) / P_0$
1×10^{-1}	0.0675	6×10^{4}	0.00566
1×10^{0}	0.0625	8	0.00505
2	0.0590	1×10^{5}	0.00475
4	0.0552	2	0.00400
6	0.0533	4	0.00339
8	0.0512	6	0.00310
1×10^{1}	0.0500	8	0.00282
2	0.0450	1×10^{6}	0.00267
4	0.0396	2	0.00215
6	0.0365	4	0.00166
8	0.0346	6	0.00143
1×10^{2}	0.0331	8	0.00130
2	0.0275	1×10^{7}	0.00117
4	0.0235	2	0.00089
6	0.0211	4	0.00068
8	0.0196	6	0.00062
1×10^{3}	0.0185	8	0.00057
2	0.0157	1×10^{8}	0.000550
4	0.0128	2	0.000485
6	0.0112	4	0.000415
8	0.0105	6	0.000360
1×10^{4}	0.00965	8	0.000303
2	0.00795	1×10^{9}	0.000267
4	0.00625		

The calculations for the decay heat generation following the LOCA indicate the following:

- The pool would take about 31.4 hours to reach the middle of the core assuming no operator action and a simultaneous failure of make-up water, a highly unlikely occurrence.
- Further draining of the core to partially expose fuel would not lead to fuel melting.
- Administrative controls provide that proper design, installation and operation of fixed beam port experiments.
- Operation procedures have been established to provide that necessary reactor operator actions take place within a reasonable amount of time following a LOCA. For example, the lowering of the shutter, closing of the beam port drain valves, adding water to the pool if available, and repositioning the core into the fuel storage area (radiation levels being permissible) and placing the gate in place to isolate the leak.

Based upon the above analysis, the consequences of the potential LOCA are acceptable.

13.2.4 LOSS OF COOLANT FLOW

The forced convection mode of operation requires downward flow of coolant through the core. Upon loss of flow, the reactor scrams, coolant flow stops and reverses direction after a short time. The natural convection flow takes over and enters through the outlet plenum and upward through the core to the pool.

13.2.4.1 Loss of electrical power to primary pumps

Upon a loss of electrical power to the primary pumps the coolant temperature rises as the flow rate drops. When the flow drops to 80% of initial flow, the reactor automatically scrams and within a 0.1 second time delay the control blades start to insert, resulting in a rapid power and clad temperature drop. The power levels off at about 10% of its initial value but the flow rate continues to drop to a stagnant condition at 3.6 seconds after which it reverses direction. This causes the clad temperature to reach a minimum and to begin to rise at an increasing rate until the flow reversal. After the reversal, the upward flow increases while the power continues to slowly decrease. The clad temperature increase then begins to slow and peaks at 103°C shortly after 9 seconds.

13.2.4.2 Failure of a pump or other component in the primary coolant system

Failure of the inlet plenum coolant gate valve to open on a loss of coolant flow would not cause any problem with the natural circulation of the primary coolant. The natural circulation flow is up through the core and through the open top plenum to the pool. The coolant gates are held closed by the forced circulation and are opened by gravity.
Failure of the outlet plenum coolant gate valve to open on a loss of coolant flow would prevent the natural circulation path of pool water into the bottom of the core causing overheating/boiling and an erratic/uneven cooling of the fuel elements. A monthly test

and annual inspection of the gate valves insure that they will open on a loss of coolant flow. An automatic scram occurs if either gate valve is opened during forced convection modes of operation.

Failure of an isolation or check valve in the primary cooling loops could reduce the coolant flow without causing a scram. If coolant flow is reduced to the alarm set point of 90% of nominal flow it will alert the operator to the low flow condition. A check of primary temperatures and coolant flows will indicate to the operator any trends in system parameters. The reactor can be shutdown and the cause of the low coolant flow can be determined.

13.2.5 Start-Up Accident

This accident was analyzed using a digital computer program PARET (Reference 13.8). The accident is postulated to proceed under the following assumptions:

a. The reactor is in the cold clean condition with power at source level.
b. The servo regulating blade is withdrawn, followed by continuous withdrawal of all safety blades in succession at their maximum rate.
c. Period scram protection fails.
d. The reactor is scrammed by the high flux sensor instrumentation when the power level reaches 2.4 MW (20% overpower).
e. The delay time from generation of a high flux scram signal to the instant when the safety blades are free to drop is conservatively taken as 0.5 seconds.

The analyses indicate that the maximum fuel temperature (i.e., hot spot in the hottest channel) reaches 88.1°C (191°F) for the LEU fuel. Thus, it can be concluded that this accident results in no harm to the reactor.

If assumptions "c & d" are modified to – "period and high flux scram protection fails" – then reactor power would continue to rise beyond the scram trip point (2.4 MW) until the negative reactivity introduced by the void and temperature coefficients is greater than the net positive reactivity inserted by blade withdrawal. Table 13-9 provides the peak power and the maximum cladding temperature reached in the cladding for the LEU fuel cases. In each case, the maximum cladding temperature is less than 150°C (302°F) – much lower than the 582°C (1080°F) melting temperature of 6061 cladding. Table 13-10 provides the reactivity addition rate and the total energy released. The core would operate in the nucleate boiling range without physical damage until the accident could be terminated by a manual scram.

TABLE 13 – 9

Peak Power and Cladding Temperatures

Case	Peak Power, MW	Peak Cladding Temperature, °C
LEU Startup	14.9	148.3
LEU Equilibrium	16.2	148.5

TABLE 13 - 10

	LEU Startup	Equilibrium
Reactivity Addition Rate, %Δk/k/sec	0.0196	
(Total Ramp Addition)	3.104 %Δk/k / 157.9 seconds	
Minimum Period, seconds	0.79	
Total Energy Release, MW	287	344

13.2.6 Loss of Normal Electrical Power

Loss of normal electrical power will de-energize all reactor systems with the exception of emergency lighting, pool fill circuit, etc. switching to the emergency generator. The reactor parameters will react in the same way as section 13.2.4.1 with the addition of the reactor scramming upon loss of the normal electrical power.

13.2.7 External Events

Tornadoes and floods are extremely rare in the area around the RINSC reactor. Therefore, these events are not considered to be viable causes of accidents for the reactor facility. The RINSC facility has demonstrated it's ability to withstand the infrequent wind storms during the hurricane season. In addition, seismic activity is low in the area. The probability of these external events is discussed further in Chapter 2.
The RINSC site is located on the somewhat remote Graduate School of Oceanography Campus of the University of Rhode Island and is protected by an approved physical security plan. Overall security is far stricter than the surrounding civilian business and residential areas. Therefore, accidents caused by human controlled events which would damage the reactor, such as explosions or other unusual actions, are considered to be of very low probability.

13.2.8 Mishandling or Malfunction of Equipment

An analysis was made of the possibility that the loss of water from the reactor or from the radioactive liquid storage tanks could affect the environment. The reactor main floor drain empties into the storage tanks. Should the entire contents of the reactor pool be let out into the reactor room, some water could escape through the basement area into the campus storm drains. The pool water would then be directed and diluted by the campus drain network and released to the Narragansett Bay waters.
Malfunction of the confinement systems would have the greatest impact during the maximum hypothetical accident (MHA). Surveillances are conducted on these systems to ensure operability should an accident occur. Radioactive liquid leaks have been previously addressed. Although no damage to the reactor occurs as a result of these leaks, the details of the analyses provide a more comprehensive explanation.

13.3 Summary and Conclusions

This chapter of the Safety Analysis Report contains a conservative analysis of many different types of hypothetical accidents as they relate to the RINSC reactor and the surrounding environment. Beginning with the maximum hypothetical accident and continuing on through an entire array of other accidents, it has been shown that the consequences of such accidents will not result in occupational radiation exposure of the RINSC staff or radiation exposure of the general public in excess of applicable NRC limits.

REFERENCES

13.1 Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Standard Format and Content, NUREG-1537, Part 1, February 1996.

13.2 "The Health Physics and Radiological Health Handbook", 1992-Edited by Bernard Shleien, page 559.

13.3 USAEC "Meteorology and Atomic Energy", 1968, page 102-103.

13.4 USAEC Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors", June 1974.

13.5 "Reactor Site Criteria" Title 10, Code of Federal Regulations, Part 100, (10 CFR 100), February 1961.

13.6 "Calculation of Distance Factors for Power and Test Reactor Sites", J. J. DiNunno, et. al., TID-14844, March 1962 (Table IV).

13.7 "A Radiological Consequence Analysis with HEU and LEU Fuels," W.L. Woodruff, D.K. Watinner, J.E. Matos.

13.8 Obenchain, C.F., "PARET – A Program for the Analysis of Reactor Transients", IDO-17282 (1969).

CHAPTER FOURTEEN

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1.0 DEFINITIONS

1.1 Certified Operator

An individual authorized by the U. S. Nuclear Regulatory Commission to carry out the responsibilities associated with the position requiring the certification.

1.1.1 Senior Reactor Operator

An individual who is licensed to direct the activities of reactor operators. Such an individual may be referred to as a class A operator.

1.1.2 Reactor Operator

An individual who is licensed to manipulate the controls of a reactor. Such an individual may be referred to as a class B operator.

1.2 Confinement

Confinement means an enclosure on the overall facility which controls the movement of air into it and out through a controlled path.

1.3 Experiment

Any operation, component, or target (excluding devices such as detectors, foils, etc.), which is designed to investigate non-routine reactor characteristics or which is intended for irradiation within the pool, on or in a beam tube or irradiation facility and which is not rigidly secured to a core or shield structure so as to be part of their design.

1.3.1 Experiment, Moveable

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.

1.3.2 Experiment, Secured

A secured experiment is any experiment, experiment facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining force must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible conditions.

1.3.3 Experimental Facilities

An experimental facility is any structure or device which is intended to guide, orient, position, manipulate, or otherwise facilitate a multiplicity of experiments of similar character.

1.4 Explosive Material

Explosive material is any solid or liquid which is categorized as a severe, dangerous, or very dangerous explosion hazard in DANGEROUS PROPERTIES OF INDUSTRIAL MATERIALS by N.I. Sax, third Ed. (1968), or is given an identification of Reactivity (Stability) Index of 2, 3, or 4 by the National Fire Protection Association in its publication 704-M, 1966.

1.5 Instrumentation Channel

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.

1.5.1 Channel Test

Channel test is the introduction of a signal into the channel for verification that it is operable.

1.5.2 Channel Check

Channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification, where possible, shall include comparison of the channel with other independent channels or systems measuring the same variable.

1.5.3 Channel Calibration

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.

1.6 Limiting Conditions of Operation (LCO)

Lowest functional capability or performance levels of equipment required for safe operation of the reactor (10CFR50.36).

1.7 Limiting Safety System Setting (LSSS)

Settings for automatic protective devices related to those variables having significant safety functions, and chosen so that automatic protective action will correct an abnormal situation before a safety limit is exceeded (10CFR50.36).

1.8 Measured Channel

A measured channel is the combination of sensor, amplifiers, and output devices which are used for the purpose of measuring the value of a parameter.

1.9 Measured Value

The measured value of a parameter is the value of the variable as indicated by a measuring channel.

1.10 Operable

Operable means that a component or system is capable of performing its intended function.

1.11 Operating

Operating means that a component or system is performing its intended function.

1.12 Operational Reactor Core

An operational core is a standard core for which the core parameters of excess reactivity, shutdown margin, fuel temperature, power calibration, and reactivity worths of control blades and experiments have been determined to satisfy the requirements set forth in the Technical Specifications.

1.13 Protective Action

Protective action is the initiation of a signal or the operation of equipment within the reactor safety system in response to a variable or condition of the reactor facility having reached a specified limit.

1.14 Reactivity Excess

Excess reactivity is that amount of reactivity that would exist if all the control blades were moved to the maximum reactive condition from the point where the reactor is exactly critical.

1.15 Reactivity Limits

The reactivity limits are those limits imposed on the reactor core excess reactivity. Quantities are referenced to a reference core condition.

1.16 Reactivity Worth of an Experiment

The reactivity worth of an experiment is the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter equipment position or configuration.

1.17 Reactor Operating

The reactor is operating whenever it is not secured or shut down. The reactor has two modes of operation: natural circulation - not to exceed 0.1 MW and forced circulation - not to exceed 2 MW.

1.18 Reactor Safety Systems

Reactor safety systems are those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

1.19 Reactor Secure

The reactor is secure when:

1.19.1 Subcritical:

There is insufficient fissile material or moderator present in the reactor, control blades or adjacent experiments, to attain criticality under optimum available conditions of moderation and reflection, or the following conditions exist:

a. The minimum number of neutron absorbing control blades are fully inserted in shutdown position, as required by technical specifications.

b. The master switch is in the off position and the key is removed from the lock.

c. No work is in progress involving core fuel, core structure, installed control blades, or control blade drives unless they are physically decoupled from the control blades.

d. No experiments are being moved or serviced.

1.20 Reactor Shutdown

The reactor is shut down if it is subcritical by at least the shutdown margin in the reference core condition with the reactivity of all installed experiments included.

1.21 Readily Available on Call

Readily available on call shall mean a licensed senior operator shall insure that he /she, can be contacted within ten minutes and is within a 30 minute driving time from the reactor building when the reactor is being operated by a licensed operator.

1.22 Reference Core Condition

The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible < 0.05% $\Delta k/k$.

1.23 Regulating Blade

The regulating blade is a control blade of low reactivity worth fabricated from stainless steel and used to control reactor power. The blade may be controlled by the operator with a manual switch or by an automatic controller.

1.24 Removable Experiment

A removable experiment is any experiment, experimental facility, or component of an experiment, other than a permanently attached appurtenance to the reactor system, which can reasonably be anticipated to be moved one or more times during the life of the reactor.

1.25 Reportable Occurrence

A reportable occurrence is any of the following:

1. A safety system setting less conservative than the limiting setting established in the Technical Specifications;

2. Operation in violation of a limiting condition for operation established in the Technical Specifications;

3. A safety system component malfunction or other component or system malfunction which could, or threaten to, render the safety system incapable of performing its intended safety functions;

4. Release of fission products from a failed fuel element;

5. An uncontrolled or unplanned release of radioactive material which results in concentrations of radioactive materials inside or outside the restricted area in excess of the limits specified in Appendix B of 10CFR20;

6. An uncontrolled or unanticipated change in reactivity in excess of 0.5 %$\Delta K/K$;

7. Conditions arising from natural or man-made events that affect or threaten to affect the safe operation of the facility;

8. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or threatens to cause the existence or development of an unsafe condition in connection with the operation of the facility.

1.26 Research Reactor

A research reactor is defined as a device designed to support a self-sustaining neutron chain reaction for research, development, educational training, or experimental purposes, and which may have provisions for the production of radioisotopes.

1.27 Rundown

A rundown is the automatic insertion of the shim safety blades.

1.28 Safety Channel

A safety channel is a measuring channel in the reactor safety system.

1.29 Safety Limits

Safety limits are limits on important process variables which are found to be necessary to reasonably protect the integrity of the principal barriers which guard against the uncontrolled release of radioactivity. The principal barrier is the fuel element cladding.

1.30 Scram Time

Scram time is the elapsed time between reaching a limiting safety system set point and specified control blade movement.

1.31 Shim Safety Blade

A shim safety blade is a control blade fabricated from borated aluminum, which is used to compensate for fuel burnup, temperature, and poison effects. A shim safety blade is magnetically coupled to its drive unit allowing it to perform the function of a safety blade when the magnet is de-energized.

1.32 Shall, Should and May

The word "shall" is used to denote a requirement. The word "should" is used to denote a recommendation. The word "may" is used to denote permission, neither a requirement nor a recommendation.

1.33 Shutdown Margin

Shutdown margin shall mean 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 blade in its most reactive position, and that the reactor will remain subcritical without further operator action.

1.34 Secured Experiment

Any experiment, experimental facility, or component of an experiment is deemed to be secured, or in a secured position, if it is held in a stationary position relative to the reactor by mechanical means. The restraint shall exert sufficient force on the experiment to overcome the expected effects of hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or of forces which might arise as a result of credible malfunctions.

1.35 Static Reactivity Worth

The static reactivity worth of an experiment is the absolute value of the reactivity change, which is measurable by calibrated control blade comparison methods.

1.36 Standard Reactor Core

A standard core is an arrangement of (14) 22-plate LEU fuel elements in the reactor grid plate and may include installed experiments.

1.37 Surveillance Activities

Surveillance activities (except those specifically required for safety when the reactor is shutdown), may be deferred during reactor shutdown, however, they must be completed prior to reactor startup unless reactor operation is necessary for performance of the activity. Surveillance activities scheduled to occur during an operating cycle which cannot be performed with the reactor operating may be deferred to the end of the cycle.

1.38 Surveillance Intervals

Maximum intervals are to provide operational flexibility and to reduce frequency. Established frequencies shall be maintained over the long term. Allowable surveillance intervals shall not exceed the following:

1. 5 years (interval not to exceed 6 years).

2. 2 years (interval not to exceed 2 1/2 years).

3. Annual (interval not to exceed 15 months).

4. Semiannual (interval not to exceed 7 1/2 months).

5. Quarterly (interval not to exceed 4 months).

6. Monthly (interval not to exceed 6 weeks).

7. Weekly (interval not to exceed 10 days).

8. Daily (must be done during the calendar day).

1.39 True Value

The true value is the actual value of a parameter.

1.40 Unscheduled Shutdown

An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions, which could adversely affect safe operation, not including shutdowns which occur during testing or check-out operations.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

2.1.1 Safety Limits in the Forced Convection Mode

Applicability:

This specification applies to the interrelated variables associated with core thermal and hydraulic performance in the steady state with forced convection flow. These variables are:

Reactor Thermal Power, P
Reactor Coolant Flow through the Core, m
Reactor Coolant Outlet Temperature, T_o
Height of Water Above the Top of the Core, H

Objective:

To assure that the integrity of the fuel clad is maintained.

Specifications:

1. The true value of reactor power (P) shall not exceed 2.4 MW.
2. The true value of reactor coolant flow (m) shall not be less than 1580 gpm.
3. The true value of the reactor coolant outlet temperature (T_o) shall not exceed 125 °F.
4. The true value of water height above the active core (H) shall not be less than 23.54 feet while the reactor is operating at any power level.

Bases:

The basis for forced convection safety limits is that the calculated maximum cladding temperature in the hot channel of the most compact core will not be exceeded. The thermal hydraulic analysis (Part B, of the SAR) shows that if the safety limits are not exceeded the coolant will not reach the onset of nucleate boiling even at the safety limit of 2.4 MW. Additionally, the limit on coolant outlet temperature will prevent exceeding the temperature limit for the cleanup system resin.

2.1.2 Safety Limits in the Natural Convection Mode

Applicability:

This specification applies to the interrelated variables associated with core thermal and hydraulic performance in the natural convection mode of operation. These variables are:

Reactor Thermal Power, P
Height of Water Above the Top of the Core, H
Pool Temperature, T_p

Objective:

1. To assure that the integrity of the fuel clad is maintained.

2. To assure consistency with other defined safety system parameters.

Specification:

1. The true value of the reactor thermal power shall not exceed 217 kw.

2. The height of pool water above the core shall not be less than 23.54 feet.

3. The pool temperature does not exceed 130 ^oF.

Bases:

The basis for natural convection safety limits is that the calculated maximum cladding temperature in the hot channel of the most compact core will not reach nucleate boiling of the water coolant at a pool depth of 23.54 feet.

2.2 Limiting Safety System Settings (LSSS)

2.2.1 Limiting Safety System Setting in the Forced Convection Mode

Applicability:

LEU Fuel Temperature - Forced Convection Mode

Objective:

This specification applies to the setpoint for the safety channels monitoring reactor power, primary coolant flow, pool level and core outlet temperature to assure that the maximum fuel temperature permitted is such that no damage to the fuel cladding will result in the forced convection mode.

Specification:

The limiting safety system settings for reactor thermal power (P), primary coolant flow through the core (m), height of water above the top of the core (H), and reactor coolant outlet temperature (T_o) shall be as follows:

Parameter		LSSS
P	(Max)	2.30 MW
m	(Min)	1600 gpm
H	(Min)	23.7 ft
T_o	(Max)	121 °F

Bases:

These specifications were determined to prevent fuel temperatures from exceeding NRC temperature limits of 530 °C. This temperature was found acceptable as a result of NUREG-1313, "Safety Evaluation Report related to the Evaluation of Low-Enriched Uranium Silicide-aluminum Dispersion Fuel for Use in Nonpower Reactors". The SAR (Part B) provides the analyses showing fuel cladding temperatures well below the NUREG limit at normal operation. Flow and temperature limits were chosen to prevent incipient boiling even if transient power rises to the 2 MW trip limit of 2.4 MW. Variables used in the SAR were analyzed using uncertainties in flow measurement (3%) and temperature measurement (3%). These uncertainties were incorporated in the hot channel factors (1) used in the SAR thermal hydraulic studies. These same uncertainties were applied to the inlet and outlet temperature measurements.

The LSSS for the pool level is set for a scram upon a 2 inch drop in water level. The reference height of 23.7 feet (16 inches below suspension frame base plate elevation) is the depth of water above the top of the active fuel sitting in the existing reactor grid box. This depth was used in the SAR Loss of Coolant Analysis (Part B of the SAR). The safety limit settings chosen provide acceptable safety margins to the maximum fuel cladding temperature. The startup accident transient analysis (Part A, Section XI of the SAR) also provides results showing that the cladding temperature limit is not exceeded. The LOCA analysis (Design Basis Accident, Part A, Section IX and Part B, Section X and Appendix D of the SAR) shows that the fuel cladding limit is not exceeded.

The LSSS for the pool level results in a higher number since the pool level scrams upon a 2 inch drop in water level.

(1)Reference: Report on the Determination of Hot Spot Factors for the RINSC Research Reactor, August 1989.

2.2.2 Limiting Safety System Settings in the Natural Convection Flow Mode

Applicability:

These specifications apply to the setpoint for the safety channels monitoring reactor thermal power level (P), monitors for pool level (H), and pool water temperature (T_p), in the natural convection mode.

Objective:

To assure that automatic protective action is initiated to prevent a safety limit from being exceeded.

Specification:

1. The limiting safety system setting for reactor thermal power (P), height of water above the top of the core (H), and pool water temperature (T_p) shall be as follows:

Parameter		LSSS
P	(Max)	115 kw.
H	(Min)	23.7 ft.
T_p	(Max)	126 °F

Bases:

The SAR has determined that up to 217 kw. can be removed by natural convection, however, the existing license requirement of 100 kw. operation will be maintained and with a 15% overpower trip, 115 kw. will be the LSSS. The pool level scram (2 inch drop) is the same as the forced convection mode. The pool temperature 130 °F safety limit, having a 3% error, results in a LSSS of 126 °F. The LSSS for natural convection assures that automatic protective action will prevent a safety limit from being exceeded.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactivity Limits

Applicability:

This specification applies to the reactivity of the reactor core and to the reactivity worths of control blades and experiments.

Objective:

To assure that the reactor can be controlled and shut down at all times and that the safety limits will not be exceeded.

Specification:

1. The shutdown margin relative to the reference core condition shall be at least 1.0 %ΔK/K with the most reactive shim safety blade and the regulating blade fully withdrawn.

2. The overall core excess reactivity including movable experiments shall not exceed 4.7 %ΔK/K.

3. The total reactivity worth of all experiments shall not exceed 0.6 %ΔK/K.

4. The reactivity worth of each experiment shall be limited as follows:

Experiment	Maximum Reactivity Worth
Moveable	0.08 %ΔK/K
Secured	0.60 %ΔK/K

5. The reactor shall be subcritical by at least 3.0 %ΔK/K during fuel loading changes.

6. The reactivity worth of the regulating blade shall not exceed 0.6 %ΔK/K.

7. Experiments which could increase reactivity by flooding, shall not remain in or adjacent to the core unless the shutdown margin required in Specification 3.1.1 would be satisfied after flooding.

8. The temperature coefficient will be negative and surveillance will be conducted at initial startup and change in fuel type.
9. For operation at power levels in excess of 0.1 MW in the forced convection mode, all grid positions shall contain fuel elements, baskets, reflector elements, grid plugs or experimental facilities.

10. For operation at powers in excess of 0.1 MW, the pool gate must be in its storage location.

Bases:

Specification 3.1.1 assures that the reactor can be shutdown from any operating condition and will remain subcritical after cool down and xenon decay even if the blade of the highest reactivity worth should be in the fully withdrawn position. The SAR (Part A, Section V) demonstrates that the shutdown margin conservatively exceeds the 1% in Specification 3.1.1.

Specification 3.1.2 limits the allowable excess reactivity to the value necessary to overcome the combined negative reactivity effects of: (1) an increase in primary coolant temperature; (2) fission product xenon and samarium buildup in a clean core; (3) power defect due to increasing from a zero power, cold core to a 2 MW, hot core; (4) fuel burnup during sustained operation; and (5) moveable experiments.

Specification 3.1.3 limits the reactivity worth of experiments to values of reactivity which, if introduced as positive step changes, will not cause fuel melting.

Specification 3.14 limits the individual reactivity worth of an experiment to a value that will not produce a stable period of less than 30 seconds and which can be compensated for by the action of the control and safety system without exceeding any safety limits.

Specifications 3.1.5 provide assurance that the core will remain subcritical during fuel loading changes.

Specification 3.1.6 assures that failure of the automatic control system will not introduce sufficient excess reactivity to produce a prompt critical condition.

Specification 3.1.7 assures that the shutdown margin required by Specification 3.1.1 will be met in the event of a positive reactivity insertion caused by the flooding of an experiment.

Specification 3.1.8 assures that the power increase is self limiting.

Specification 3.1.9 will prevent the degradation of flow rates due to flow bypassing the active fueled region through an unoccupied grid plate position.

Specification 3.1.10 assures that the full volume of the pool water is available to provide cooling of the core during normal operation and in the event of a loss of coolant accident.

3.2 Reactor Safety System

Applicability:

These specifications apply to the reactor safety system and other safety related instrumentation.

Objective:

To specify the lowest acceptable level of performance or the minimum number of acceptable components for the reactor safety system and other safety related instrumentation.

Specification:

The reactor shall not be made critical unless:

1. The reactor safety systems and safety related instrumentation are operable in accordance with Tables 3.1 and 3.2 including the minimum number of channels and the indicated maximum or minimum setpoint;

2. All shim safety blades are operable in accordance with Technical Specification 4.1.1 and 4.1.2.

3. The time from initiation of a scram condition until the control element is fully inserted shall not exceed 1 second in accordance with Technical Specification 4.2.5 and 4.2.6.

4. The reactivity insertion rates of individual control and regulating blades will not exceed 0.02 %ΔK/K per second.

Bases:

Neutron flux level scrams provide redundant automatic protective action to prevent exceeding the safety limit on reactor power. The period scram limits the rate of rise of the reactor power to periods which are manually controllable without reaching excessive power levels or fuel temperatures.

The loss of flow scram assures that an automatic scram will occur in the event of a loss of flow when the reactor is operating at power levels above 0.1 MW.

The reactivity insertion rate limit was determined in the SAR, Section XI and predicts a safe fuel clad temperature.

TABLE 3.1

REQUIRED SAFETY CHANNELS

Reactor Safety System Component/Channel	Minimum Required	Function	Reactor Operating Mode in Which Required
Reactor Power Level	2	Automatic scram when ≥ 115% of range scale with 2.3 MW max	Both Modes
Coolant Flow Rate	1	Automatic scram at ≤ 1600 gpm	Forced Convection above 0.1 MW
Seismic Disturbance	1	Automatic scram at Modified Mercalli Scale IV	Both Modes
Bridge Misalignment	1	Automatic scram	Forced Convection above 0.1 MW
Pool Water Level	1	Automatic scram at 16" below suspension frame base plate elevation	Both Modes
Coolant Outlet Temperature	1	Automatic scram ≥ 121°F	Forced Convection above 0.1 MW

Reactor Safety System Component/Channel	Minimum Required	Function	Reactor Operating Mode in Which Required
Bridge Movement	1	Automatic scram	Both Modes
Coolant Gates Open	1	Automatic scram if either the coolant riser or coolant downcomer gates open	Forced Convection above 0.1 MW
Detector High Voltage Failure	3	Automatic scram if Voltage decreases 50V max	Both Modes
Log N Period	1	Automatic scram if period ≤ 4 sec	Both Modes
No Flow Thermal Column	1	Automatic scram	Forced Convection above 0.1 MW
Manual Scram Switch (console, bridge)	2	Manual scram	Both Modes
Pool Temperature	1	Automatic Scram at ≥ 126 °F	Natural Convection

TABLE 3.2

Required Safety Related Instrumentation

	Instrumentation	Setpoint	Minimum Number Required	Function	Reactor Operating Mode In Which Reqd.
1.	Reactor Coolant Inlet Temperature	≤ 111°F	1	Alarm	FC ≥ 0.1MW
2.	Reactor Coolant Outlet Temperature	≤ 119°F	1	Alarm	FC ≥ 0.1MW
3.	Log Count Rate	< 3 cps	1	Blade withdrawal interlock	Both Modes
4.	Servo Control Interlock	≥ 30 sec (fullout)	1	Auto Control Interlock	Both Modes
Facility Radiation (a) Monitoring System					
5.	Building Air Gaseous Exhaust (Stack)	2.5 x normal particulate 2 x normal	1	Alarm	Both Modes
6.	Reactor Bridge	2 x normal	*	Alarm	Both Modes
7.	Fuel Safe	2 x normal or 5mR/hr, which ever is higher	*	Alarm	Both Modes

	Instrumentation	Setpoint	Minimum Number Required	Function	Reactor Operating Mode In Which Reqd.
8.	Thermal Column	2 x normal or 2mR/hr, which ever is higher	*	Alarm	Both Modes
9.	Heat Exchanger	2 x normal	*	Alarm	Both Modes
10.	Primary Demineralizer (Hot DI)	2 x normal	*	Alarm	Both Modes
11.	Continuous Air Monitoring Unit	2 x normal	1	Alarm	Both Modes

NOTES

(a) The facility radiation monitoring system consists of 8 radiation detectors which alarm and readout in the control room except for #11 which has a local alarm and readout only. The normal setpoints for this system are shown in Table 3.2. Use of higher than normal setpoints will require approval of the Director or the Assistant Director. Any senior operator member may adjust a setpoint lower than the normal value.

(b) *The reactor shall not be continuously operated without a minimum of one radiation monitor on the experimental level of the reactor building and one monitor over the reactor pool operating and capable of warning personnel of high radiation levels.

3.3 Coolant Water

(a) Primary Coolant Water

Applicability:

This specification applies to the limiting conditions for primary coolant pH, resistivity, available pool water volume and radioactivity.

Objective:

To maintain the primary coolant in a condition to minimize the corrosion of the primary coolant system, fuel cladding, and other reactor components, and to assure proper conditions of coolant for normal and emergency requirements.

Specification:

1. The primary coolant pH shall be maintained between 5.5 and 7.5.

2. The primary coolant resistivity shall be maintained at a value greater than 500K ohms/cm (conductivity 2 micromhos/cm).

3. The primary coolant shall be analyzed for radioactivity.

Bases:

Experience at this and other facilities has shown that the maintenance of primary coolant system water quality in the ranges specified in specification 3.3.1 and 3.3.2 will control the corrosion of the aluminum components of the primary coolant system and the fuel element cladding. Conductivity Specification 3.3.2 also insures adequate water purity to control activation of coolant water impurities.

The requirement in specification 3.3.3 ensures that the presence of unusual impurities or corrosion products is detected.

(b) Secondary Coolant Water

Applicability:

This specification applies to the limiting conditions for secondary coolant pH, cycles of chloride, resistivity and radioactivity.

Objective:

To maintain the secondary coolant in such a condition as to minimize corrosion and/or scale buildup on the heat exchanger tubes and to detect a primary to secondary system leak.

Specification:

1. The secondary coolant water pH shall be maintained between 5.5 and 9.0.

2. The sample will be analyzed for the presence of sodium-24.

Bases:

The facility has maintained the above coolant water conditions for many years based on consultant recommendations and have good results in maintaining heat exchanger tube and shell cleanliness.

Radioactivity in the secondary system would indicate a leak and therefore samples are analyzed for detectable concentrations of sodium-24.

3.4, 3.5, 3.6 Confinement and Emergency Exhaust System and Emergency Power

Applicability:

This specification applies to the operation of the reactor confinement and emergency exhaust system which must be operable during reactor operation, fuel handling and any operation that could cause the spread of airborne radioactivity in the confinement area.

Objective:

To assure that the confinement and emergency exhaust system is capable of operation to mitigate the consequences of possible release of radioactive materials resulting from reactor operation, fuel movement and handling of radioactive material.

Specification:

The reactor shall not be operated unless the following equipment is operable and/or conditions met:

Equipment/Condition	Function
Personnel access doors to reactor closed (except for entrance and egress). Roof hatch closed.	To maintain confinement system integrity

Truck door closed;	
Reactor Room fresh air intake valve and exhaust ventilation valve to the stack are open;	To maintain confinement system integrity
Initiation system for confinement isolation, i.e. evacuation buttons and alarm horns;	To initiate system operation and alert personnel
Emergency cleanup exhaust system;	To maintain a negative building pressure without unloading any large fraction of possible airborne activity
Emergency generator. source to clean up	To insure power system and other designated systems

Bases:

The confinement system will be activated by depressing any one of five emergency evacuation buttons when an unsafe radiological situation develops as defined in the facility operating and emergency procedures. In the unlikely event of a release of fission products, or other airborne radioactivity, the confinement isolation initiation system will secure the normal ventilation exhaust fan, will bypass the normal ventilation supply up the stack, and will close the normal inlet and exhaust valves. In confinement, the emergency exhaust system will tend to maintain a negative building pressure with a combination of controls intended to prevent unloading any large fraction of airborne activity. The emergency exhaust purges the building air through charcoal and absolute filters and controls the discharge which is diluted by supply air through a 115 foot stack.

3.7 Radiation Monitoring Systems and Effluents

3.7.1 Radiation Monitoring Systems

Applicability:

This specification applies to the availability of radiation monitoring equipment which must be operable during reactor operation, fuel movement and handling of radioactive materials in the reactor building.

Objective:

To assure that radiation monitoring equipment is available for evaluation of radiation conditions and that the release of airborne radioactive material is maintained below the limits established in 10CFR20.

Specification:

1. When the reactor is operating, gaseous and particulate sampling of the stack effluent shall be monitored by a stack monitor with a readout in the control room.

   The particulate activity monitor and the gaseous activity monitor for the facility exhaust stack shall be operating. If either unit is out of service for more than one shift (6 hours), either the reactor shall be shut down or the unit shall be replaced by one of comparable monitoring capability.

2. When the reactor is operating, at least one constant air monitoring unit (Table 3.2.11) located in the confinement building shall be operating. Temporary shutdown of this unit shall be limited as in 3.7.1 above.

3. The reactor shall not be continuously* operated without a minimum of one area radiation monitor (Table 3.2.8) on the experimental level of the reactor building and one area monitor (Table 3.2.6) over the reactor pool (reactor bridge) operating and capable of warning personnel of high radiation levels.

   *In order to continue operation of the reactor, replacement of an inoperative monitor must be made within 15 minutes of recognition of failure, except that the reactor may be operated in a steady-state power mode if the monitor is replaced with portable gamma-sensitive instruments having their own alarm.

Bases:

A continuing evaluation of the radiation levels within the reactor building will be made to assure the safety of personnel. This is accomplished by the monitoring systems described in Table 3.2.

3.7.2 Effluents

a. Airborne Effluents

Applicability:

This specification applies to the monitoring or airborne effluents from the Rhode Island Nuclear Science Center (RINSC).

Objective:

To assure that containment integrity is maintained during reactor operation and that the release of airborne radioactive material from the RINSC is maintained below the limits established in 10CFR20.

Specification:

1. The concentration of radioactive materials in the effluent released from the facility exhaust stacks shall not exceed 10^5 times the concentrations specified in 10CFR20, Appendix B, Table II, when averaged over time periods permitted by 10CFR20.

Bases:

The limits established in specification 3.7.2 incorporate a dilution factor of $4x10^4$ for effluent concentrations released through the exhaust stacks. This dilution factor is based on a dispersion factor ($\chi/Q = 10^{-5}$ sec/M^3) calculated from actual meteorological data, which is determined using the highest frequency of wind in any sector. Because of the use of the most conservative measured values of wind directional frequency and dispersion factors, this dilution factor will assure that concentrations of radioactive material in unrestricted areas around the Rhode Island Nuclear Science Center will be far below the limits of 10CFR20. (Refer to letter dated April 16, 1963 sent to the NRC in connection with license questions.) This dilution factor is used for calculating maximum ground concentration of noble gases down wind vs. exhaust stack effluent concentrations. The SAR contains calculations for doses from the iodine at the 48 meter distance.

b. Liquid Effluents

Applicability:

This specification applies to the monitoring of radioactive liquid effluents from the Rhode Island Nuclear Science Center.

Objectives:

The objective is to assure that exposure to the public resulting from the release of liquid effluents will be within the regulatory limits and consistent with as low as reasonably achievable requirements.

Specification:

The liquid waste retention tank discharge shall be batch sampled and the gross activity per unit volume determined before release. All off-site releases shall be directed into the municipal sewer system.

Bases:

All radioactive liquid and solid wastes disposed of off-site shall be within the limits established by 10CFR20 or shall be removed from the site by a commercial licensed organization.

3.8 Limitations on Experiments

Applicability:

This specification applies to experiments to be installed in the reactor and associated experimental facilities.

Objectives:

To prevent damage to the reactor or release of radioactive materials in excess of 10CFR20.

Specification:

The reactor shall not be operated unless the following conditions governing experiments exist;

1. All materials to be irradiated shall be either corrosion resistant or encapsulated within corrosion resistant containers to prevent interaction with reactor components or pool water. Corrosive materials shall be doubly encapsulated.

2. Irradiation containers to be used in the reactor, in which a static pressure will exist or in which a pressure buildup is predicted, shall be designed and tested for a pressure exceeding the maximum expected by a factor of 2.

3. Fissionable materials shall have total iodine and strontium inventory less than that allowed by the facility by-product license.

4. Explosive materials, in any quantity, shall not be allowed in the reactor pool or experimental facilities.

5. All experiments shall be designed against failure from internal and external heating at the true values associated with the LSSS for reactor power level and other process parameters.

6. Experimental apparatus, material or equipment to be irradiated shall be positioned so as not to cause shadowing of the nuclear instrumentation, interference with control blades, or other perturbations which may interfere with safe operation of the reactor.

7. Cryogenic liquids shall not be used in any experiment within the reactor pool without approval from the Nuclear Regulatory Commission.

8. No highly water reactive materials shall be used in an experiment in the reactor pool.

9. No experiment should be performed unless the material content (with the exception of trace constituents) is known.

10. If a capsule fails and releases material which could damage the reactor fuel or structure by corrosion or other means, removal and physical inspection shall be performed to determine the consequences and need for corrective action. The results of the inspection and any corrective action taken shall be reviewed by the Director, or his designated alternate, and determined to be satisfactory before operation of the reactor is resumed.

    Experimental materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under: (1) normal operating conditions of the experiment or reactor, (2) credible accident conditions in the reactor, and (3) possible accident conditions in the experiment shall be limited in activity such that: if 100% of the gaseous activity or radioactive aerosols produced escaped to the reactor room or the atmosphere, the airborne concentration of radioactivity averaged over a year would not exceed the occupational limits for maximum permissible concentration.

    In calculations pursuant to the above, the following assumptions shall be used: (1) If the effluent from an experimental facility exhausts through ductwork which closes automatically on high radiation level, at least 10% of the gaseous activity or aerosols produced will escape. (2) If the effluent from an experimental facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of these vapors can escape. (3) For materials whose boiling point is above 55°C and where vapors formed by boiling this material can escape only through an undisturbed column of water above the core, at least 10% of these vapors can escape. (4) Limits for maximum permissible concentrations are specified in the appropriate section of 10CFR20.

Bases:

Specifications 1 through 5, 8 and 9 are intended to reduce the likelihood of damage to reactor components and/or radioactivity releases resulting from experiment failure and, along with the reactivity restriction of pertinent specification in 3.1, serve as a guide for the review and approval of new and untried experiments by the operations staff as well as the Nuclear and Radiation Safety Subcommittee.

Specifications 3 and 4 are self explanatory.

Specification 6 assures that no physical or nuclear interferences compromise the safe operation of the reactor by, for example, tilting the flux in a way that could effect the peaking factor used in the Safety Analysis.

Specification 7 insures NRC review of experiments containing or using cryogenic materials. Cryogenic liquids present structural and explosive problems which enhance the potential of an experiment failure.

Specification 10 is self explanatory.

3.9. Reactor Core Components

a. Beryllium Reflectors

Applicability:

This specification applies to neutron flux damage to the standard and plug type beryllium reflectors.

Objective:

To prevent physical damage to the beryllium reflectors in the core from accumulated neutron flux exposure.

Specification:

1. The maximum accumulated neutron flux shall be 1×10^{22} neutrons/cm^2.

Bases:

The RINSC SAR (Part A Section VIII) has addressed this limit as a conservative limit.

b. LEU Fuel

Applicability:

This specification applies to the physical condition of the fuel elements.

Objective:

To prevent operation with damaged fuel elements.

Specification:

Fuel elements to be inspected for physical defects and reactor core box fit in accordance with manufactured specifications.

Bases:

The RINSC inspects and tests each fuel element for reactor core box fit in accordance with written procedures to assure operation with fuel elements that are not damaged and meet specifications.

4.0 SURVEILLANCE REQUIREMENTS

Surveillance tests for Reactivity Limits (4.1), Reactor Safety System (4.2), Surveillance of Experiments (4.8) and Reactor Components (4.9) may be deferred for periods of reactor shutdown providing they are performed prior to restart (ANS 15.1, 4.1). Surveillance tests for the following will be performed as stated in the appropriate sections:

Water Coolant System (4.3)
Confinement and Emergency Exhaust System (4.4, 4.5, 4.6)
Radiation Monitoring System and Effluents (4.7)

4.1 Reactivity Limits

Applicability:

This specification applies to the surveillance requirements for reactivity limits.

Objective:

To assure that the reactivity limits of Specification 3.1 are not exceeded.

Specification:

1. Shim safety blade reactivity worths and insertion rates shall be measured:

   a. annually;

   b. whenever the core is changed from the startup core to the three other cores as analyzed and specified in the SAR (Part A, Section V).

2. Shim safety blades shall be visually inspected and checked for swelling at least annually.

3. The reactivity worth of all experiments shall be measured prior to the experiment's initial use.

Bases:

Specification 4.1.1 will assure that shim safety blade reactivity worths are not degraded or changed by core arrangements.

Shim safety blade inspections are the single, largest source of radiation exposure to facility personnel. In order to minimize personnel radiation exposure and provide an inspection frequency that will detect early evidence of swelling and cracking, an annual inspection interval was selected for Specification 4.1.2.

The specified surveillance relating to the reactivity worth of experiments will assure that the reactor is not operated for extended periods before determining the reactivity worth of experiments. This specification also provides assurance that experiment reactivity worths do not increase beyond the established limits due to core configuration changes.

4.2 Reactor Safety System

Applicability:

This specification applies to the surveillance of the reactor safety system.

Objective:

To assure that the reactor safety system is operable as required by Specification 3.2.

Specification:

1. A channel test of the neutron flux level safety channels and period safety channel shall be performed:

    a. Prior to each reactor startup following a period when the reactor was secured;

    b. After a channel has been repaired or de-energized.

2. A channel calibration of the safety channels listed in Table 3.1, which can be calibrated, shall be performed annually.

3. The radiation monitoring system required in Table 3.2 shall be operable prior to every reactor startup for which safety system channel tests are required as in 4.2.1. If the system has been repaired, the system shall be operable prior to use.

4. Shim safety blade release-drop time shall be measured annually.

5. Shim safety blade release-drop time shall be measured whenever the shim safety blade's core location is changed or whenever maintenance is performed which could effect the blade's drop time. (Specification 3.2.3)

6. Shutdown Margin (Specification 3.1.1)

    The shutdown margin shall be determined annually. It shall be determined when a new core is configured as described in the SAR (Part A, Section V). The determination will be made in accordance with operating procedures.

7. Excess Reactivity (Specification 3.1.2)

The excess reactivity shall be determined annually. It shall be determined when a new core is configured as described in the SAR (Part A, Section V). The determination will be made in accordance with operating procedures.

8. Reactivity Insertion Rate (Specification 3.2.4)

The reactivity insertion rate shall be measured annually. It shall be determined when a new core is configured as described in the SAR (Part A, Section V). The determination will be made in accordance with written procedures.

Bases:

Pre startup tests of the safety system channels assure their operability. Annual calibration detects any long term drift that is not detected by normal inter comparison of channels. The channel operability check of the neutron flux level channels assures that the detectors are properly adjusted to accurately monitor the parameter they are measuring.

Radiation monitors are checked for proper operation in Specification 4.2.3. Calibration and setpoint verification involve use of a calibration source and significant personnel radiation exposure. It is determined that annual calibration of radiation monitors is adequate since they displayed excellent stability over many years of operation.

The measured release-drop times of the shim safety blades have been consistent over many years. Annual check of these parameters is considered adequate to detect any deterioration, which could change the release-drop time. Binding or rubbing caused by blade misalignment could result from maintenance; therefore, release drop times will be checked after such maintenance.

4.3 Water Coolant System

a. Primary Coolant System

Applicability:

This specification applies to the surveillance of the primary coolant system.

Objective:

To assure high quality pool water and to detect the deterioration of components in the primary coolant loop.

Specification:

1. The pH of the primary coolant shall be measured weekly.

2. The resistivity of the primary coolant shall be measured weekly.

3. The radioactivity of the primary coolant shall be analyzed weekly for gross activity and quarterly for isotopic activity.

4. Pool water level scram switch shall be checked for operation monthly.

5. Pool inspections shall be made annually in accordance with operating procedures.

6. Pool level shall be visually inspected daily in accordance with operating procedures.

Bases:

Regular surveillance of pool water quality and radioactivity provides assurance that pH and resistivity changes that could accelerate the corrosion of the primary system components would be detected before significant damage would occur, and that the presence of leaking fuel elements in the reactor is detected.

The low pool level switch is checked for operation monthly. Upon a one inch pool level drop, the automatic fill begins; upon a two inch drop, the reactor scrams (if operating) and a local and remote alarm sounds. The remote alarm is continuously monitored offsite.

Annual pool system inspections are made to provide assurance that other cooling system components (eg. gate valves, gasketing etc.) are functioning properly.

b. Secondary Coolant System

Applicability:

This specification applies to the surveillance of the secondary coolant water.

Objective:

To assure the conditions of the coolant meet specification 3.3.(b) and to detect a primary to secondary water leak.

Specification:

1. The pH shall be measured weekly during reactor operation.

2. A sample shall be drawn weekly and analyzed for sodium-24 activity, during reactor operation.

Bases:

Proper secondary coolant conditions are obtained by blowdown and makeup water systems which maintain the proper water quality pH. Radioactive concentrations are measured in accordance with written procedures.

4.4, 4.5, 4.6 RINSC Confinement and Emergency Exhaust System

Applicability:

This specification applies to the surveillance of the facility openings and dampers.

Objective:

To assure that the condition of the closure devices for the building openings are in satisfactory condition and to assure their ability to provide adequate confinement of any airborne radioactivity released into the building.

Specification:

1. The confinement and emergency exhaust system described in Specification 3.4 shall be tested weekly for operability and after any maintenance that could affect system operability. The system operation is as described in the operating procedures and as herein discussed. The building cleanup system shall be activated by pressing an evacuation button, then automatically:

    a. the evacuation horn sounds

    b. the building ventilation blowers deenergize (air conditioner, exhaust blower, off gas blower, rabbit system blower, heating system blowers);

    c. the building ventilation dampers close (air intake and exhaust system);

    d. the cleanup system blower (through the scrubber filter) and air dilution blower (chem.. lab) are energized;

    e. the negative differential pressure between the inside and outside of the building is at least 0.5 inches of water. This is determined by reading the pressure gauge in the control room;

    f. the exhaust rate through the emergency cleanup system shall not be more than 1500 CFM coming from the reactor building and passing through the scrubber filters. Dilution air will be provided by a separate blower from an uncontaminated source.

2. The condition of the following equipment shall be inspected in accordance with written operating procedures every 6 months.

a. Building ventilation blowers and dampers (including solenoid valves, pressure switches, piping, etc.);

b. Personnel access and reactor room overhead doors.

3. The testing and maintenance of the emergency generator will be performed in accordance with the RINSC operating procedures and manufacturer recommendation.

4. The efficiency test for the charcoal filter shall be tested annually as specified in the operating procedures.

Bases:

The weekly check of the confinement system provides assurance that the automatic function will be actuated when confinement isolation is required. The semiannual inspection of valves and doors will provide assurance that the closures will perform their function of limiting leakage through these openings in the event of a release of airborne activity into the building.

The testing of the emergency generator assures reliable response and operation. The load testing assures proper handling of expected system loads. The emergency generator system has maintenance performed in accordance with manufacturer recommendations.

4.7 Radiation Monitoring Systems and Effluents

a. Airborne Effluents

Applicability:

This specification applies to the surveillance of the monitoring equipment used to measure airborne radioactivity.

Objective:

The objective is to assure that accurate assessment of airborne effluents can be made.

Specification:

1. The particulate air monitors shall be calibrated annually.

2. The gaseous activity monitor shall be calibrated annually.

3. A channel check of the stack monitor and the main floor monitor shall be performed daily when the reactor is in operation.

Bases:

Experience with the electronic reliability and calibration stability of the units used by the Rhode Island Nuclear Science Center Reactor demonstrates that the above periods are reasonable surveillance frequencies.

b. Liquid Effluents

Applicability:

This specification applies to the surveillance of the monitoring equipment used to measure the radioactivity in liquid effluents.

Objective:

The objective is to assure that accurate assessment of liquid effluents can be made.

Specification:

1. The monitoring equipment used to measure the radioactive concentrations in the waste retention tanks shall be calibrated annually.

2. The contents of every tank released shall be sampled and evaluated for radioactive concentrations and pH prior to its release.

Bases:

Experience with the electronic reliability and calibration stability of the units used by the Rhode Island Nuclear Science Center Reactor demonstrates that the above periods are reasonable surveillance frequencies.

4.8 Surveillance of Experiments

Applicability:

This specification applies to the surveillance of experiments and the limitations on experiments as described in Technical Specification 3.8.

Objective:

To assure that the experiments and their limitations are reviewed with respect to 10CFR50.59 for reactor operation and personnel safety and prevent release of radioactive materials in excess of 10CFR20.

Specification:

Experiments shall be reviewed, approved and properly installed and operational in accordance with written operating procedures.

Experiments in progress shall undergo a review annually.

Bases:

Review of the experiments using the appropriate LCO's and the Administrative Controls assures that the insertion of experiments will not negate the consideration implicit in the Safety Limits.

4.9 Reactor Core Components

Applicability:

This specification applies to the surveillance requirements for reactor core components affecting reactor power.

a. Beryllium Reflectors

Applicability:

This specification applies to the surveillance of beryllium lifetime for the standard and plug type beryllium reflectors.

Objective:

To prevent physical damage to the beryllium reflectors in the core from accumulated neutron flux exposure.

Specification:

The maximum accumulated neutron flux shall be 1×10^{22} neutrons/cm^2. The exposure shall be determined annually in accordance with the operating procedures. Inspections and core fit shall be conducted annually.

Bases:

The RINSC SAR (Part A Section VIII) has addressed this limit as a conservative limit. (Annual inspections and core box fit as well as calculated total exposure serve as a method to monitor the beryllium lifetime.)

b. LEU Fuel Elements

Applicability:

This specification applies to surveillance of LEU fuel elements.

Objective:

To prevent operation with damaged fuel elements and verify the physical condition of the fuel element.

Specification:

The fuel elements shall be visually examined and functionally fit into the core grid box annually.

Bases:

Fuel elements are initially inspected for manufactured specifications and then inserted into the grid box in accordance with QA/QC program requirements for functional fit. Core reloading is performed in accordance with operating procedures. Routine fuel movements are logged and visual inspections are conducted during fuel movements. Pool sampling also is used to detect a ruptured element (Tech. Spec. 4.3.3). The fission density limit for this reactor cannot be exceeded (reference SAR, Part A, Section VI). Burnup calculations are made quarterly (4.9.1).

5.0 DESIGN FEATURES

The basic design features of the facility are described in "General Electric's Operation and Maintenance Manual-GEI-77793", Oct. 1962, also in the "Safety Analysis Report for the Low Enriched Fuel Conversion of the Rhode Island Nuclear Science Center Research Reactor", Rev. 1, 1992. These documents are on file at the Science Center. A general description of the important components is included in the following sections.

5.1 Description

The reactor is located at the Rhode Island Nuclear Science Center on 3 acres of a 27-acre former military reservation, originally called Fort Kearney and now called the Narragansett Bay Campus of the University of Rhode Island. The 27-acre reservation is controlled by the State of Rhode Island through the University of Rhode Island. The reservation is in the Town of Narragansett, Rhode Island, on the west shore of Narragansett Bay, approximately 22 miles south of Providence, Rhode Island, approximately six miles north of the entrance of the Bay from the Atlantic Ocean. The Rhode Island Nuclear Science Center and various buildings used for research, education and training purposes are located on this 27-acre campus.

5.2 Reactor Fuel

The fuel assemblies shall be of the MTR type, consisting of plates containing uranium silicide fuel enriched to less than 20% in the isotope U-235 clad with aluminum. Each fuel element will contain 22 plates for a total [redacted] of U-235 per element.

5.3 Reactor Core

The reactor core consists of a 9 x 7 array of 3 inch square modules with the 4 corners occupied by posts. The reference core for these technical specifications consists of 14 standard LEU fuel elements arranged symmetrically within 4 safety control blades as shown in Figure 4 of the SAR (Revision 1, Section V, Dec. 1992) as approved by the NRC in the conversion order (letter of March 17, 1992).

5.4 Reactor Building

The reactor shall be housed in a building capable of meeting the following functional requirements:

In the event of an accident which could involve the release of radioactive material, the confinement building air shall be exhausted through a clean-up system and stack creating a flow of air into the building with a negative differential pressure between the building and the outside atmosphere. The building shall be gas tight in the sense that a negative differential pressure can be maintained dynamically with all gas leaks occurring inward. The confinement and cleanup systems shall become operative when a building evacuation button is pressed. This action shall: (1) turn off all ventilation fans and the air conditioner system and (2) close the dampers on the ventilation intake and exhaust, other than those which are a part of the clean-up system. No further action shall be required to establish confinement and place the clean-up system in operation. An auxiliary electrical power system shall be provided at the site to insure the availability of power to operate the clean-up system.

The reactor building exhaust blower operates in conjunction with additional exhaust blower(s) which provide dilution air from non-reactor building sources.

Upon activation, the clean-up system shall exhaust air from the reactor building through a filter and a 115 foot high stack, creating a pressure less than atmospheric pressure. The clean-up filter shall contain a roughing

filter, an absolute particulate filter, a charcoal filter for removing radioiodine and an absolute filter for removing charcoal dust which may be contaminated with radioiodine. Each absolute filter cartridge shall be individually tested and certified by the manufacturer to have an efficiency of not less than 99.97% when tested with 0.3 micron diameter dioctylphthalate smoke. The minimum removal efficiency of the charcoal filters shall be 99%, based on ORNL data and measurements performed locally.

Gases from the beam ports, thermal column, pneumatic system, and all other radioactive gas exhaust points shall be exhausted to the stack through a roughing and absolute filter system.

5.5 Fuel Storage

All reactor fuel element storage facilities shall be designed in geometrical configuration where k_{eff} is less than 0.8 under flooding with water. A maximum of four fuel elements will be stored in the fuel safe with no two elements in adjacent positions in the storage box. The adjacent row will be an empty box. Irradiated fuel is stored in the underwater storage racks as described in the SAR (Part A, Section XII).

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization and Management

1. The Rhode Island Atomic Energy Commission (RIAEC) shall have the responsibility for the safe operation of the reactor. The organization of RIAEC is shown in Figure 6-1. The RIAEC shall appoint a Director and a Nuclear and Radiation Safety Committee (NRSC) consisting of a minimum of seven members, as follows:

    a. The Director, RIAEC (Non Voting)

    b. The Assistant Director for Reactor Operations

    c. The Radiation Safety Officer (Non Voting)

    d. A qualified representative from the faculty of Brown University

    e. A qualified representative from the faculty of Providence College

    f. A qualified representative from the faculty of the University of Rhode Island

    g. A qualified representative from the academic community of the State of Rhode Island

    A qualified alternate may serve in lieu of one of the above. The Director, Assistant Director and Radiation Safety Officer are not eligible for Chairperson of the Committee.

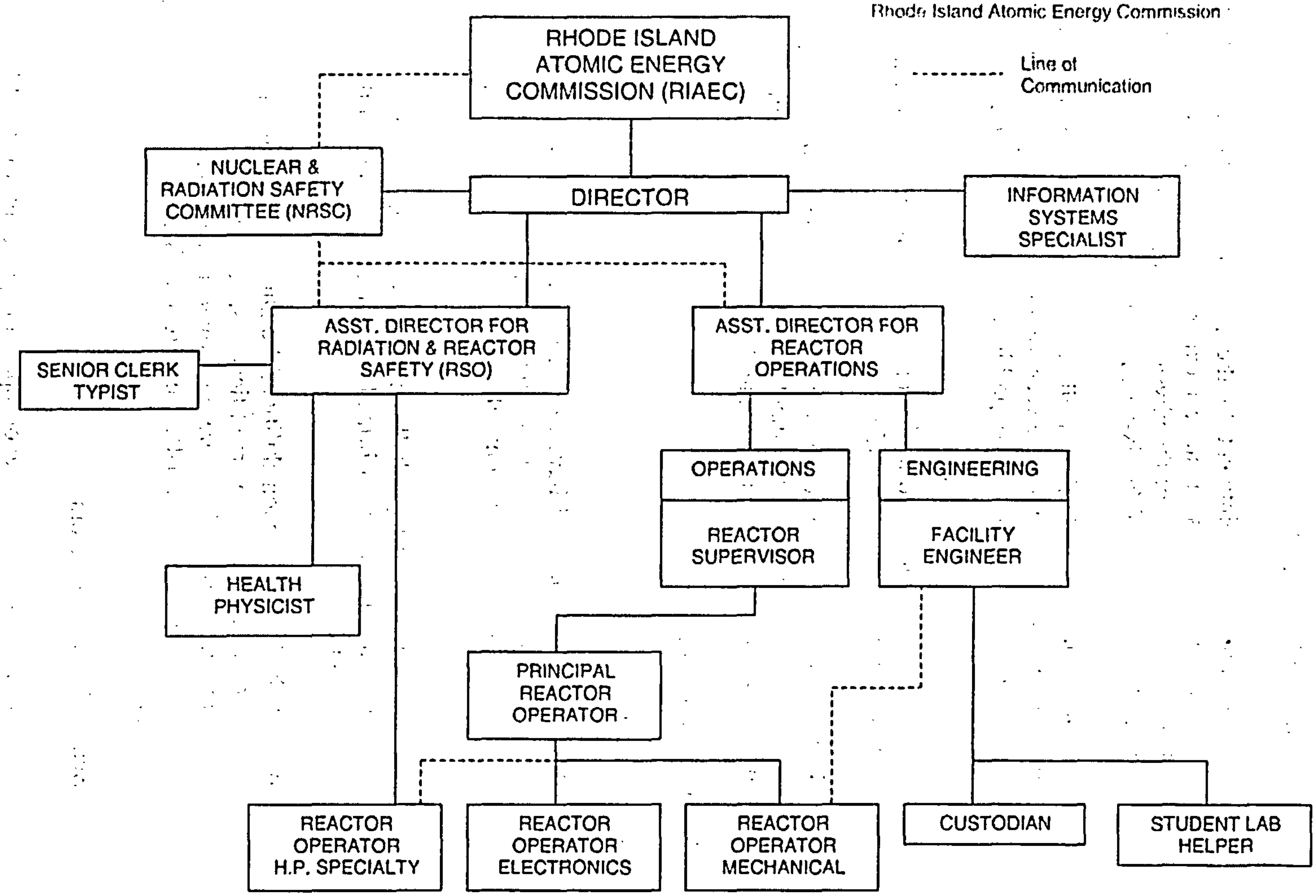
2. An operator or senior operator licensed pursuant to 10CFR55 shall be present in the control room unless the reactor is secured as defined in these specifications. The minimum operating crew shall be two individuals.

3. A licensed senior operator shall be on duty or readily available on call whenever the reactor is in operation.

FIGURE 6.1 - ORGANIZATIONAL CHART

RIAEC Organization Chart
FIGURE 6-1

TECHNICAL SPECIFICATIONS
Docket 50-193 License R-95
Rhode Island Atomic Energy Commission



4. In accordance with the emergency plan, a list of emergency personnel, management and offsite agencies is posted in the control room.

6.2 Qualifications of Personnel

1. At the time of appointment to the position, the Director shall have a minimum of six years of nuclear experience. The Director shall have an advanced degree in one of the physical sciences or engineering. The degree will fulfill four years of the six-year requirement.

2. The Radiation Safety Officer shall have a master's degree in health physics or radiological health and three years of applied health physics experience in a program with radiation safety problems similar to those in the program to be managed.

3. The reactor operators and senior operators shall be licensed in accordance with the provisions of 10CFR55.

4. In the event, of a temporary vacancy in the position of Director or the Radiation Safety Officer, the functions of that position shall be assumed by qualified alternates appointed by the RIAEC.

6.3 Responsibilities of Personnel

1. Director

a. The Director shall have responsibility for all activities in the reactor facility which may affect reactor operations or involve radiation hazards, including controlling the admission of personnel to the building.

   This responsibility shall encompass administrative control of all experiments being performed in the facility including those of outside agencies.

b. It shall be the responsibility of the Director to insure that all proposed experiments, design modifications, or changes in operating and emergency procedures are performed in accordance with the license. Where uncertainty exists, the Director shall refer the decision to the NRSC.

2. Senior Reactor Operators

a. A licensed senior reactor operator pursuant to 10CFR55 shall be assigned each shift and be responsible for all activities during his shift which may affect reactor operation or involve radiation hazards. The reactor operators on duty shall be responsible directly to the senior operator.

b. The identity of and method for rapidly contacting the on-call senior reactor operator shall be known to the reactor operator on duty. The on-call senior reactor operator must be capable of being contacted by the duty reactor operator within ten minutes. The senior reactor operator shall be present at the facility during initial startup and approach to power, recovery from an unplanned or unscheduled shutdown or significant reduction in power, and refueling. The name of the person serving as senior reactor operator as well as the time he/she assumes the duty shall be entered in the reactor log. When the senior operator is relieved, he/she shall turn the operation duties over to another licensed senior operator.

In such instances, the change of duty shall be logged and shall be definite, clear, and explicit. The senior reactor operator being relieved of his duty shall insure that all pertinent information is logged. The senior reactor operator assuming duty shall check the log for information or instructions.

3. Reactor Operators

a. The responsible senior reactor operator shall pursuant to 10CFR55 designate for his shift a licensed operator (hereafter called "operator") who shall have primary responsibility under the senior reactor operator for the operation of the reactor and all associated control and safety devices, the proper functioning of which is essential to the safety of the reactor or personnel in the facility. The operator shall be responsible directly to the senior reactor operator.

b. Only one operator shall have the above duty at any given time. Each operator shall enter in the reactor log the date and time he/she assumed duty.

c. When operations are performed which may affect core reactivity, a licensed operator shall be stationed in the control room. When it is necessary for him/her to leave the control room during such an operation, he/she shall turn the reactor and the reactor controls over to a designated relief, who shall also be a licensed operator. In such instances, the change of duty shall be definite, clear, and explicit. The relief shall acknowledge his entry on duty by proper notation in the reactor log.

d. The operator, under the senior reactor operator on duty, shall be responsible for the operation of the reactor according to the approved operating procedures.

e. The operator shall be authorized at any time to reduce the power of the reactor or to scram the reactor without reference to higher authority, when in his judgment such action appears advisable or necessary for the safety of the reactor, related equipment, or personnel. Any person working on the reactor bridge shall be similarly authorized to scram the reactor by pressing a scram button located on the bridge.

4. Radiation Safety Officer

The Radiation Safety Officer shall be responsible for assuring that adequate radiation monitoring and control are in effect to prevent undue exposure of individuals to radiation.

6.4 Review and Audit

1. The NRSC shall review reactor operations to assure that the facility is operated in a manner consistent with public safety and within the terms of the facility license.

2. The responsibilities of the NRSC include, but are not limited to, the following:

   a. Audit of operating, and emergency procedures and records.

   b. Review and audit of proposed tests and experiments utilizing the reactor facilities.

   c. Review and audit of proposed changes to the facility systems or equipment, procedures, and operations.

   d. Determination of whether a proposed change, test, or experiment would constitute an unreviewed safety question or which may require a change to the Technical Specifications or facility license.

   e. Review of all violations of the Technical Specifications and Nuclear Regulatory Commission Regulations, and significant violations of internal rules or procedures, with recommendations for corrective action to prevent recurrence.

   f. Review of the qualifications and competency of the operating organization to assure retention of staff quality.

   g. Review changes to the NRSC charter.

   h. Review, at least annually, the radiation safety aspects of the facility.

3. The NRSC shall have written direction defining such matters as the authority of the Committee, the subjects within its purview, and other such administrative provisions as are required for effective functioning of the Committee. Minutes of all meetings of the Committee shall be kept. All minutes of the previous Reactor Utilization Committee shall be retained for the life of the facility.

4. A quorum of the NRSC shall consist of not less than four (4) members and shall include the Radiation Safety Officer or designee, the Director or the Assistant Director for Operations and the Chairperson or designee.

5. The NRSC shall meet at least annually.

6.5 Operating Procedures

Written procedures, reviewed and approved by the NRSC, shall be used for items 1-9 listed below. The procedures shall be adequate to assure the safe operation of the reactor, but should not preclude the use of independent judgment and action should the situation require such.

1. Startup, operation and shutdown of the reactor;

2. Installation and removal of fuel elements, control blades and incore devices where necessary;

3 Maintenance procedures which could have an effect on reactor safety;

4. Periodic surveillance of reactor instrumentation and safety systems, area monitors, and continuous air monitors;

5. Implementation of the physical Security Plan and Emergency Plan;

6. Radiation control procedures;

7. Receipt, inspection, and storage of new fuel elements;

8. Storage and shipment of irradiated fuel elements.

9. Experiment review on a case-by-case basis assuring that section 3.8.3(2) of ANSI/ANS 15.1 is satisfied. Operational approval shall be by written approval by a licensed senior operator. Written procedures should be established and supervision of the installation of such experiments shall be defined and exercised.

Substantive changes to the above procedures shall be made only with the approval of the NRSC. Temporary changes to the procedures that do not change their original intent may be made by a Senior Operator. Temporary changes to procedures shall be documented and subsequently reviewed by the NRSC Subcommittee.

6.6 Action to be Taken in the Event of a Reportable Occurrence

In the event of a reportable occurrence:

1. The Senior Reactor Operator shall be notified promptly and corrective action shall be taken immediately to place the facility in a safe condition until the cause of the reportable occurrence is determined and corrected.

2. The Director shall report the occurrence to the NRSC and RIAEC. The report shall include an analysis of the cause of the occurrence, corrective actions taken, and recommendations for appropriate action to prevent or reduce the probability of a repetition of the occurrence.

3. The NRSC shall review the report and the corrective actions taken.

4. Notification shall be made to the NRC in accordance with Paragraph 6.8 of these specifications.

6.7 Action to be Taken in the Event a Safety Limit is Exceeded

In the event a Safety Limit has been exceeded:

1. The reactor will be shut down and reactor operations will not be resumed until authorization is obtained from the NRC.

2. Immediate notification shall be made to the NRC in accordance with paragraph 6.8 of these specifications and to the Director.

3. A prompt report shall be prepared by the Senior Reactor Operator. The report shall include a complete analysis of the causes of the event and the extent of possible damage together with recommendations to prevent or reduce the probability of recurrence. This report shall be submitted to the NRSC for review and appropriate action, and a suitable similar report shall be submitted to the NRC in accordance with Paragraph 6.8 of these specifications and in support of a request for authorization for resumption of operations.

6.8 Reporting Requirements

In addition to the requirements of applicable regulations, all written reports shall be sent to the U. S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, DC 20555, with a copy to the Region I Administrator. The written reports include the following:

1. Within 24 hours, a report by telephone through the NRC Operations Center, Washington, DC and the NRC Region 1:

   a. Any accidental release of radioactivity to unrestricted areas above permissible limits, whether or not the release resulted in property damage, personal injury or exposure.

   b. Any significant variation of measured values from a corresponding predicted or previously measured value of safety related operating characteristics occurring during operation of the reactor.

   c. Any reportable occurrences as defined in Paragraph 1.25 of these specifications.

   d. Any violation of a Safety Limit.

   e. Discovery of any substantial variance from performance specifications contained in the technical specifications and safety analysis.

2. The written report shall be sent within 14 days. The report shall:

   a. Describe, analyze, and evaluate safety implications;

   b. Outline the measures taken to assure that the cause of the condition is determined;

   c. Indicate the corrective action taken, including any changes made to the procedures and to the quality assurance program, to prevent repetition of the occurrence and of similar occurrences involving similar components or systems;

   d. Evaluate the safety implication of the incident in light of the cumulative experience obtained from the record of previous failure and malfunctions of similar systems and components.

3. Unusual Events

   A written report shall be forwarded within thirty(30) days in the event of:

   a. Discovery of any substantial errors in the transient or accident analyses or in the methods used for such analyses, as described in the safety analysis or in the bases for the technical specifications;

Discovery of any condition involving a possible single failure which, for a system designed against assumed failure, could result in a loss of the capability of the system to perform its safety function;

b. Permanent changes in the facility organization involving the Director or Assistant Director.

4. An annual report shall be submitted in writing within 60 days following the 30th of June of each year. The report shall include the following information:

a. Tabulation showing the energy generated by the reactor (in megawatt days), the number of hours the reactor was critical, and the cumulative total energy output since initial criticality.

b. The number of emergency shutdowns and inadvertent scrams, including the reasons.

c. Discussion of the major maintenance operations performed during the period, including the effect, if any, on the safe operation of the reactor, and the reasons for any corrective maintenance required.

d. A description of each change to the facility or procedures, tests, and experiments carried out under the conditions of Section 50.59 of 10CFR50 including a summary of the safety evaluation of each.

e. A description of any environmental surveys performed outside the facility.

f. A summary of annual radiation exposures in excess of 500 mrem received by facility personnel, including the dates and times of significant exposures.

g. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge.

6.9 Plant Operating Records

In addition to the requirements of applicable regulations and in no way substituting therefore, records and logs of the following items, as minimum, shall be kept in a manner convenient for review and shall be retained as indicted:

1. Records to be retained for a period of at least five years:

a. Reactor operations;

b. Principal maintenance activities;

c. Experiments performed including aspects of the experiments which could affect the safety of reactor operation or have radiological safety implications;

d. Reportable occurrences;

e. Equipment and component surveillance activities;

f. Facility radiation monitoring surveys;

g. Fuel inventories and transfers; and

h. Changes to procedures systems, components, and equipment.

2. Records to be retained for the life of the facility:

a. Gaseous and liquid radioactive effluents released to the environs;

b. Off-site environmental monitoring surveys;

c. Personnel radiation exposures;

d. Updated, "as-built" drawings of the facility; and

e. Minutes of the NRSC (and previous Reactor Utilization Committee) meetings.

CHAPTER FIFTEEN

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CHAPTER 15

15.0 Financial Qualifications

This chapter describes and discusses the financial information submitted to the NRC for a non-power reactor license to establish that the applicant is financially qualified to own, construct, operate, and decommission a non-power reactor. Financial qualifications cover three areas:

1. Financial ability to construct the non-power reactor facility authorized by the construction permit.
2. Financial ability to safety operate the facility.
3. Financial ability to safety decommission the facility so that the NRC can terminate the facility license at the end of the facility's use.

15.1 Financial Ability to Construct a Non-Power Reactor

The Rhode Island Atomic Energy Commission was authorized to construct a research reactor by Rhode Island General Law, Title 42, Chapter 142, using state bonds. These bonds were paid off prior to 1984. The land is leased from the Rhode Island Higher Education Board of Governors. The reactor, adjoining offices and laboratories are owned by the State of Rhode Island under the control of the Rhode Island Atomic Energy Commission. The Commission was established pursuant to Title 42, Chapter 27 of the General Laws of Rhode Island. Section 42-27-2 authorizes the Commission to construct and operate a research reactor. Section 42-27-4 authorized the General Assembly to annually appropriate a sum sufficient for the commission to carry out the purposes of the Act. The Act providing for the establishment of the reactor is contained in Chapter 142, Laws of 1958.

Fuel cycle costs have been historically covered by Department of Energy (DOE) Fuel Assistance Grants. DOE also funded the conversion to LEU fuel. Given the fact that there is currently no inventory of spent fuel, it will be several years before a shipment is required. Also, a significant amount of new fuel is being held at the vendor for future use. This fuel became available when the only other research reactor with 22 plates per element LEU fuel ceased operation. State funding will cover any fuel cycle costs not covered by DOE.

15.2 Financial Ability to Operate a Non-Power Reactor

Under Section 42-27-2 of Rhode Island General Law, the Rhode Island Atomic Energy Commission is directed to operate a nuclear reactor within the state for the purpose of research, experimentation, training of personnel, testing of materials and techniques and for such other purposes related thereto which the Commission shall deem necessary for the health, welfare, and economy of the people of the state; and in this respect to cooperate with and make available, under proper safeguards, the use of the reactor by colleges, universities, and industries of this state. The Commission may charge fees for the use of the reactor facilities. Historically, the Commission has waived or reduced fees associated with research grants in order to make them more competitive. Tours and training have been conducted at no cost to the citizens of the state. Due to the small staff size, only limited commercial activities associated with the reactor have been permitted and the fees charged for these activities have been utilized to defray additional costs associated with reactor operation.

The RIAEC budget is funded by an annual appropriation from the state, which is part of the state's annual budget as enacted by the legislature and approved by the governor. Any funds generated from commercial activities are deposited into the state's general funds for use by the legislature. Under Rhode Island General Law 42-27-4, the state will meet its obligation to adequately support safe reactor operation.

Historically, approximately 80% of the State supported budget has been utilized for staff salaries. The remaining 20% has been utilized to fund operations and maintenance. Each year, the budget fluctuates based on grant money received and capital budget projects. These two items are designated for specific purposes and are excluded from the calculation of funds required for continued reactor operation.

In fiscal year 1999, the Commission spent $506,926 on personnel and $140,063 for operations. In fiscal year 2000, the Commission spent $591,661 on personnel and $91,661 on operations. In fiscal year 2001, the state enacted legislation for $638,794 for personnel and $103,661 for operations. These figures are driven by several factors, which will determine future funding levels. On the personnel side, union negotiated cost of living increases usually average around 3-4%. On the operating side, energy costs are the major variable besides the general inflation rate. It is therefore reasonable to assume that the operating cost of the reactor will increase by 3-4% per year over the next several years.

15.3 Financial Ability to Decommission the Facility

Non-power reactor licensees using a statement of intent, may be Federal, State, or local government entities. The chairman of the Rhode Island Atomic Energy Commission has signed a copy of the statement of intent regarding decommissioning of the Rhode Island Nuclear Science Center reactor, which is maintained in the licensee's records and is available for inspection.

While there are no current plans to decommission the reactor, the state has required that this project be identified in the State's Annual Capital Budget for liability purposes. The decommissioning project estimate is reviewed annually by the House and Senate Finance Committees and is part of the Capital Budget that is signed into law by the Governor. It is therefore prudent to consider that there is adequate assurance that funds will be available for decommissioning when that decision is made.

15.4 References

a. Rhode Island State General Laws

b. 10 CFR Part 50 "*Domestic Licensing of Production and Utilization Facilities: NRC Regulatory Guide DG-1106* "Assuring the Availability of Funds for Decommissioning Nuclear Reactors"

c. Rhode Island State Budget – RI Atomic Energy Commission

CHAPTER 16

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Chapter 16

16.0 Other License Considerations

16.1 Prior Use of Reactor Components

There are no components in use at the Rhode Island Nuclear Science Center that have had prior use at any other facility or organization. It is conceivable that prior use components could be integrated into the reactor systems at some future time. Appropriate analysis and reviews of component replacement will be conducted in accordance to applicable standards, regulations and facility procedures and licensed technical specifications.

16.0 Medical Use of Non-Power Reactors

The Rhode Island Nuclear Science Center is not engaged nor licensed to conduct any activities for medical use of the facility. Future medical use of the RINSC would be conducted pursuant to appropriate license applications and approvals as authorized by the Atomic Energy Act of 1954 as amended.