STATE OF RHODE ISLAND AND PROVIDENCE PLANTATIONS

Rhode Island Atomic Energy Commission
NUCLEAR SCIENCE CENTER
South Ferry Road
Narragansett, R.I. 02882-1197

July 17, 1992

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Gentlemen:

Enclosure (1), "Safety Analysis Report for the Low Enriched Fuel Conversion of the Rhode Island Nuclear Science Center Research Reactor" dated November 1991, is forwarded per conversation between Mr. Marvin Mendonca (NRC) and Mr. Terry Tehan (RIAEC) as a description of the Low Enriched Fuel Conversion project.

Very truly yours,

Terry Tehan

Terry Tehan
Director, RIEAC

TT:cd

Enclosure (1)

Cert No P389790228

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1/1

STATE OF RHODE ISLAND AND PROVIDENCE PLANTATIONS

RHODE ISLAND ATOMIC ENERGY COMMISSION
Nuclear Science Center
South Ferry Road
Narragansett, R.I. 02882-1197

"SAFETY ANALYSIS REPORT FOR THE LOW ENRICHED FUEL CONVERSION OF THE RHODE ISLAND NUCLEAR SCIENCE CENTER RESEARCH REACTOR"

NOVEMBER, 1991

SAFETY ANALYSIS REPORT

PART A LEU CONVERSION ANALYSIS PAGE(S)

Part A LEU Conversion Analysis

INTRODUCTION

This safety analysis report is submitted pursuant to 10CFR 50.64 which requires the Rhode Island Atomic Energy Commission to convert its open pool research reactor from the use of high enriched uranium (HEU) fuel to the use of low enriched (LEU) fuel. The studies 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.

The operating license for this reactor was issued on July 21, 1964 with an expiration date of August 27, 2002. The original license permitted operation at a power level of 1 MW. An amendment to the license was issued on September 12, 1988 and permitted operation as 2 MW. Since that time the reactor has operated at 2 MW.

The reactor is multipurpose with capabilities usually associated with open pool facilities. Because of staffing and funding limitations, utilization has concentrated in two areas-neutron scattering and neutron activation analysis. To meet the needs of the research programs, the reactor operates one shift, five days per week. As of September 1, 1991, the accumulated operation of the reactor was 47066.3 megawatt-hours. This operation has required the use of HEU fuel elements distributed as follows:

returned to reprocessing	110
spent, awaiting shipment to reprocessing	26
currently in use	35
new, available for use	13

There are no plans to change this duty cycle. This duty cycle allows for operation with an excess reactivity less than that required for continuous operation. Because of the control blade configuration, this duty cycle also requires special start-up considerations when converting to an LEU core.

The studies performed for the LEU conversion have included calculations for operation at power levels above 2 MW and for advanced core designs. This was done to insure that the conversion process did not compromise the future capabilities of the reactor. This safety analysis report however contains only information necessary for the initial conversion to LEU.

DESCRIPTION OF REACTOR SYSTEMS

The reactor system is described in the initial safety analysis report /1/. Only a synopsis of system components important to the conversion will be presented here.

The reactor core sits on a 7x9 grid plate with the four corner grid positions occupied by the suspension frame corner posts. These corner posts connect the grid plate to the reactor bridge which spans the open pool. The hollow corner posts each contain a neutron detector required for the operation of the reactor. The grid plate is suspended about 8 meters (26.33 feet) below the pool water surface.

This grid plate is installed near the bottom of a grid box whose four sides are enclosed, top is open to the pool and bottom connects to an enclosed plenum for coolant flow. The grid box also contains two permanently installed shrouds in which four boral control-safety blades (rods) move. This arrangement is shown in Figure 1.

The grid location of the four boral control-safety blades cannot be moved. The boral regulating rod, however, while fixed in the reflector region of the HEU core, can be relocated. While some grid positions are shown vacant for clarity, during operation each grid position must contain a fuel element, a reflector piece, an irradiation basket, or a plug. Otherwise the coolant flow will by-pass the core through the vacant grid position.

The HEU fuel element consists of 18 flat aluminium plates with a thickness of 1.52 mm (.060 inches). The fuel meat is

0.508 mm (.020 inches) and consists of 93% enriched uranium in a UAlx matrix. The clad is 0.508 mm (.020 inches) and consists of aluminum. The spacing between fuel plates is 2.54 mm (0.1 inches). When new, each plate contains 6.889 grams of Uranium 235 for a total Uranium 235 element weight of 124 gm.

The operating HEU core is made up of these fuel elements and consists of between 28 and 35 elements surrounded on four sides by graphite reflector pieces. This core may be characterized as large with a very low power density resulting in a low thermal flux per unit power. The lightly loaded fuel elements makes the core large enough to encompass the fixed control blades. Even with extra ordinary techniques, the maximum burn-up achievable is about 14%. Positions in the grid plate not containing a fuel element or a reflector piece are filled with an irradiation basket. Figure 2 presents a typical 30 element HEU core.

Operation is also permissible with water reflection of the reactor. During the initial startup of the reactor, many measurements were made of the characteristics of a water reflected core. However, such a core has never been operated above 100 KW because of the requirement of the experimental program and the lack of sufficient irradiation baskets to plug grid positions vacated by graphite reflectors. (Operation using natural convection cooling is limited to a maximum power level of 100 KW).

The core may be positioned anywhere on the center line of a three section, interconnected pool. Operation using forced convection is only possible in the circular end section where a connection can be made to cooling pipes. In this high power section, the core neutrons are available to six radial beam tubes (three in use), a through tube, a graphite thermal column through which a hole has been cut creating an additional radial beam tube, and the terminals of two pneumatic irradiation (rabbit) systems. Between the graphite thermal column and the core is a permanently installed slab of lead serving as a thermal shield. The thermal shield is cooled by water which is currently forced around the shield using the pressure difference between the inlet and outlet primary coolant lines.

The control of make-up water to the pool is automatic using a float activated motorized valve. This system activates for a drop in pool level of 2.54 cm (1 inch). For a drop of 5.08 cm (2 inches), the reactor, if running, will scram.

The neutron detectors in each corner post provide signals to the control and safety system. Although most of this system is the original equipment, it has been well maintained and is reliable.

Using a primary pump the core is cooled at 2 megawatts by downward flow of 0.109 m^3/sec (1730 gpm). Using a stainless steel heat exchanger, the heat in this primary water is transferred to a secondary cooling system operating with a nominal flow of .0631 m^3/sec (1000 gpm) and using a forced draft cooling tower.

The reactor is housed in a semi gas-tight, windowless building which uses the confinement concept for the controlled release of radioactivity in the unlikely event of a reactor accident. The controlled release is produced by a blower and is through HEPA and charcoal filters and a stack. The release creates a negative pressure differential between the atmosphere and the building insuring that leaks through the building are inward.

During the initial start-up phase for the reactor, criticality determinations were made for 17 graphite and water reflected cores. Excess reactivity measurements were made as the core size increased towards the operating core. In addition, control blade calibrations and the core thermal flux distribution were experimentally determined.

CONVERSION CRITERIA AND OBJECTIVES

There are six basic criteria and objectives of the LEU conversion program. These are:

1. Convert the reactor to the use of LEU using the standard fuel plate which will be provided to university research reactors by the U. S. Department of Energy.
2. Design a LEU core and an operating scheme to achieve burn-up greater than the current 14%.
3. Design an LEU core which will optimize the thermal neutron flux in the beam tubes and will allow for further improvement.
4. Design a reactor core with a flux trap for small sample irradiation.
5. Design a core which does not preclude future operation at power levels up to 5 MW with the appropriate primary coolant flow.
6. Design a LEU core whose initial cost is about the same as the cost of 30 HEU fuel elements since this is the amount allocated for the core by the U. S. Department of Energy.

LEU NEUTRONIC CORE DESIGN

The neutron core design has been performed using the standard fuel plate which the Department of Energy will provide for university reactors. Figure 3 presents a comparison of this standard LEU plate with the current HEU plate. Also shown are the characteristics of a LEU direct replacement plate. The standard plate is thinner and contains more Uranium-235 than the HEU or direct replacement LEU plate.

The not-readily movable control safety blades are an important consideration for LEU neutronic core design. Because of the more heavily loaded standard LEU fuel plate, the core may become so small that the control blade looses effectiveness.

During extensive scoping studies, many core configurations were considered /2/. These studies included consideration of:

a. 18 fuel plate elements
b. 22 fuel plate elements
c. several fuel element arrangements

d. graphite and beryllium reflectors

e. relocation of the regulating rod position, if necessary

f. use of stainless steel as the regulating rod.

The neutronic calculations have been performed by Argonne National Laboratory using the EPRI Cell, DIF 3D, and VIM Monte Carlo Codes. Incorporating all of the information gathered during these scoping studies and remembering the six conversion criteria and objectives, a LEU conversion design has emerged.

LEU CONVERSION CORE

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 presents the startup version of the conversion core which consists of 14 fuel elements. The elements contain a total of [illegible]75 grams of U-235 each. A central beryllium piece with a 38mm hole is incorporated as a flux trap. The regulating rod is stainless steel and has been moved one grid position so as to be adjacent to the compact core. The LEU fuel used is the uranium silicide-aluminum dispersion fuel approved for use by the NRC under NUREG-1313.

Table 3 presents reactivity data on this core. The core is graphite and beryllium reflected with an excess reactivity of 3.1% a regulating rod worth of 0.45%, a shutdown margin with blade 3 struck out of 6.7% and a total power peaking factor of 2.64. This design allows the use of existing graphite reflectors along wit[illegible]ly acquired beryllium reflectors.

Because of the [illegible]e shift operator, the xenon behavior of this core is cyclical and this core can be operated as long as it is possible to operate on Friday morning. Using computer simulation, this core has been "run down" until a Friday morning startup is no longer possible. The reactivity balance is shown in Table 2.

The reactivity requirements for Xe, Sm, long lived fission products, control, and the cold-hot swing is approximately 3%

which will allow for approximately 14 weeks of operation before it will not be possible to start up on Friday morning.

After this initial operation, ten beryllium and ten graphite reflector pieces will be reconfigured to provide additional reactivity. Figure 4 also presents this second core showing the fuel remaining in each fuel element after the initial 14 weeks of operation. The reactivity balance is shown in Table 1 and it allows for an additional 70 weeks of operation.

Following this second phase of operation, the graphite and beryllium reflectors will again be reconfigured. This third core is shown in Figure 4 which also shows the fuel in each element at the start of this phase. Table 1 again presents the reactivity balance which now allows for an additional 60 weeks of operation.

Note that the core is now almost completely beryllium reflected. The core has operated for about 3 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 flatest flux and greatest neutron leakage. Eventually an equilibrium core will be reached.

Figure 4 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%, which is 50% more burn-up than in the current HEU core. The LEU fuel used is the uranium silicide-aluminum dispersion fuel approved for use by the Nuclear Regulatory Commission under NUREG-1313.

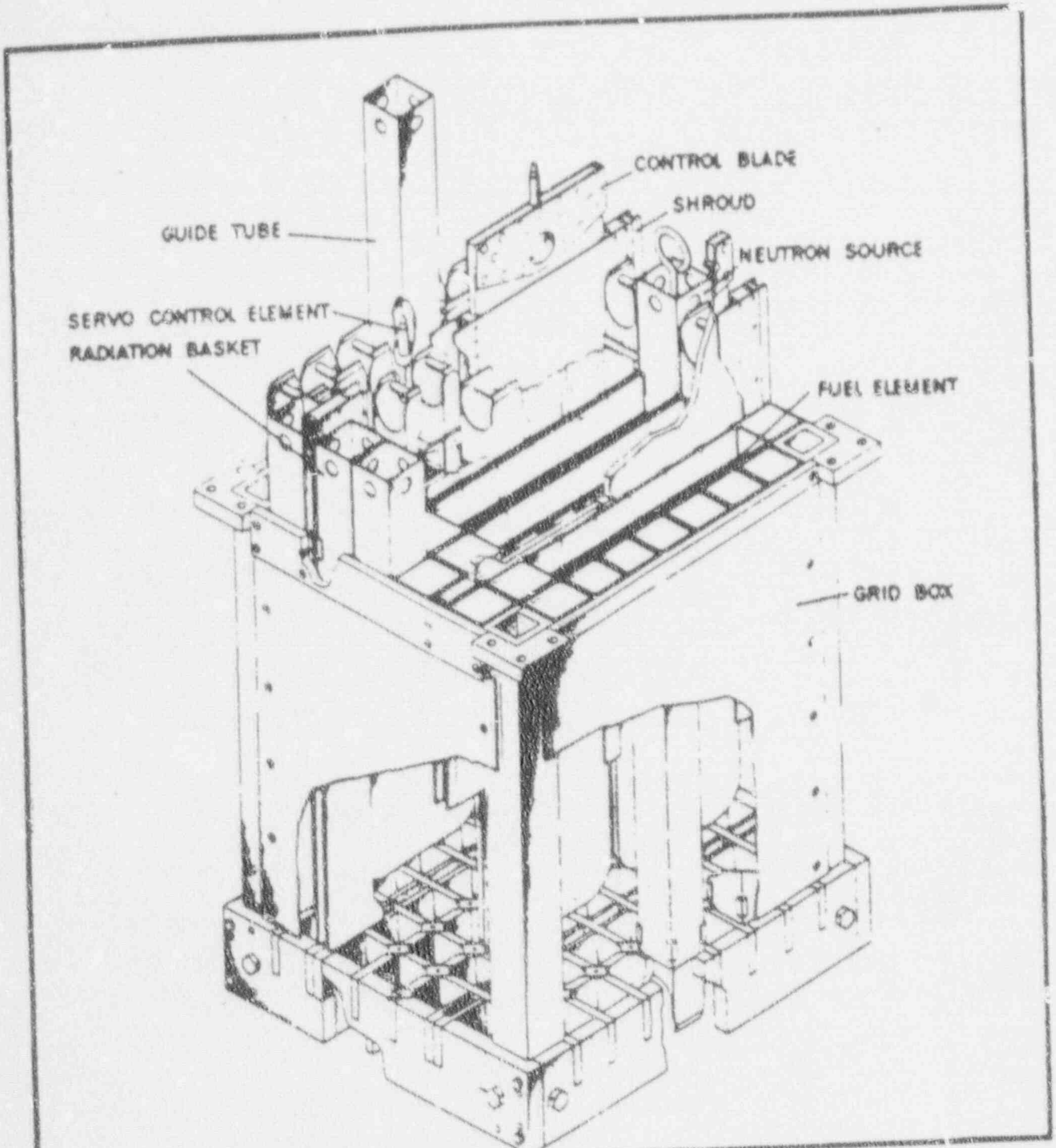

CONTROL BLADE
SHROUD
GUIDE TUBE
NEUTRON SOURCE
SERVO CONTROL ELEMENT
RADIATION BASKET
FUEL ELEMENT
GRID BOX

Figure 1

HEU CORE

February 24, 1986

30 Fuel and 23 Graphite Reflector Elements

Approx. U-235 Loadings, g per Element

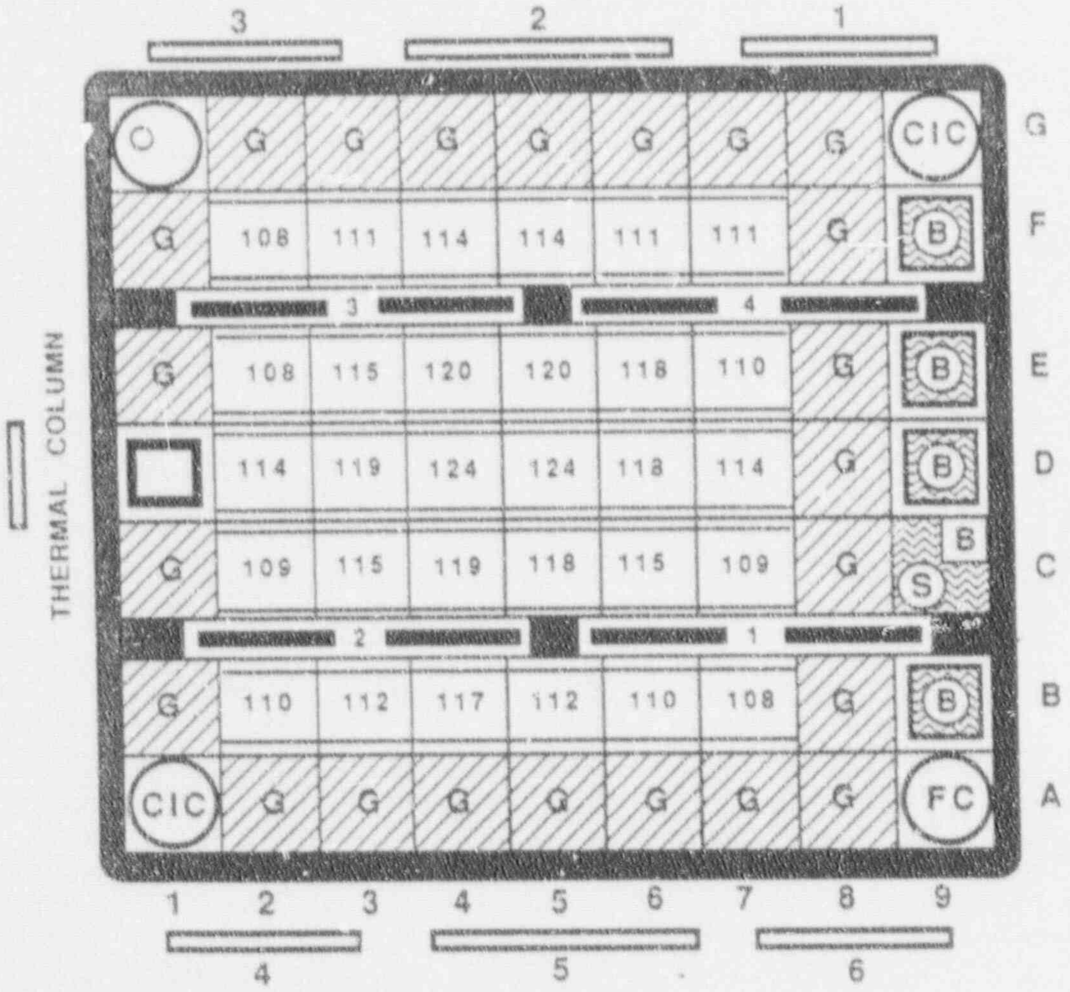


Figure 2

Figure 3 - Description of HEU and LEU Fuel Elements

		HEU	LEU
Number of Fueled Plates/Element		18	22
Fissile Loading/Element, g ^{235}U		124	275
Fuel Meat Composition		UAl_x-Al	U_3Si_2-Al
Cladding Material		1100 Al[1]	6061 Al[2]
Fuel Meat Dimensions[3]			
thickness,	mm	0.508	0.508
width,	mm	52.1 - 61.0	62.7 - 71.1
length,	mm	559 -597	572 - 610
Cladding Thickness,	mm	0.508	0.381

[1] 10 ppm natural boron was added to the composition of the cladding and all fuel element structural materials to represent the alloying materials, boron impurity, and other impurities in the 1100 Al of the HEU elements.

[2] 20 ppm natural boron was added to the composition of the cladding and structural materials of the LEU elements to represent the alloying materials, boron impurity, and other impurities in 6061 Al. Aluminum with no boron or other impurities was used in the fuel meat of both the HEU and LEU elements.

[3] Reference Drawings:

HEU : EG&G #411647
Plate
LEU " EG&G #422873
Plate

Fig. 4. Startup, Transition, and Equilibrium Cores
(Lifetimes Based on Operation for 8 Hr/D, 5 D/Wk)

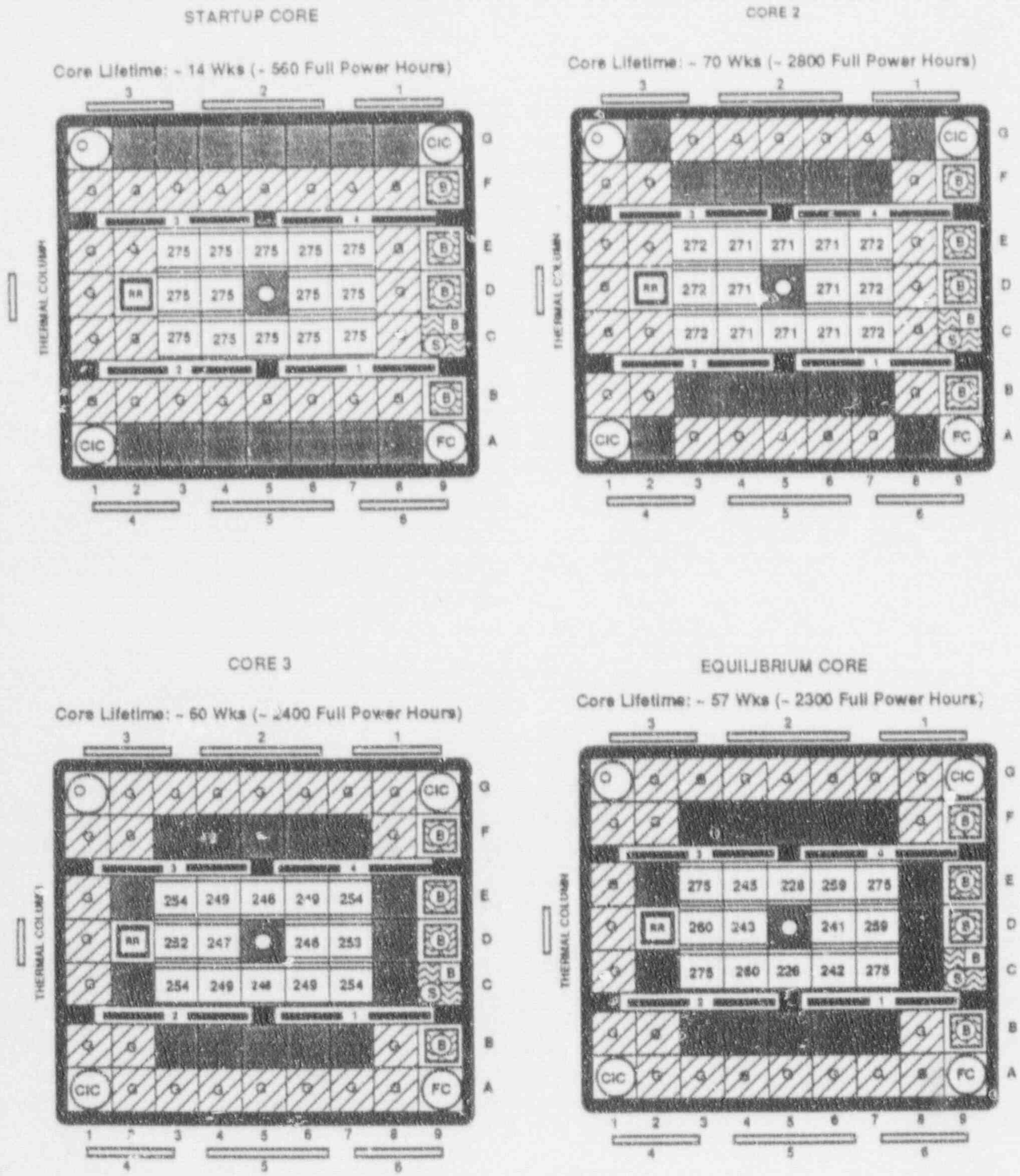


Table 1

Calculated Data and BOC Excess Reactivity for First Ten Cores

	Core Lifetime Weeks	Accum. Weeks	Operation Years	BOC Express %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 10	57	515	9.9	4.0

TABLE 2

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
Xe	1.54	1.54	1.54	1.54	1.54	1.54	1.54	1.54	1.54	1.54
Sm	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
	3.00	5.19	6.91	6.91	6.89	6.92	6.91	6.89	6.90	6.90

Table 3

Reactivity Data and Power Peaking Factors

	Start-up Core	Core 2	Core 3	Core 10
Excess Reactivity %Δk/k	3.1	4.1	3.7	4.0
Shutdown Margin %Δk/k (Blade 3 stuck out)	6.7	6.1	-	6.4
Worth of Reg rod %Δk/k	.45	.41	-	.47
Total Power Peaking Factor/Grid Position (Control Blades Full Out)	2.64/D6	2.60/D6	-	2.36/D6
Total Power Peaking Factor/Grid Position (Control Blades 50% Inserted)	3.06/D6	3.05/D6	-	2.81/D6

For the LEU cores, additional kinetic parameters and reactivity coefficients were calculated by ANL. The comparisons are shown in Table 4.

Table 4

	HEU Ref. Core	LEU Startup Core	LEU Transition Core 2	LEU Equilib. Core 10
Kinetics Parameters				
Delayed Neutron Fraction, β-eff, %	0.762	0.782	0.776	0.764
Prompt Neutron Generation Time, μs	76.3	66.2	66.0	68.3
Reactivity Coefficients: 20-40°C				
Change in Water Temperature Only %$\Delta k/k \times 10^{-4}/°C$ Coolant	-1.51	-0.80	-0.86	0.89
Change in Water Density Only %$\Delta k/k \times 10^{-4}/°C$ Coolant	-0.44	-0.82	-0.75	-0.69
Coolant Coeff., %$\Delta k/k \times 10^{-4}/°C$ Coolant	-2.0	-1.6	-1.6	-1.6
Doppler Coeff., %$\Delta k/k \times 10^{-4}/°C$ Fuel	0.0	-0.18	-0.18	-0.18
Temperature Coeff*, %$\Delta k/k \times 10^{-4}/°C$	-2.0	-1.8	-1.8	-1.8
%$\Delta k/k$/% Void	0.0015	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}.

It can be seen that there are no significant differences between the HEU and LEU kinetic and reactivity parameters.

START-UP ACCIDENT

This accident was analyzed using a digital computer program PARET/3/. The accident is postulated to proceed under the following assumptions:

i. The reactor is in the cold clean condition with power at source level.

ii. The servo regulating rod is withdrawn, followed by continuous withdrawal of all safety blades in succession at their maximum rate.

iii. Period scram protection fails.

iv. The reactor is scrammed by the high flux sensor instrumentation when the power level reaches 2.4 MW (20% overpower).

v. 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 67.3°C (153°F) for the HEU fuel and 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 assumption "iii" is modified to - "period and high flux scram protection fails" - then reactor power would continue to rise beyond the trip point (2.4 MW) until the negative reactivity introduced by the void and temperature coefficients is greater that the net positive reactivity inserted by blade withdrawal. Table I provides the peak power and the maximum cladding temperature reached in the cladding for both HEU and LEU fuel cases. In each case, the maximum cladding temperature is less that 150°C (302°F) - much lower than the 582°C (1080°F) melting temperature of 6061 cladding. The core in each case would operate in the nucleate boiling range without physical damage until the accident could be terminated by a manual scram.

Table 5

Peak Power and Cladding Temperatures

Case	Peak Power, MW	Peak Cladding Temperature, ºC
HEU Equilibrium	32.1	149.1
LEU Startup	14.9	148.3
LEU Equilibrium	16.2	148.5

REFERENCES

(1) Atomic Energy Commission, Facility License No.R-95, Docket No. 50-193, July 21, 1964 and Construction Permit No. CPRR-73

(2) DiMeglio, A.F., Matos, J.E., Freeese, K.E., and Spring, E.F.: The Conversion of the 2 MW Reactor at the Rhode Island Nuclear Science Center. Proceedings of 1989 International Meeting on Reduced Enrichment for Research and Test Reactors, Berlin, West Germany, September, in press

(3) Obenchain, C.F., "PARET - A Program for the Analysis of Reactor Transients"" IDO-17282 (1969)

REPLACEMENT REGULATING ROD

The current regulating rod in the HEU core is located in the D-1 grid position (refer to Figure 2 in the "Description of the Reactor System" section of this SAR). The rod is fabricated from boral plate and aluminum.

Calculations from ANL[1] indicate that the regulating rod worth in its present core position (D-1) is reduced from its present value of .48% Δk/k to .2-.3% Δk/k which is too little. If the present rod were to be relocated to the D-2 position, it increases to .8-.9% which is too large. The regulating rod limit by technical specification is .6% Δk/k. Therefore a new stainless steel regulating blade is necessary in the D-2 position. Calculations indicate a satisfactory worth of .4-.5% Δk/k results. The new regulating blade will be fabricated with the same dimensions to properly fit the core grid box. All references in this SAR relating to the new LEU cores are made with the new regulating rod as part of the core arrangement.

(1) Memo from James Matos, ANL to RINSC, Eugene Spring, September 16, 1991

USE OF BERYLLIUM REFLECTORS IN THE RINSC-LEU CORE

The proposed use of Beryllium reflectors in the LEU start up and equilibrium cores has been reviewed. The Beryllium reflectors are currently being designed by EG&G Idaho. The University of Missouri Reactor[12] has been using Beryllium and has conducted tests to determine a lifetime limit based upon a fast fluence level. Reference[10] indicates that embrittlement of Beryllium is first noted often approximately $3x10^{21}$ NVT. As a result of HFIR determination of small cracks occurring at a $1.8x10^{(22)}$ NVT level, a proposed changeout level of $1x10^{21}$ is proposed. Using a maximum flux of $3.3x10^{13}$ and a 5 day, 7 hrs/day reactor operating cycle, a proposed changeout of 45.8 years is predicted.

Little change in other Beryllium properties at our operating temperatures and integrated fission neutron dose occurs up to the propose limit.[2][3][4][7][8][9] Gamma heating has been reviewed[6][13] and poses no problem. At present EG&G has reviewed the Beryllium materials available and has developed the specification for use in our element fabrication.[11] Final drawings are due shortly. A standard element is proposed similar to the graphite elements. A special element, with a 1.25cm center hole, will be designed and used as a flux trap for special experiments (see Figure 3). The maximum calculated flux in the Be portion of the flux trap is $3.3x10^{13}$. A removable plug would be used to fill the hole when not in use. The RINSC would require a technical specification change for use of Beryllium reflectors.

Table 5 shows the average midplane flux in the Be of the central flux trap.

Figure 5

Grp No.	Upper Energy	Lower Energy	Startup	Core 2	Core 10
1	14.0 MeV	0.821 MeV	$2.5x10^{13}$	$2.4x10^{13}$	$2.4x10^{13}$
2	0.821 MeV	5.531 MeV	$3.3x10^{13}$	$3.3x10^{13}$	$3.2x10^{13}$
3	5.531 keV	1.855 eV	$2.7x10^{13}$	$2.7x10^{13}$	2.[illegible]
4	1.855 eV	0.625 eV	$3.7x10^{12}$	$3.7x10^{12}$	3.[illegible]
5	0.625 eV	0.251 eV	$3.3x10^{12}$	$3.3x10^{12}$	3.[illegible]
6	0.251 eV	0.057 eV	$1.8x10^{13}$	$1.8x10^{13}$	1.[illegible]
7	0.057 eV	0.00025 eV	$2.5x10^{13}$	$2.5x10^{13}$	2.[illegible]

REFERENCES FOR BERYLLIUM REFLECTOR USE

(1) ASME, 74-PUP-44, "Stress and Deformation Analysis or Irridation Induced Swelling" by B. V. Winkel, 1974

(2) "Surveillance Testing and Property Evaluation of Beryllium in Test Reactors", J.M. Beeston, M.R. Martin, C.R. Brinkman, G.E. Korth, and W.C. Francis, Aerojet Nuclear Company, Idaho Falls, Idaho (U. S. Atomic Energy Commission Idaho Operations Office under contract number AR(10-1)-1375)

(3) The Mechanical Properties of Some Highly Irradiated Beryllium, J.B. Rich, G.P. Walters and R.S. Barnes, Atomic Research Establishment, Metallurgy Division, March 1961

(4) Properties of Irradiated Beryllium Statistical Evaluation J.M. Beeston, EG&G Idaho, October 1976

(5) Missouri University TM-ERS-62-1, MOU-30203, June 22, 1962, "Stress and Thermal Analysis of the Beryllium Reflector for the University of Missouri Reactor"

(6) General Electric Co. Atomic Power Equipment Department Standard 788, "Beryllium, Hot Pressed, Nuclear Grade"

(7) "The Effects of Neutron Irradiation on Beryllium Metal", B.S. Hickman, The Institute of Metals Conference on the Metallurgy of Beryllium, October 1961

(8) "The Effect of High-Temperature Reactor Irradiation on Some Physical and Mechanical Properties of Beryllium, J.R. Weir, The Institute of Metals, Conference on the Metallurgy of Beryllium, October 1961

(9) "The Behavior of Irradiated Beryllium, R.S. Barnes, The Institute of Metals, Conference on the Metallurgy of Beryllium, October 1961

(10) University of Missouri, Inter-Department Correspondence, Gerald Schapper; Beryllium Reflector Changeout, December 16, 1975

(11) EG&G Idaho, Material Specification, Beryllium Pressings and Components for Nuclear Reactors and Reactor Systems, Document No. ANC-80005G, April 26 1978

(12) University of Missouri, Specification Drawing Beryllium Reflector, Drawing No. 193, October 6, 1988

(13) FAX memo from Argonne National Laboratory to Rhode Island Nuclear Science Center, W.L. Woodruff to Eugene Spring; Subject: Gamma Heating in Beryllium Reflectors, March 19, 1991

(14) FAX memo from Argonne National Laboratory to Rhode Island Nuclear Sciences Center, James Matos to Eugene Spring, Subject: Flux in Beryllium, September 19, 1991

DESIGN BASIS ACCIDENT

The design basis accident for this reactor has been a loss of coolant accident with the water draining through a beam port containing no plugs. Recall that the core sits in a grid box and draining of this box is through a 1.25 cm hole drilled in the bottom. Because of this, about 17 minutes is required to complete the draining, after which the bottom 21 cm of fuel remains in water. It has been possible to show that the low power density HEU core will not melt after this hypothetical loss of coolant accident.

The LEU core has a higher power density than the HEU core. Using the same accident sequence and calculations which were used for the HEU core, it is not possible to conclude that the LEU core will not suffer some melting following a loss of coolant accident. The LOCA assumes a gillotine severence of the end of a beam port in the pool with water leaving an open beam port end to the reactor room main floor level. The data, discussions and calculations are shown in the thermal hydraulic section of the report (Part B).

APPENDIX A

LEU FUEL SPECIFICATIONS AND DRAWINGS

EG&G IDAHO INC.

(A) TRTR-6 Specification for Test Research Training Reactor LEU Silicide U_3Si_2 Fuel Plates
Rev. 4, 20 May 1988

(B) TRTR-11 Specification for Low Enriched U Metal for Reactor Fuel Plates
Rev. 1, 1 April 1987

(C) TRTR-14 Specification for Reactor Grade Uranium Silicide U_3Si_2 Powder
Rev. 2, 1 July 1987

(D) TRTR-15 Specification for Aluminum Powder for Fuel Plate Core Matrix
Rev. 2, 1 July 1987

EG&G DRAWINGS

(A) Test Research Training Reactor LEU Fuel Plate
No. 422264

(B) Rhode Island Nuclear Science Center Test Research Training Reactor 5 Fuel Plate
No. 422873

(C) Rhode Island Nuclear Science Center Test Research Training Reactor 5 Side Plate
No. 432325

(D) Rhode Island Nuclear Science Center Test Research Training Reactor 5 End Box
No. 411649

(E) Rhode Island Nuclear Science Center Test Research Training Reactor 5 Fuel Element Assembly
No. 411650

SAFETY ANALYSIS REPORT

Part B Thermal Hydraulic Analysis

INTRODUCTION

The thermal hydraulic studies for the LEU core have been a joint effort by the Rhode Island Nuclear Science Center (RINSC) and Argonne National Laboratory. The proposed new fuel elements have been described in the main introduction of the Safety Analysis Report. Pertinent documents reviewed by the RINSC for LEU fuel use are referenced in Appendix A. Fuel plate, channel dimensions and other parameters used in the thermal hydraulic studies are hereby referenced in the Appendix A documents.

DESCRIPTION OF COMPUTER PROGRAMS USED IN THE THERMAL HYDRAULIC ANALYSIS

The computer programs used by the Rhode Island Nuclear Science Center Steady-State Analysis, Hot Channel Analysis and Natural Convection Analysis were obtained from Argonne National Laboratory. The programs were supplied as a VAX/Fortran Version and were subsequently converted for use on an Apple (Macintosh II) computer using the "Absoft Compiler". This was performed so that the staff could utilize in-house computers.

The program entitled "PLTEMP" was used to perform the "Steady-State" and single Hot Channel Analyses.

The program entitled "NATCON" was use to perform the Natural Convection Calculations.

LEU PARAMETERS

The parameters used in the "PLTEMP" and "NATCON" programs were calculated using the "LEU Fuel Element" and the proposed core configurations. Previous sections address nuclear parameters. 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. This information is as shown in Appendix A.

Another parameter studied is the "Critical Velocity for Fuel Plate Deformation". This analysis is shown in "Appendix B". Below is a list of core components and their respective drawing numbers used as reference data.

CORE COMPONENT	DRAWING NUMBER
LEU Fuel Plate	EG&G #411650
Radiation Basket (w/orifice plate)	GE #798D413
Control Blade	GE #197E647
Servo Control Element (Regulating Blade)	GE #612D964, 762D407
Graphite Reflector (also Be reflector)	GE #985C248
Antimony Beryllium Source	GE #655C430
Radiation Basket (center hole type)	GE #798D413

HOT SPOT FACTORS

The use of the LEU fuel element necessitated an evaluation of the engineering hot spot factors to be used in the single hot channel analysis. The Rhode Island Nuclear Science Center has prepared a report entitled "Report on the Determination of Hot Spot Factors for the Rhode Island Nuclear Science Research Using LEU Fuel." The report was performed in August 1989. The results are shown below:

Fb (Bulk Water Temperature Rise)	= 1.62
Fq (Heat Flux)	= 1.46
Fh (Heat Transfer)	= 1.41

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

STEADY STATE FULL CORE ANALYSIS

The PLTEMP Program was used to analyze the full core for the axial peaking factors of 1.32 (blades out) and 1.536 (blades 50% in) core conditions. The full core analysis included the various components (fuel elements, reflectors, baskets etc.) as shown in Figure 4. 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 A. The fuel plate surface temperature vs flow rate is shown in Table B. It is important to note the 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 tabulated data and our pump flow of about 1730 GPM, the core flow (1100 GPM±) and a by-pass flow (625 GPM±) is determined. The corresponding ΔP =.0055 MPA. The results are graphically depicted as LEU core "Flow vs. DP".

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 C. The axial peaking factor vs. relative blade position for the core is tabulated for both the blades out condition (F axial = 1.32) and the blades 50% in (F axial = 1.536). This data was obtained from the nucleonic studies of the core[1] (see Table D). This data was input to the PLTEMP Program to calculate the various parameters at a point by point basis along the axial plate length. The maximum plate surface temperatures shown in Table B reflects these values.

It should be noted that the highest power peaking factor occurs in core position D-6[2], for both the blades out and blades 50% in situations.

REFERENCES

(1) Memo from Bill Woodruff, Argonne National Laboratory to Eugene Spring, Rhode Island Nuclear Science Center, 1/30/91

(2) Memo from James Matos, Argonne National Laboratory to A. F. DiMeglio, Rhode Island Nuclear Science Center, 1/22/91

LEU FULL CORE ANALYSIS - 14 ELEMENT CORE

2 MW

TABLE A

DP (MPA)	Core Flow (kg/s)	By Pass Flow (kg/s)	Total Flow (kg/s)	Core Flow (GPM)	By-Pass (GPM)	Total Flow (GPM)
.0025	43.81	25.49	69.30	698.00	407.00	1105
.0030	48.56	28.15	76.71	774.00	449.00	1223
.0035	52.98	30.63	83.61	844.5	488.50	1333
.0040	57.14	33.66	90.80	910.8	536.20	1447
.0045	61.07	35.13	96.20	973.5	559.50	1533
.0050	64.83	37.21	102.04	1033.4	593.60	1627
.0055	68.43	39.19	107.62	1090.4	624.20	1715
.0060	71.89	41.09	112.98	1146.00	654.00	1800
.0065	75.23	24.77	118.16	1199.00	684.00	1883

TABLE B

ΔP (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
.0025	407.0	698.0	1105	54.70	74.06	54.70	77.52
.0030	449.0	774.0	1223	53.49	71.49	53.44	74.71
.0035	488.5	844.5	1333	52.56	69.69	52.56	72.49
.0040	536.2	910.8	1447	51.81	68.03	51.81	70.69
.0045	559.5	973.5	1533	51.20	66.65	51.20	69.18
.0050	593.6	1033.4	1627	50.69	65.29	50.69	67.89
.0055	624.2	1090.8	1715	50.25	64.28	50.25	66.78
.0060	654.0	1146.0	1800	49.86	63.56	49.86	65.80
.0065	684.0	1199.0	1883	49.53	62.77	49.53	64.94
.0070	719.0	1250.0	1969	49.23	62.07	49.23	64.16

NOTES: (1) Normal Primary Pump Operation 1730 GPM

(2) Calculations Based on Inlet Temp. to Core of 42.3°C

TABLE C

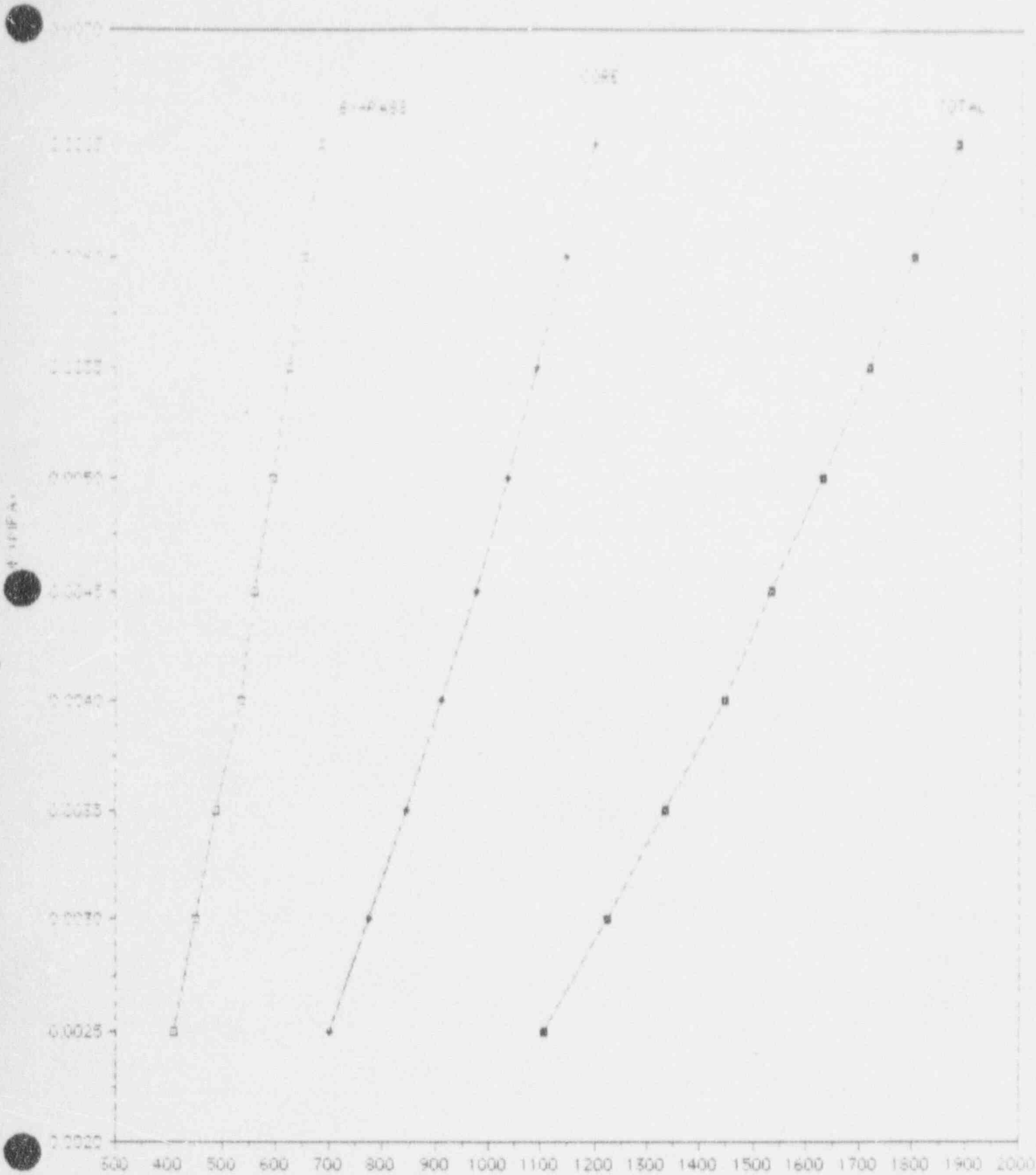
[illegible] PLTEMP full core analysis for each fuel element [illegible]es additional parameters. A <u>typical</u> output value is shown for these parameters. (Element #8 data)

PARAMETER		VALUE
Maximum Surface Temp. °C		64.45
Clad Temp °C		64.78
Peak Axial Heat Flux	(MW/M^2)	.156
Channel Flow Rate	(kg/s)	.2298
Velocity	(M/S)	1.605
Outlet Pressure	(MPA)	.1727
CHF (Critical Heat Flux)	(MW/M^2)	2.39
Flow Instability	(MW/M^2)	.889
Exit Saturation Heat Flux	(MW/M^2)	1.087
Minimum DNB Ratio		8.461
F axial		1.32

TABLE D

INTERFACE	RELATIVE DISTANCE	AXIAL DISTRIBUTION BLADE-OUT	BLADE-IN 50%
1	0.00	.4105	.0553
2	0.05	.5506	.2682
3	0.10	.6836	.4759
4	0.15	.8076	.6744
5	0.20	.9219	.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	.9219	.8597
18	0.85	.8076	.6744
19	0.90	.6836	.4759
20	0.05	.5506	.2682
21	1.00	.4105	.0553

Data from "LEU CORE-FLOW VS DP"



SINGLE CHANNEL (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 terminations.

The tabulated results for axial factors of 1.32 (blades in position) and for axial factors of 1.536 (blades 50% out) are shown in Table E. The results are also presented as a "Hot Channel Fuel Surface Graph" depicting "Fuel Temperature" vs. Total Core Flow.

The normal primary flow rate for the Rhode Island Nuclear Science Center reactor is about 1730 GPM. From the data it can be seen that incipient boiling occurs at about 2.6 MW or 130% of the normal 2 MW power level. At a reduced flow of abut 1580 GPM incipient boiling is reached at about 2.4 MW or 120% of normal power. The proposed limiting safety settings are then chosen as shown below:

Normal Power Level 2MW	Over Power Trip (scram) 120% (2.4 MW)
Normal Flow 1730 GPM	Reduced Flow Trip (scram) 1580 GPM

These values are more restrictive than the present trip levels of 130% for overpower trip (2.6 MW) and 1260 GPM flow. This is due to the fact that the compact core of 14 elements and higher fuel density have more effect than the increase in number of fuel plates from 18 to 22.

The maximum surface temperature of the fuel resulting at the 1580 GPM pump flow is from Tabel E about 110°C.

The corresponding coolant velocity from the PLTEMP output for 1533 GPM (ΔP=.0045) = 1.44 M/S and for 1627 GPM (ΔP=.0050) = 1.53 M/S. An extrapolated value for the 1580 GPM condition is about 1.48 M/S.

These safety settings will require a technical specification change upon NRC approval.

TABLE E

HOT CHANNEL ANALYSIS

F AXIAL = 1.536

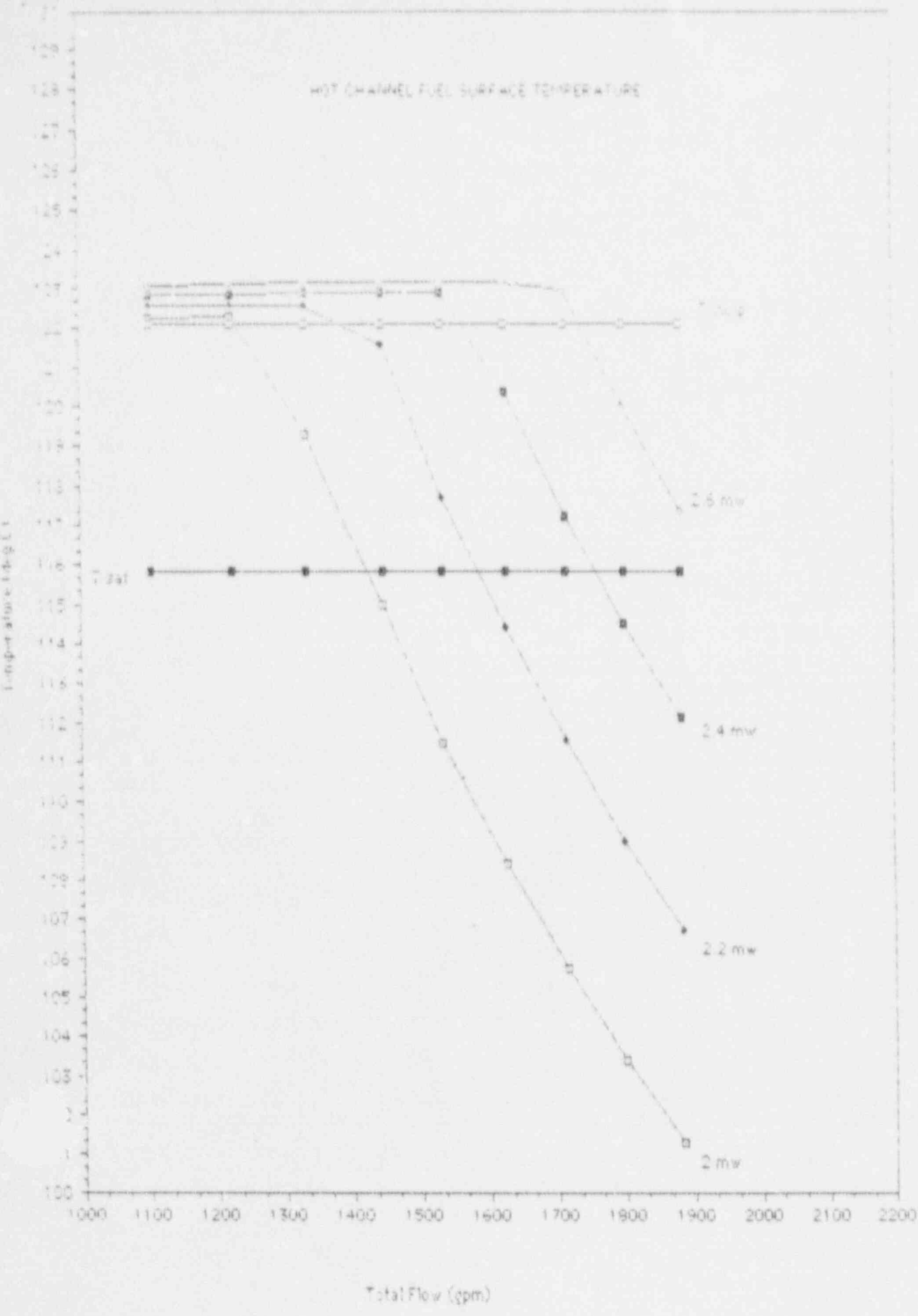
ΔP (MPA)	Total Flow (GPM)	T Surface 2 MW	T Surface 2.2 MW	T Surface 2.4 MW	T Surface 2.6 MW	T SAT.	T onb
.0025	1105	122.26	122.57	122.86	123.10	115.82	122.1
.0030	1223	122.27	122.57	122.87	123.15	115.82	122.1
.0035	1333	119.27	122.58	122.88	123.16	115.82	122.1
.0040	1447	114.99	121.60	122.89	123.17	115.82	122.1
.0045	1533	111.44	117.71	122.90	123.19	115.82	122.1
.0050	1627	108.40	114.41	120.36	123.20	115.82	122.1
.0055	1715	105.14	111.54	117.24	122.93	115.82	122.1
.0060	1800	103.39	109.00	114.53	120.00	115.82	122.1
.0065	1883	101.30	106.75	112.10	117.39	115.82	122.1
.0070	1963	99.42	104.71	109.93	115.06	115.82	122.1

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. °C	T onb °C
.0025	1105	120.80	122.09	122.36	122.62	115.82	122.1
.0030	1223	115.01	121.52	122.37	122.63	115.82	122.1
.0035	1333	110.36	116.52	122.38	122.64	115.82	122.1
.0040	1447	106.55	112.40	118.16	122.65	115.82	122.1
.0045	1533	103.38	108.94	114.44	119.86	115.82	122.1
.0050	1627	100.67	105.99	111.26	116.48	115.82	122.1
.0055	1715	98.30	103.43	108.49	113.51	115.82	122.1
.0060	1800	96.21	101.18	106.07	110.90	115.82	122.1
.0065	1883	94.36	99.17	103.91	108.58	115.82	122.1
.0070	1963	92.69	97.36	101.97	106.51	115.82	122.1

Data from 'HOT CHANNEL ANALYSIS'



NATURAL CONVECTION

The present HEU core has a licensed limit of 100 kw operation for the reactor in the natural convection mode (No Primary Pump Operation).

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 blade out situation (F axial = 1.32) and the blades 50% in (F axial = 1.536).

The results are shown in Table F. Note that for the most conservative case, (hot channel) the power level is 217.3 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 from the data. The program run terminates when the fuel surface temperature reaches incipient boiling.

Since the 217.3 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.

TABLE F

NATURAL CONVECTION

REGULAR CHANNEL FAXIAL = 1.536

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	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	-.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

NATURAL CONVECTION

REGULAR CHANNEL FAXIAL = 1.32

Power Level (kw)	Exit Temp. °C	Maximum Wall Temp. °C	Incipient Boiling °C	T sat-T wall °C	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

RHODE ISLAND NUCLEAR SCIENCE CENTER WATER SUPPLY

The Wakefield Water Company supplies water to the University of Rhode Island (URI) Narragansett Bay Campus. The water is pumped to the 300,000 gallon water storage via a 8" feedline. Water is then distributed to the URI Bay Campus (including the Rhode Island Nuclear Science Center (RINSC)) through a 12" main. Water is supplied to the RINSC through an 8" line which feeds fire protection (6" line) and potable supply (2" line) and a reactor building fire hose (4" line). The entire pumping system has backup generators for total supply reliability.

The enclosed letter from the URI Graduate School of Oceanography which oversees the Bay Campus water supply, can meet a minimum demand of 5 GPM or greater for 24 hours, even in the event of a power failure.

The Bay Campus can provide uninterrupted water supply in the event of a line rupture or planned shutdown by utilizing various cross connections and hydrant connections located in the system network.

The University of Rhode Island Graduate School of Oceanography
Narragansett Bay Campus, Narragansett, RI 02882-1197

July 12, 1991

Mr. Eugene F. Spring, Sr. Reactor Facility Engineer
Nuc ear Science Center
South Ferry Road
Narragansett, RI 02882-1197

Subject: Emergency Water Supply

Dear Eugene:

The chart below indicates that our water system, which includes a stand-by generator, will fullfill your cooling water requirements under various conditions.

Condition	Pressure (PSI) w/o Fire Pump	Volume (Gal) Available	Duration
Normal Town supply→Reservoir→Booster Pumps	70	*	*
Reservoir With Booster Pumps	70	300,000	24 Hrs.
Town Supply With Booster Pumps	70	*	*
Town Supply Only	30	*	*

* = Unlimited within present demand

Campus (Max)	200 GPM
Reactor (Min)	5 GPM
Total	205 GPM

Kenneth W. Morrill
Asst. Dir. Physical Plant

LOSS OF COOLANT ANALYSIS

Following a postulated loss of coolant accident, the pool drain time is calculated by using the falling head calculational method.[1] This is considered the maximum credible accident (refer to SAR Part A-Section XI). It is assumed that water drains from the beam tube (8") from the pool surface (el.139.417) to the top of the core box (115.916). Water then drains from the 1/2" diameter hole in the core box. Water cannot drain below the bottom (invert) of the 8" beam tube and therefore about .7' of water remains in the core box above the active fuel plate edge.

Appendix C shows the schematic and the calculations determining the pool drain time and the flow rate to keep the core box full. This is the minimum drain time, conservatively assuming that the beam port shutter is up, no plugs are in the tube, and no flange cover bolted over the outlet flange.

The original RINSC license amendment for 2 MW operation calculated the decay generation and heat removal following a postulated LOCA for the HEU core. The proposed HEU core has a higher heat density per plate.

Appendix D shows the same simplistic calculation. The results show that the heat removal is not sufficient to remove the decay heat after a LOCA. The conclusion is that an emergency core cooling system is necessary.

The design and operation of such a system is discussed in Section X.

(1) Handbook of Hydraulics, Ernest F. Brater, 6th Edition, McGraw-Hill Book Company, 1976

EMERGENCY CORE COOLING SYSTEM OPERATION

Under normal operating conditions, make-up water is supplied to the reactor pool via the automatic make-up system. In an emergency, for a pool level drop of 2", the automatic value (NC) opens to supply water at the pool level at a design rate of 20 GPM. A manual by-pass can be opened to supply additional flow. Present operating procedures describe the procedure for piping and valve alignment and procedures for normal and emergency filling of the pool. An emergency core cooling line will be installed directly to the reactor core grid box to provide a water supply directly to the fuel elements in the core.

The Emergency Plan (4.1.5 Utilities Failure) directs specific actions to be taken following a drop in water pressure (Implementing Procedure 3.3.1). At present, the detection of a loss of/or drop in city water pressure alarms at the secretary's "desk alarm" box which notifies the operator at the reactor console with other alarms lumped together as a "vital access alarm". The operator must check with the "desk alarm" in order to take appropriate action. It is proposed, as part of the ECCS, that a low city water pressure signal be directly connected to the reactor scram circuitry.

Modifications to the current Emergency Plan and Implementing Procedures would need to be performed.

To insure that such a proposed ECCS be adequate, the RINSC has conducted a water supply analysis to calculate the expected flow. The water flow to the pool has been observed in the make-up system to be about 25 GPM. (Tests will be conducted to verify the actual flow rates and pressures). The design and installation of the ECCS would be performed in accordance with the RINSC QC/QA program.

WATER SUPPLY ANALYSIS

The analysis of the facility water supply system was performed using a computer program called "Service Sizer".[1] It calculates pipe size anc demand. The program has built in piping tables, valve and fitting tables and fixture unit tables. Standard computation techniques are used to determine losses. The program allows any fixture to be specified in either a public or private use situation. Input to the pipe size calculation includes demand flow, demand pressure elevation difference, supply pressure, pipe length, other equivalent pipe length losses, numbers of valve and fittings and also a permitted velocity. The program calculates pipe size, actual velocity, head loss and demand pressure.

The demand calculation includes options for flushometer units; public use or private use. Input for the calculation includes numbers and types of fixtures, a continuous demand flow and additional fixture option. The program calculates total fixture units, continuous and fixture demand and total demand.

The enclosed report shows both the demand and calculated supply size for a proposed water line extension to insure an adequate supply to the reactor core in case of a LOCA.

The report shows that a 2" line will produce 42 GPM. This line size is adequate for normal reactor pool supply and certainly for a 5 GPM supply in a LOCA situation.

(1) Parkcon, Inc., 250 N. Center Street, P. O. Box 5980, Woodland Park, Colorado 8086-5980

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--SIZING CALCULATION-------------------------------------Printed On:  7/18/1991

Supply Location:

                    60.0 psi, supply pressure available during demand

Demand Location:

                      42.0 gpm demand flowing at  40.0 psi pressure

--Head Loss Data----------------------------------------------------------------

Elevation Difference:   30.0 ft   (minus if demand location lower than supply)

Pipe Length:  142.0 ft         Other Loss In Equivalent Pipe Length:         ft

Number of Valves & Fittings:

   :Corp Stop      :Curb Stop     3:Gate Valve     :Globe Valv     :Angle Valv
   :Bfly Valve     :Swing Chk      :Side Tee       :Straight T   13:Std Elbow
   :Long Elbow    3:45 Elbow       :               :               :

Backflow Prev:  1.0 psi  Water Meter:      psi  PRV:      psi  Other:      psi

--Design Calculation------------------------------------------------------------

Permitted Velocity:       fps   Pipe Type: CUM  Calculated Pipe Size:      2 in

   Actual Velocity:  4.2 fps   Head Loss:  17.1 psi  Pres at Demand:  42.9 psi



--DEMAND CALCULATION------------------------------------------------------------

Predominantly Flushometers: N                                     Public Use: N

--Number of Fixtures------------------------------------------------------------

   :Bathtub           :Bar Sink           :Bidet               :Clothes Washr
   :Cuspidor          :Dishwasher        1:Drinking Ftn        :Hose Bib
  1:Kitchen Sink      :Lavatory           :Laundry Tub        1:Shower Head
  1:Service Sink      :Urinal Pedest     2:Urinal Wall         :Urinal Tank
   :Wash Sink         :WC Flushometr     2:WC Tank             :
   :                  :                   :                    :

Additional:     fixture units                     Total:    23.0 fixture units

--------------------------------------------------------------------------------

Continuous Demand:   25.0 gpm                        Fixture Demand:   17.0 gpm

                                                       Total Demand:   42.0 gpm
```

APPENDIX A

LEU THERMAL CONDUCTIVITY CALCULATION

<u>Density of U_3Si_2</u>

The densities of the dispersants are taken from reference (1) with the volume fraction related to the uranium density, Pu, in the fuel by:

$$Pu = 11.28Vf$$

where Vf is the volume fraction of the dispersant

for the purposes fuel loading of 12.5 $\frac{g/cc}{plate}$ (22 plate element, 275 g U-235)

the U density is 3.4682 per reference (2)

the volume fraction of U_3Si_2 in fuel meat is

$$Vf^{U_3Si_2} = \frac{3.4682}{11.28} = .3068 \text{ or } 30.68\%$$

From reference (3), page 11

$$Vp = .072\ Vf - .275\ Vf^2 + 1.32\ Vf^3$$

therefore $Vp = .072(.3068) - .275(.3068)^2 + 1.32(.3068)^3 = .0343$

where Vp and Vf are volume fractions or porosity and fuel in the meat, respectively.

<u>Thermal Conductivity of U_3Si_2</u>

Volume fraction of fuel plus voids = .3068 + .0343 = .3411 the thermal conductivity is obtained from Figure 6, page 16 of reference (3)

$$K = 88\ W/m.k$$

REFERENCES

(1) R.F. Domagala, T.D. Wiencek, and H.R. Tresh, "Some Properties of U-Si Alloys in the Composition Range U_3Si to U_3Si_2," CONF-8410173, ANL, RERTR/TM-6, 47, July 1985.

(2) Memo from W. Woodruff (B17681 at ANLOS) to Eugene Spring (RMA101 at URI MUS), Sept 5, 1989.

(3) J.L. Snelgrove, R.F. Domagala, G.L. Hofman, T.C. Wiencek, G.L. Copeland, R.W. Hobbs, and R.L. Senn, "The Use of U_3Si_2 Dispersed in Aluminum in Plate-type Fuel Elements for Research and Test Reactors," Argonne National Laboratory (ANL/RERTR/TM-11), October 1987.

APPENDIX B

CRITICAL VELOCITY FOR FUEL PLATE DEFORMATION

It has been shown that a critical flow velocity exists for a given plate assembly.[1] At this critical velocity, the plate becomes unstable and large deflections of the plate can occur. These plate deflections can cause local overheating of the fuel plates and possibly a complete blockage of the coolant flow.

Miller[2] derived a formula for the critical velocity based on the interaction between the changes in channel cross-sectional areas, coolant velocities, and pressures in two adjacent channels. For design purposes, reference (3) and (4) recommends that the coolant velocity be limited to 2/3 of the critical velocity given by Miller,[2] therefore for a flat plate;

$$V\ \text{Critical} = 2/3 \left[\frac{15 \times 10^{5} \cdot E \cdot (t_p^3 - t_m^3) \cdot tw}{\rho \cdot W^4 \, (1 - \gamma^2)} \right]^{1/2}$$

where

E	= Young's Modulus of elasticity, bar	: 10.4×10^6 psi/14.5 (Al)
tp	= Fuel plate thickness, cm	: .12[illegible]
tm	= Fuel meat thickness, cm	: .0[illegible]
ρ	= Density of water, kg/m^3	: 1000.0
tw	= Water channel thickness, cm	: .381
w	= Fuel plate width, cm	: 6.096
γ	= Poisson's ratio, dimensionless[5]	: .3 (Al)

V Critical = 16.6 m/sec

The average core velocity of the 14 element LEU core calculated for the normal primary pump flow of 1730 gal/min (.109 m^3/sec) is about 1.6 m/sec. For a projected 5 MW core, the velocity would be increased to about 4 m/sec. This is well below the limiting value of 16.6 and therefore is not a problem in the proposed RINSC core.

REFERENCES

(1) International Atomic Energy Agency, Research Reactor Core Conversion from the Use of Highly Enriched Uranium to the Use of Low Enriched Uranium Fuels, Guidebook IAEA-TECDOC-233 (1980)

(2) Miller, D.R., "Critical Velocities for Collapse of Reactor Parallel Plate Fuel Assemblies", Trans ASME, J. Eng.for Power, 82,83 (1960)

(3) Mishima, K. and Shibata, T. "Thermall-Hydraulic Calculations for KUHFR with Reduced Enrichment Uranium Fuel, "KURRI-TR-223 (1982)

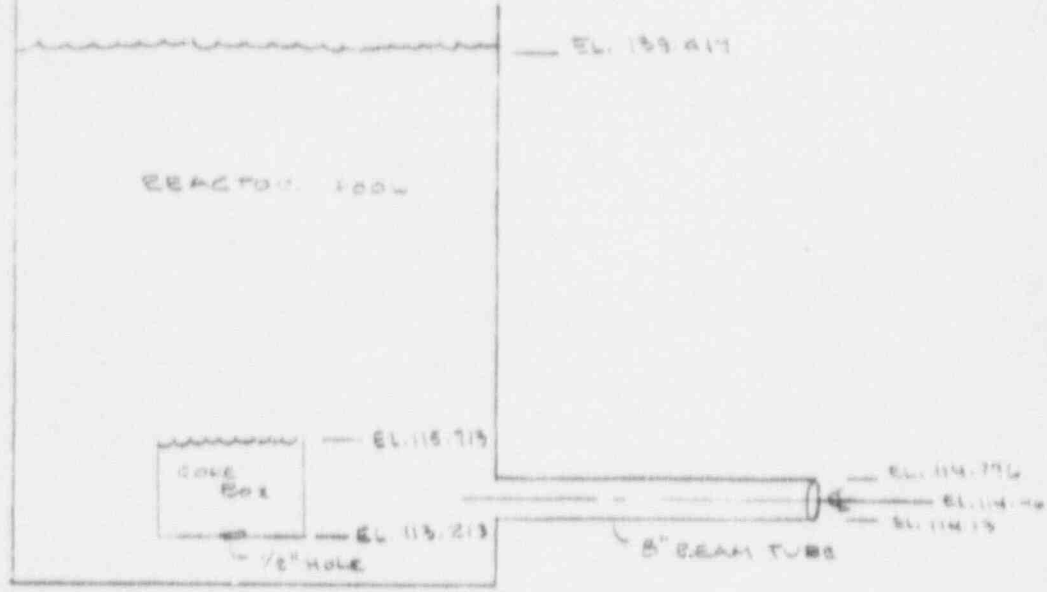
(4) S. McLain and J. H. Martens, Reactor Handbook, Vol. IV, Interscience Publishers (1964)

(5) Standard Handbook for Mechanical Engineers, Baumeister and Marks, 7th Edition, 1967 pg. 5-6

APPENDIX C

LOSS OF COOLANT

Schematic



<u>SURFACE AREAS</u> (FREE FLOW AREA)

Area of entire pool surface	:	150 ft^2
Area of core box	:	5.06 ft^2
Area of core (loaded)	:	.917 ft^2
Area of 1/2 diameter hole in core box	:	.00136 ft^2
Area of 8" pipe	:	.349 ft^2

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" beam tube.

The amount of water remaining in the core box after draining = 114.13 - 113.213 = .917'

<u>ASSUMPTION</u>:

Gravity draining of pool from pool surfaces to the top of the core box out the 8" beam tube which has no plug in place and the shutter in the up position. There is no cover flange.

COMPUTATIONAL METHOD:

Discharge under falling head[1]

$$t = \frac{2A}{Ca(2g)^{1/2}} \left[(H_1)^{1/2} - (H_2)^{1/2} \right]$$

Datum is el.114.13 (invert of bottom of 8" beam tube)

H1 = 139.417 = 25.287

H2 = 115.713 - 114.13 = 1.583

C = .6

A = 150 ft2

a = .349 ft2

$$t = \frac{2x150}{6x.349\ (2x32.2.2)^{1/2}} \left[(25.2871/2-(1.583)1/2 \right] = 673.15 \text{ sec}$$

$$=11.219 \text{ min}$$

Drain Time of Core Box (loaded with components)

t_2 = 2x.917 $\left[(H_1)^{1/2} - (H_2)^{1/2} \right]$

H_1 = 115.713 - 114.130 = 1.583

H_2 = 11′.130 - 114.130 = 0

t_2 = 335.6 sec/60 = 5.59 min

TOTAL TIME = 16.8 min

Minimum required water flow rate to keep the core box full

From (1) a

$F = .61A(2_gH)^{1/2}$ where H = 115.713-114.13 = 1.583

$F = .61 \times .00136\ (64.4 \times 1.583)^{1/2}$

F = .61 x .00136 x 10.097 = .008376 cfs

$F = .008376\ \frac{ft^3}{sec} \times 7.48\ \frac{gal}{ft^3} \times 60\ sec = 3.76\ gpm^{\pm}$

(this assumes core is full)

Discharge under Falling Head. Figure 4-5 shows a vessel filled with water to a depth h_1. The time required to lower the water surface to a depth h_2 is required. a is the area of orifice, and A is the area of water surface for a depth y. C is the coefficient of discharge. The increment of time dt required to lower the water the infinitesimal distance dy is

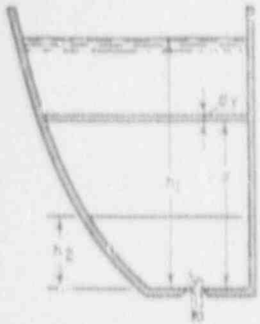


Fig. 4-5. Discharge under falling head.

$$dt = \frac{A\,dy}{Ca\sqrt{2gy}} \qquad (4\text{-}18)$$

From (4-18), if A can be expressed in terms of y, by integrating between limits h_1 and h_2, the time needed to lower the water surface the distance $h_1 - h_2$ can be gotten. Placing $h_2 = 0$ gives the time of emptying the vessel. Equation (4-18) applies to horizontal or inclined orifices provided the water surface does not fall below the top of the orifice. For a cylinder or prism with vertical axis, A is constant, and Eq. (4-18), after integration, becomes

$$t = \frac{2A}{Ca\sqrt{2g}}\left(\sqrt{h_1} - \sqrt{h_2}\right) \qquad (4\text{-}19)$$

Orifice Coefficients. One of the earliest experimenters on sharp-edged orifices was Hamilton Smith, Jr.[1] His values of the coefficient of discharge for round and square orifices are given in Table 4-3.

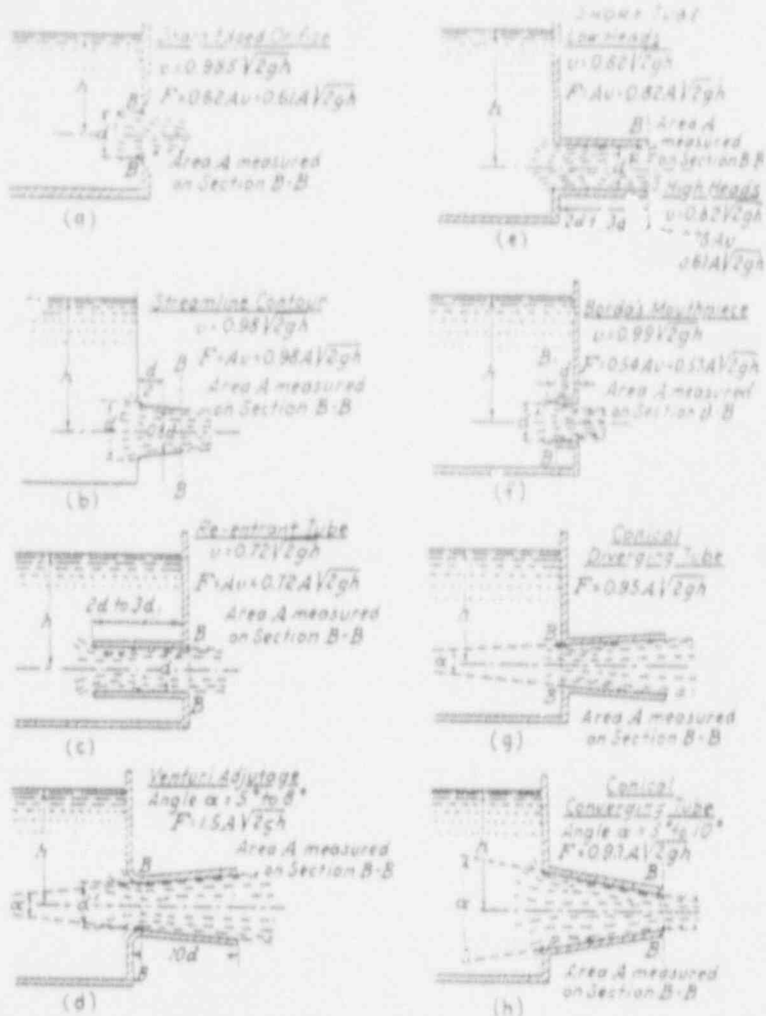


Table 4-3. Smith's Coefficients of Discharge for Circular and Square Orifices with Full Contraction

Diameter of circular orifices, feet							Head, feet	Side of square orifices, feet						
0.02	0.04	0.07	0.1	0.2	0.6	1.0		0.02	0.04	0.07	0.1	0.2	0.6	1.0
.....	0.637	0.624	0.618				0.4	.....	0.643	0.628	0.621			
0.655	0.630	0.618	0.613	0.601	0.593		0.6	0.660	0.636	0.623	0.617	0.605	0.598	
0.648	0.626	0.615	0.610	0.601	0.594	0.590	0.8	0.652	0.631	0.620	0.615	0.605	0.600	0.597
0.644	0.623	0.612	0.608	0.600	0.595	0.591	1	0.648	0.628	0.618	0.613	0.605	0.601	0.599
0.637	0.618	0.608	0.605	0.600	0.596	0.593	1.5	0.641	0.622	0.614	0.610	0.605	0.602	0.601
0.632	0.614	0.606	0.604	0.599	0.597	0.595	2	0.637	0.619	0.612	0.608	0.605	0.604	0.602
0.629	0.612	0.605	[illegible]	0.599	0.598	0.596	2.5	0.634	0.617	0.610	0.607	0.605	0.604	0.602
0.627	0.611	0.604	0.603	0.599	0.598	0.597	3	0.632	0.616	0.609	0.607	0.605	0.604	0.603
0.623	0.609	0.603	0.602	0.599	0.597	0.596	4	0.628	0.614	0.608	0.606	0.605	0.603	0.602
0.618	0.607	0.603	0.600	0.598	0.597	0.596	6	0.623	0.612	0.607	0.605	0.604	0.603	0.602
0.614	0.605	0.601	0.600	0.598	0.596	0.596	8	0.619	0.610	0.606	0.605	0.604	0.603	0.602
0.611	0.603	0.599	0.598	0.597	0.596	0.595	10	0.616	0.608	0.605	0.604	0.603	0.602	0.601
0.601	0.599	0.597	0.596	0.596	0.596	0.594	20	0.606	0.604	0.602	0.602	0.602	0.601	0.600
0.596	0.595	0.594	0.594	0.594	0.594	0.593	50	0.602	0.601	0.601	0.600	0.600	0.599	0.599
0.593	0.592	0.592	0.592	0.592	0.592	0.592	100	0.599	0.598	0.598	0.598	0.598	0.598	0.598

APPENDIX D

DECAY HEAT - CALCULATIONS

(1) ASSUMPTIONS

A. The reactor has been operating for 40 hours continuously.

B. The reactor scrams when the loss of coolant sequence begins at time zero.

C. Reactor pool water level reaches its lowest level as shown in Appendix C.

D. Following the LOCA, the decay heat is conducted away to the remaining water in the core box.

E. The conduction loss will occur with the plate temperature reaching the melting point of aluminum.

(2) DECAY HEAT GENERATION

$$Q\ 2\ MW = \frac{6.86x\ 10^6\ Btu/hr}{14\ el\ x\ 22\ plates/element} \times H12/3600\ sec = 6.87\ Btu/sec$$

From Table 5.1 at Time = 16.8 min (1008.75 sec) ~ 10^3 sec

$$\frac{P}{Po} = .0185$$

$$P = .0185 \times 6.187 = .114\ Btu/sec$$

(3) 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

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

where Qmax = 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:

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

For conduction downwards (-x direction) and using the heat conduction equation from reference (1)

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

kf = fuel plate thermal conductivity

for a sinusoidal heat generation

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

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

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

substituting (6) into (4)

$$d^2t/dx^2 = -Q_{ave}/l.w.t \,.\, \Pi/2 \,.\, 1/kf \,.\, \sin \Pi x/2 \qquad (7)$$

integrating (7)

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

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

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

then

$$dt/dx = Q_{ave}/w.t.kf \,.\, \cos \Pi x/l - Q_{ave}/2\, wt.kf$$

then integrating with the limits

T = 1200°F at top of fuel plate x = 2'
T = 212°F at surface of water in core box x = .7'

then Q_{ave} = .013 Btu/sec

Since this value is less than the decay heat generation of .114 Btu/sec, it is assumed that melting will occur.

(5) DECAY TIME TO HAVE GENERATION EQUAL TO REMOVAL
The length of time that core cooling would be needed to have the decay heat to reduce to .049 Btu/sec can be calculated using Table 5.1.

the Power Ratio = .049/6.187 = .00792

and time = 2×10^4 sec = 5.56 hrs

therefore emergency core cooling is required for at least 5.56 hours. Water supply can be supplied for at least 24 hours.

(6) TOTAL CORE COOLANT LOSS UNDER LOCA CONDITIONS
Total loss would include evaporation of the water from the core box (assumed to be the rate at the maximum value associated with the initial LOC at time equal to the drain time, or 16.8 min).

From Table 5.1 - P_s/P_o = .0185 @ T = 1008 sec

The heat generation then = .0185 x 6.87 Btu/sec

= .1144 Btu/sec

The maximum evaporation rate for water at atmospheric saturation (100°C) = .1144 Btu/sec x 1/970 Btu/lb

Then: .00011794 lb/sec x 60 sec/min x 1 ft^3/59.8 lb/ft^3 x 7.48 gal/ft^3 = .000851277 gal/min (liquid)

This is added to the drainage loss and the total loss is still about 3.76 ± gpm.

(4) 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 = .05" x 2.79" = .1395"^{2}

Max Fuel Cross Section = .02" x 2.47" = .0494"^{2}

Non Fuel Plate Cross Section = .1395 - .0494 = .0901"^{2}

22 plates x .0901 = 1.9822"^{2}

B. Side Plates of the Element

Average width = .187"

22 grooves (.187-.088) x .058"

Cross Section

2 Side Plates x [(.187" x 3.045")-22 x (.87-.088) x .058)]

Area = .8862"^{2}

C. Total Area for the Element

Area = 1.9822 + .8862 = 2.8684

Per Plate Basis

A = 2.8684/22 = .13038"^{2}/144 = .0009054'^{2}

Heat Conducted from the Aluminum to the Water

$Q = k_{ae}\ A\ dt/dx$

$Q = k_{al}A\ (Tmax-Tsat/l)$

$Q = \frac{131 x 0009 x (1200-212)}{l = (2-.7) = 1.3'} = 89.604$ Btu/hr x 1/3600

Q = .02489 Btu/sec

total heat conducted = fuel + aluminum

= .013 + .02489 = .03789 Btu/sec

From the original SAR it was assumed that about 30% of the heat was used in steam formation

therefore .3 x .03789 = .011367 Btu/sec

and the total heat removal = .03789 + .011367

= .049 Btu/sec

Table 5.1

The Ratio, $P(t_s) / P_o$, 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 $P(t_s) / P_o$	Time After Shutdown, t_s (seconds)	Power Ratio $P(t_s) / P_o$
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	0.000625		

APPENDIX E

MAXIMUM HEAT FLUX

The Rhode Island Nuclear Science Center Technical Specifications Section K,3,e,(2) specifies the maximum heat flux. Since it is not specific in regard to how this was originally calculated using an overall hot spot factor of 2.8, the LEU hot channel analysis for 2 MW calculates two conditions resulting in slightly different values.

<u>Case 1</u>

This calculation determines the maximum heat flux of .365 MW/M2 when using an axial peaking factor of 1.32. This is the case when the blades are out of the core. The hot spot factors cited in Section IV are used.

<u>Case 2</u>

This calculation determines the maximum heat flux of .424 MW/M^2 when using an axial peaking factor of 1.536. This is the case when the blades are 50% inserted in the core. Again the standard hot spot factors were included.

APPENDIX F

MAXIMUM CORE SPECIFIC POWER

The Rhode Island Nuclear Science Center Technical Specifications Section K,3,e(2) specifies the maximum specific power. For a 14 element LEU core we are simply calculating maximum core specific power as 2 MW divided by the number of fuel elements having a maximum loading 275 g U^{235}.

Therefore, the specific power is $\frac{2\times10^6 \text{ watts}}{14\times275} = 519.48 \frac{W}{g\ U^{235}}$

Since burnup increases this value, this is the limiting value which cannot be reduced.

SAFETY ANALYSIS REPORT

PART C TECHNICAL SPECIFICATION REVIEW AND MODIFICATION

I Introduction

II **Appendix A**/Rhode Island Nuclear Science Center Reactor Technical Specifications Appendix A to Facility License R-95
Dated July 21, 1964
Revised Through Amendment #16

III **Appendix B**/Proposed Rhode Island Nuclear Science Center Reactor Technical Specifications

INTRODUCTION

There are numerous Technical Specification changes required as a result of the use of the LEU fuel in the Rhode Island Nuclear Science Center reactor.

Parts A and B of the Safety Analysis Report touch on many of them. As a result of the Rhode Island Nuclear Science Center review process, additional changes which reflect current conditions or clarifications of some Technical Specification sections are also included in the final Technical Specification version. Appendix A is a copy of the Rhode Island Nuclear Science Center current Technical Specifications. Appendix B is a copy of the Technical Specifications with the changes included as a result of the SAR and review process. The double vertical lines adjacent to a section designates the section which has the proposed changes.

Implementation of the final approved Safety Analysis Report will be a difficult task for the Rhode Island Nuclear Science Center. Conditions outside the control of the licensee, such as key staff retirements, budget cuts, small operating staff etc., increase the difficulty and will curtail the operation of the facility during the conversion process. The Rhode Island Nuclear Science Center acknowledges the assistance of Argonne National Laboratory in the preparation of the Safety Analysis Report.

APPENDIX A

RHODE ISLAND NUCLEAR SCIENCE CENTER

REACTOR TECHNICAL SPECIFICATIONS

APPENDIX A

TO

FACILITY LICENSE

R-95

DATED JULY 21 1964

REVISED THROUGH AMENDMENT #16

TABLE OF CONTENTS

TABLE OF CONTENTS (CONTINUED)

A. Site

1. Location

The reactor shall be located at the Rhode Island Nuclear Science Center on three 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 University of Rhode Island is a state agency. 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 and 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.

2. Exclusion Area

Figure A.1 is a drawing of the Narragansett Bay Campus showing the three acre Nuclear Science Center site. The boundary of this area shall be posted with conspicuous signs to delineate the area. This three acre area shall be the exclusion area as defined in 10 CFR 100.

3. Restricted Area

Figure A.1 also shows the location of the reactor building on the three acre area. The reactor building and attached office laboratory wing shall be considered the restricted area as defined in 10 CFR 20.

4. Principal Activities

The principal activities carried on within the restricted and exclusion area shall be those associated with operation and utilization of the reactor. It shall be permissible to locate additional Nuclear Science Center or University of Rhode Island buildings within the exclusion area provided that these additional buildings are capable of timely evacuation and do not interfere with the operation of the reactor.

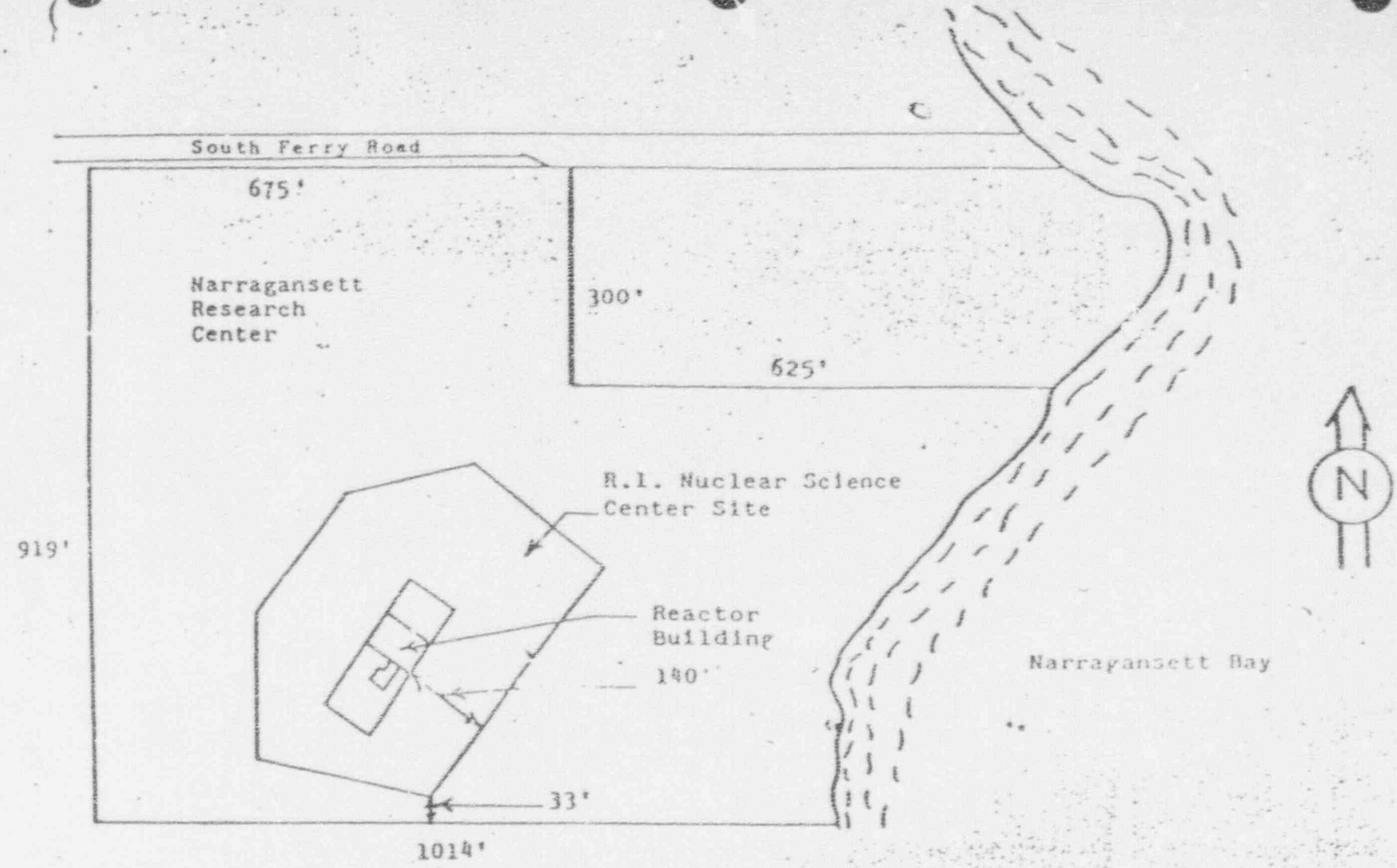


FIGURE A.1 PORTION OF NARRAGANSETT RESEARCH CENTER SHOWING LOCATION OF REACTOR BUILDING

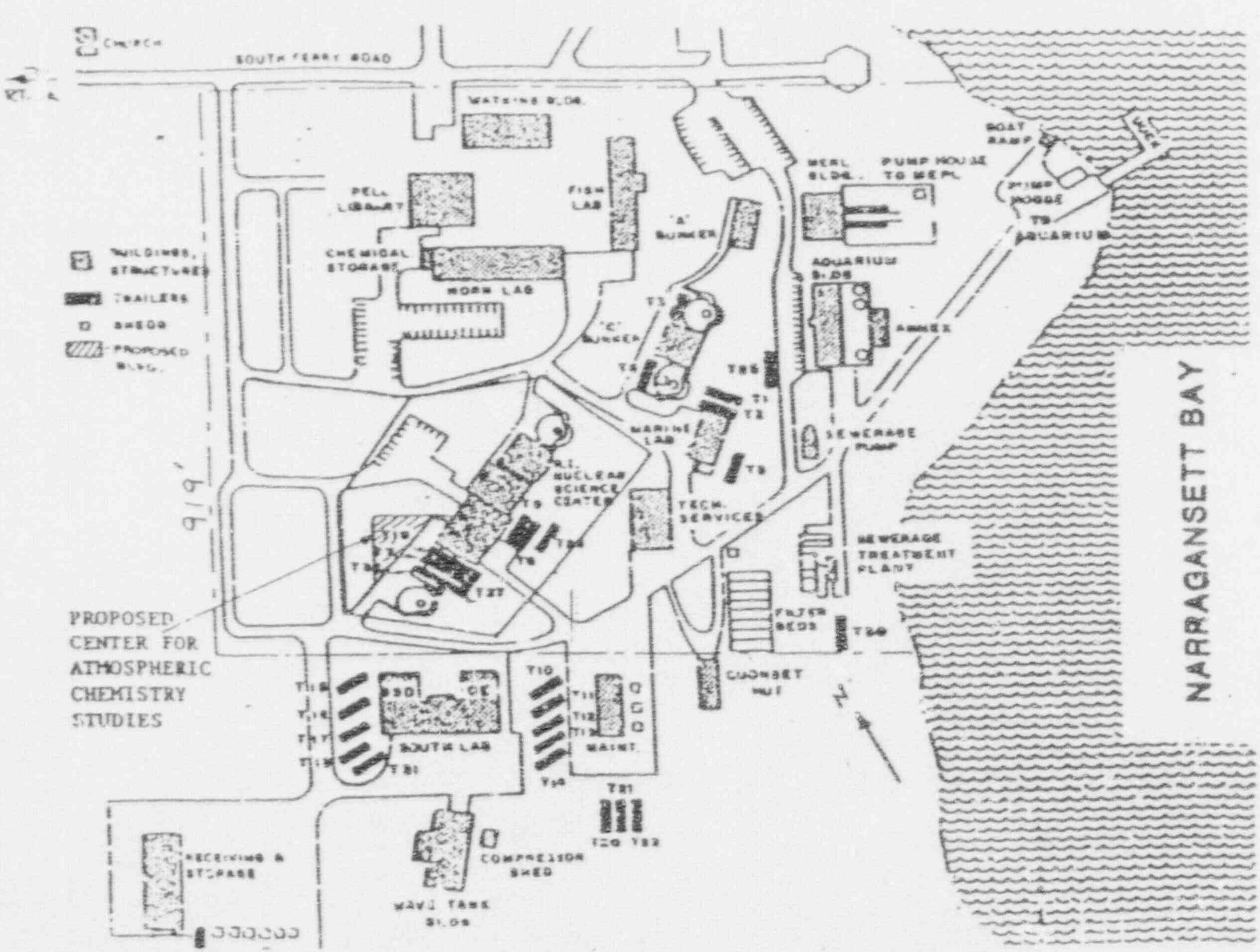


FIGURE A.1 PORTION OF NARRAGANSETT BAY CAMPUS OF URI SHOWING LOCATION OF REACTOR BUILDING

Amend #15 Oct, 1987

B. CONTAINMENT

1. 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 clean-up 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 and air conditioning system intakes 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 electrial 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, which is designed to exhaust at least 4000 cfm, operates in conjunction with additional exhaust blower(s) which provide an additional exhaust of at least 10000 cfm from non-reactor building sources and in conjunction with the air handling unit which takes air into the reactor building at less than 4000 cfm. The total exhaust rate through the stack is at least 14000 cfm. During normal operation, the building is at a pressure somewhat belong atmospheric. The control room air conditioner shall be a self-contained unit, thermostatically controlled, providing constant air temperature for the control room. If it is installed with a penetration through the wall of the reactor building, it shall have a damper at this penetration which closes when an evacuation button is pressed.

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.

C. REACTOR POOL AND PRIMARY COOLANT SYSTEM

1. General

The primary coolant system shall consist of the reactor pool, delay tank, heat exchanger, coolant pump, and the associated valves, piping, flow channels and sensors. During forced convection cooling, coolant water shall be supplied to the core by an aluminum line connected to the inlet flow channel which is on one side of the suspension frame. The coolant water shall flow from the inlet flow channel downward through the core to a plenum below the grid box. The coolant water shall then flow into the outlet flow channel on the opposite side of the suspension frame and then through a discharge line to the delay tank, coolant pump, heat exchanger and then return to the coolant inlet line.

2. Reactor Pool

The reactor pool shall be constructed of ordinary concrete with 1/4" thick 6061-T6 aluminum liner and shall have a volume of approximately 36,300 gal.

3. Shielding

The reactor pool and primary system shielding shall be adequate to meet the applicable personnel radiation protection requirements of 10 CFR 20.

4. Primary Coolant System

The primary coolant system shall conform to the following:

a. Heat Exchanger

The heat exchanger shall be designed to remove heat at the rate generated by the reactor at maximum licensed steady state power from the primary water and shall be designed to perform under the maximum primary system operating temperature and pressure. Replacement heat exchanger shell and tube bundles shall be constructed from stainless steel according to the requirements of Section III, Class C of the ASME Boiler and Pressure Vessel Code.

b. Primary Pump

Number of pumps	1
Type	Horizontal mounted, Centrifugal, Single Suction

Materials of construction	Worthite
Rating	1500 gpm
Head	59 feet
Design Pressure	75 psig minimum
Design Temperature	150°F minimum
Motor Type	Drip proof, induction, 440 v, 3-phase, 60 cycle

c. <u>Delay Tank</u>

Number of tanks	1
Material of construction	Aluminum Association Alloy 5083 and 5086
Material Thickness	
Walls	0.25 inch
Dished Heads	0.375 inch
Capacity	3000 gal., minimum

d. <u>Primary Recirculation Piping</u>

Material and thickness	Sch. 40 Al. type 3003 aluminum
Size	8 and 10 inch
Design temperature	150°F, minimum
Design pressure	100 psig, minimum

e. <u>Make-up System</u>

A check valve shall be installed in the line between the potable water supply and the make-up and cleanup demineralizer to prevent entry of potentially contaminated water into the potable water supply.

Water source	Potable water from city main
Make-up demineralizer type	Mixed-bed single shell, regenerative
Make-up demineralizer capacity	
Normal	25 gpm
Emergency	50 gpm
Water softener capacity	
Normal	50 gpm

f. <u>Cleanup System for Primary Coolant Water</u>

Cleanup pump	
Capacity	40 gpm
Head	100 ft
Cleanup demineralizer	
Type	Mixed-bed, single shell, regenerative
Cleanup demineralizer capacity	
Normal	40 gpm
Emergency	50 gpm

D. SECONDARY COOLANT SYSTEM

The secondary coolant system shall carry the heat rejected from the primary coolant at the heat exchanger to the atmosphere at cooling towers. It shall be composed of the heat exchanger, cooling towers, pumps and associated valves, piping and sensors. In this system, water flows from the heat exchanger through a control valve to the cooling towers. From the cooling tower basins, the water is then pumped back to the heat exchanger.

E. REACTOR CORE AND CONTROL ELEMENTS

The reactor core and control elements shall have the following characteristics and nominal dimensions:

1. Principal Core Materials

Fuel matrix	Alloy, UAl_x, U_3O_8
U-235 enrichment	Approximately 93%
Fuel clad	1100 and/or 6061 aluminum
Fuel element side plates	6061 aluminum
End fittings	356-T6 or 6061 aluminum
Moderator	Water
Reflector	AGOT grade (or equivalent) graphite and/or water
Control elements	Mixture of B_4C and aluminum, clad with aluminum
Servo Element	Mixture of B_4C and aluminum, clad with aluminum.

2. Fuel Elements

Plate width overall	2.8 inches
Active plate width	2.2 inches
Plate length overall	25 inches
Active plate length	24 inches
Plate thickness	0.06 inch
Clad thickness	0.024 inch
Fuel matrix thickness	0.012 inch
Water gap between plates	0.1 inch

Number of plates per fuel element	18
U-235 per fuel element	124 grams, nominal
Overall fuel element dimensions	3 in x 3 in. x 40 in.

3. <u>Reflector Elements</u>

Overall reflector element dimensions, nominal	3 in x 3 in. x 40 in.
Nominal clad thickness	.1 in.
Nominal graphite dimensions	2.8 in. x 2.8 in. x 28.7 in.

4. <u>Control Elements</u>

Width	10.6 in.
Thickness	0.38 in.
Overall length	54.1 in.
Active length	52.1 in.

5. <u>Servo Regulating Element</u>

Shape	Square boral tube
Width	2.1 in.
Overall length	28.8 in.
Active	24.9 in.

6. <u>Control Element Drive</u>

Type	Electromechanical screw
Drive to safety element connection	Electromagnet
Stroke	32 in. maximum

7. Servo Regulating Element Drive

Type	Electromechanical screw
Drive to element connection	Jack screw (no scram)
Stroke	25 in. maximum
Position indication accuracy	± 0.02 in.

8. Neutron Sources

Start-up Source	
Number	2
Type	Plutonium-beryllium
Unit Source Strength	1 x 10^6 neutrons/sec. minimum
Maximum Power Level with Plutonium-beryllium sources installed	10 Kw
Operational Source	
Number	1
Type	Antimony-beryllium
Source Strength	2 x 10^6 neutrons/sec. minimum

F. REACTOR SAFETY SYSTEMS

1. Modes of Power Operation

There shall be two modes of power operation:

a. Power Operation - Natural Circulation (NC)

Power operation - NC shall be any reactor operation performed with the reactor cooling provided by natural circulation. The reactor power shall not exceed 0.1 MW during NC operation.

b. Power Operation - Forced Circulation (FC)

Power operation - FC shall by any reactor operation performed with reactor cooling provided by forced circulation. The reactor power shall not exceed 2 MW during FC operation.

2. Design Features

a. The Reactor Control System

The reactor safety system shall consist of sensing devices and associated circuits which automatically sound an alarm and/or produce a reactor scram. The systems shall be designed on the fail-safe principle (de-energizing shall cause a scram). Table F.1 and F.2 describe the arrangement and requirements of the safety system.

b. Process Instrumentation

Process instrumentation with readout in the control room shall be provided to permit measurement of the flow rate, temperature, and conductivity of the primary coolant and the flow rate of the secondary coolant. In addition, a second primary flow indicating device with readout in the control room shall be located between the reactor outlet plenum and the reactor outlet header.

After normal working hours, an independent protection system, separate from the system described in Section K.3.a, shall be used to monitor certain items in the reactor building and alarm in the event of an abnormal condition. The alarm channels provided are:

(1) A fire in the reactor room,

(2) A fire in a location other than the reactor room,

(3) A decrease of 2 inches in reactor pool water level,

(4) A power failure in the reactor building,

(5) An alarm condition from the radiation monitors reading out in the control room,

(6) An alarm condition from any other selected feature.

c. Master Switch

A key lock master switch shall be provided with three positions; "off", "test", and "on". These positions shall have the following functions:

(1) The "off" position shall de-energize the reactor control circuit.

(2) The "test" position shall energize the reactor control circuit exclusive of the control blade magnets.

(3) The "on" position shall energize the reactor control circuit including the control blade magnets.

d. Power Level Selector Switch

A power level selector switch shall be provided with four positions; "0.1 MW", "1 MW", "2 MW", and "5 MW". These positons shall have the following functions:

(1) The "0.1 MW" position shall activate all safety system sen ors except those indicated in Table F.1.

(2) The "1 MW" and "2 MW positions shall activate all safety system sensors.

(3) The "5 MW" position shall scram the reactor.

e. Control Element Withdrawal Interlocks

Interlocks shall prevent control rod withdrawal unless all of the following conditons exist:

(1) The master switch is in the "on" position,

(2) The safety system has been reset,

(3) The Log N amplifier switch is in the "operate" position,

(4) The startup channel neutron count rate is three counts per second or greater, and

(5) The start-up counter is not being withdrawn.

It shall not be possible to withdraw mo e than one control element at a time.

f. Servo System Control Interlock

Interlocks shall prevent switching to servo control unless the period as indicated by the Log N channel is thirty seconds or greater. The Servo control system shall be designed so that immediately following a scram the Servo control shall automatically return to the manual mode of operation

TABLE E.1

REACTOR SAFETY SYSTEM

Sensor or Trip Device	No. of Switches or Sensors	Trip Set Point	Alarm Set Point
Short Period	1	3 sec. min.	7 sec. min.
High Neutron Flux	2	Max. of 130% of full scale with a 2.6 MW max.	110% max.
High Temperature of Primary Coolant Entering Core During Forced Convection Cooling*			113°F max.
High Temperature of Primary Coolant Leaving Core During Forced Convection Cooling*		125°F max.	123°F max.
Low Flow Rate of Primary Coolant*	1	1200 gpm, min.	1350 gpm, min.
Low Pool Water Level	1	2" max.decrease	2" max.decrease
Seismic Disturbance	1	IV on Modified Mercalli Scale max.	
Bridge Misalignment*	1	X	X
Coolant Gates Open*	1 per gate	X	X
Neutron Detector High Voltage Failure in Linear Level Safety Channels	1 per power supply	Decrease of 50 volts max.	
Manual Scram (Switch at bridge and on console)	2	X	X
High Conductivity of Primary Coolant	1		Equivalent to 2µmho/cm at 25°C, max.
Safety Blade Disengaged	1		X
Log N - Period Amplifier Failure	1	X	X
Regulating Rod at Either Limit of Travel	1		X
Low Flow Rate of Secondary Coolant*	1		800 gpm, min.
Bridge Movement	1	X	X
No Flow Thermal Column*	1	X	X

*These functions are bypassed when the Power Level Selector Switch is in the "0.1 MW" position.

Change 3,4,5

TABLE F.2

REACTOR NUCLEAR INSTRUMENTATION

Channel	Detector	Sensitivity	Range	Information to Operator	Information to Logic Element (Scram)	Information to Servo System	Recorded Information
Start-up	Retractable gas filled B-10 filled proportional	Neutrons- approximately 12 counts/nv	Source level to full power	Neutron Flux	None	None	Relative power level on log scale
Log N	Fixed fission counter	Neutrons- approximately .7 cps/nv	Source level to $3x10^6$ watts	Power level Period	Period scram	None	Power level log scale and period
Linear level safety	Compensated ion chamber	Neutrons- approximately $4x10^{-14}$amp/nv	1 watt to $3x10^6$ watts	Power level	Level scram	Power level	Power level linear scale (either channel)
Linear level safety	Compensated ion chamber	Neutrons- approximately $4x10^{-14}$amp/nv	1 watt to $3x10^6$ watts	Power level	Level scram	None	

G. WASTE DISPOSAL AND FACILITY MONITORING SYSTEMS

1. Waste Disposal Systems Design Features

a. Liquid Radioactive Waste Disposal System

All liquid waste (except sanitary waste) from the reactor building shall flow to retention tanks. These tanks shall be located either underground with a dirt cover or in a locked room(s) in the reactor building.

b. Gaseous Radioactive Waste Disposal System

All gaseous radioactive waste from the beam ports, thermal column, pneumatic irradiaion system and all other radioactive gas exhaust points associated with the reactor itself shall be collected in a manifold and discharged to the reactor stack through an absolute filter, blower and damper.

c. Solid Radioactive Waste Storage

Solid Radioactive wastes shall either be stored in radioactive waste storage containers located within the reactor building or removed from the site by a commercial licensed organization.

2. Area and Exhaust Gas Monitor Design Features

a. Three fixed gamma monitors employing suitable detectors shall be employed in the reactor building. Each of these shall have the following characteristics:

1) A range consistent with the expected radiation levels in the area to be monitored (0.01 to 10 mr/hr, 0.1 to 100 mr/hr, or 1 to 1,000 mr/hr).

2) A radiation dose rate output indicated in the control room.

3) An adjustable high radiation alarm which shall be annunicated in the control room.

4) The three fixed gamma monitors shall be located to detect radiation as follows: At the pool biological shield between a beam port and the thermal column, above the storage container for new fuel elements, and at the reactor bridge.

b. A gamma monitor shall be provided near the primary coolant system, and an additional one shall be provided near the secondary coolant system for use in determining the presence of abnormally high concentrations of radioactivity in these systems. The characteristics of these monitors shall be as stated in a. above.

c. Six additional direct reading area monitors employing Geiger tube detectors shall be provided to monitor the pneumatic system receiver stations, the beam port areas, and other areas as required. Each of these shall have the following characteristics:

1) A range consistent with expected radiation levels in the area being monitored (0 to 10 mr/hr or 0 to 50 mr/hr).

2) A radiation dose rate output at the instrument.

3) An adjustable high radiation alarm to alarm at the instrument and create both an audible and visual signal.

d. A stack exhaust gas monitor system shall be provided which draws a representative sample of air from the exhaust gas. The monitor with indicators and alarms in the control room, shall have the followi.. characterics:

1) A beta particulate monitor with an alarm.

2) A gas monitor incorporating a scintillation detector with high level alarm and a sensitivity for an Argon-41 concentration in air of $10^{-6}\mu c/cc$. The monitor shall have a range of at least four decades.

3. <u>Other Radiation Monitoring Equipment</u>

a. Portable survey instruments for measuring beta-gamma dose rates in the range from .01 mr/hr to 250 r/hr shall be available at the facility. Portable instruments for measuring fast and thermal neutron fluxes in the range from 1 n/cm^2-sec to 25,000 n/cm^2-sec shall also be available to the facility.

b. Reactor excursion monitors shall be placed in the facility for measuring gamma and neutron doses in the event of an accident.

c. A radiation monitor shall be provided to monitor all persons leaving the reactor room for beta-gamma contamination.

4. High Radiation Area

During reactor operation, the dose rate from the delay tank may be in excess of 100 millirem per hour. On three sides, the tank shall be shielded. On the fourth side, the tank is shielded using a "maze" so that access to the tank is posssible through a door equipped with a lock.

H. FUEL STORAGE

1. New Fuel Storage

New fuel shall be stored in a security container in "egg crate" boxes. Sheet cadmium at least 0.020 inches thick shall be fastened around the outside of the boxes in the region which contains the fuel. The number of fuel elements which can be placed in each box shall not exceed three. For all conditions of moderation possible at the site K_{eff} shall be less than 0.8.

2. Irradiated Fuel Storage

Two types of irradiated fuel element storage racks shall be provided. One type of rack shall contain spaces for nine fuel assemblies and shall have approximate over-all dimensions of 35.5 in. wide by 26 in. high by 6.25 in. thick, and shall be fixed to the pool wall. At least two of these racks shall be provided. The second type of rack shall consist of two of the nine fuel assembly racks described above attached together with a minimum space between the centerlines of fuel assemblies in adjacent racks of 12 inches. This 18 fuel assembly rack shall be covered on the two 35.5 x 26 in. outside faces with a neutron absorbing material. At least one 18 fuel assembly rack shall be provided, and the rack may be moved within the pool. The fuel storage racks may also be used to store core components other than fuel assemblies. The irradiated fuel storage racks shall have a maximum K_{eff} of 0.8 for all conditons of moderation possible at the site. Storage spaces shall be provided for at least 36 fuel assemblies.

I. EXPERIMENTAL FACILITIES

The permanent experimental facilities shall consist of the following:

1. Thermal column.
2. Beam ports; two 8 inch dia, and four 6 dia.
3. A six inch diameter through port.
4. Radiation baskets.
5. A two-tube pneumatic tube system.
6. Dry gamma cave.

J. ADMINISTRATIVE AND PROCEDURAL SAFEGUARDS

1. Organization

The Rhode Island Atomic Energy Commission (RIAEC) shall have the responsibility for the safe operation of the reactor. The RIAEC shall appoint a Director of Operations and a Reactor Utilization Committee consisting of a minimum of five members, as follows:

(1) The Director of Operations

(2) The Reactor Facility Health Physicist

(3) A qualified representative from the faculty of Brown University

(4) A qualified representative from the faculty of Providence College

(5) A qualified representative from the faculty of the University of Rhode Island.

A qualified alternate may serve in lieu of one of the above.

The Director and Health Physicist are not eligible for chairmanship of the Committee. The Reactor Utilization Committee shall have the following functions:

a. Review proposals for the use of the reactor considering the suitability of the reactor for the proposed use and the safety factors involved.

b. Approve or disapprove proposed use of the reactor.

c. Review at least annually the operating and emergency procedures and the overall radiation safety aspects of the facility.

The Reactor Utilization Committee shall maintain a written record of its findings regarding the above.

2. Qualifications of Personnel

a. The Director of Operations shall have at least a bachelors degree in one of the physical sciences or engineering, and he shall be trained in reactor technology and be a licensed senior operator.

b. The staff Health Physicist shall be professionally trained and shall have at least a bachelors degree in one of the physical or biological sciences or engineering. He shall have experience such as may have been gained through employment in a responsible technical position in the field of health physics.

c. The reactor operators and senior operators shall be licensed in accordance with the provisions of 10 CFR 55.

d. In the event of temporary vacancy in the position of Director of Operations or the Health Physicist, the functions of that position shall be assumed by qualified alternates appointed by the RIAEC.

3. Responsibilities of Personnel

a. Director

(1) 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.

(2) 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 Reactor Utilization Committee.

b. <u>Senior Reactor Operators</u>

(1) A licensed senior reactor operator 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.

(2) The reactor operations which affect core reactivity shall not be performed without the senior operator on duty or readily available on call. The senior 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 operator as well as the time he assumes the duty shall be entered in the reactor log. When the senior operator is relieved, he 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 operator being relieved of his duty shall insure that all pertinent information is logged. The senior operator assuming duty shall check the log for information or instructions.

c. <u>Reactor Operators</u>

(1) The responsible senior operator shall designate for his shift a licensed operator (hereafter called "operator") who shall have primary responsibility under the senior 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 operator.

(2) 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 assumed duty.

(3) 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 to leave the control room during such an operation, he 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.

(4) The operator, under the senior operator on duty, shall be responsible for the operation of the reactor according to the approved operating schedule.

(5) 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 judgement 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.

B. Health Physicist

The Health Physicist shall be responsible for assuring that adequate radiation montoring and control are in effect to prevent undue exposure of individuals to radiation.

4. Written Instructions and Procedures

Detailed written operating instructions and procedures shall be prepared for all normal operations and maintenance and for emergencies. These procedures shall be reviewed and approved by qualified personnel before use. Each member of the staff shall be familiar with those procedures and instructions for which he has responsibility.

5. Site Emergency Plans

The Rhode Island Nuclear Science Center shall have available the services of other state agencies for dealing with certain types of emergencies. The RIAEC shall enter into an agreement with the Rhode Island Civil Defense Agency whereby the Civil Defense Agency will maintain an emergency monitoring and communications vehicle which they shall make available to the Nuclear Science Center in the event of an emergency involving release of fission products or other radioactive isotopes to the atmosphere. The emergency vehicle shall contain equipment such as portable radiation monitors, respirators, and a particulate air sampler. Communications using the statewide emergency network shall be available.

Personnel of the Civil Defense Agency and of local fire departments shall have received training from the Civil Defense Training Officer in the use of certain radiological instruments. Future training shall be augmented by including orientation on the reactor facility.

K. OPERATING LIMITATIONS

1. General

The foll wing administrative controls shall be employed to assure the safe operation of the facility:

a. The reac.or shall not be operated whenever there are any significan defects in fuel elements, control rods, or control circuitry.

b. The reactor control and safety system must be turned on and functioning properly and an appropriate neutron source must be in the core during any change which can affect core reactivity.

c. During operations which could affect core reactivity, a licensed operator shall be stationed in the control room. Communications between the control room and the senior reactor operator directing the operation shall be maintained.

d. The operator shall not attempt to start up the reactor following an automatic scram or unexplained power decrease until the senior operator has determined the cause of the scram or power decrease and has authorized a start-up.

e. The initial start-up of the reactor shall be performed in conjunction with personnel of the General Electric Company.

f. The reactivity of all core loadings to be utilized in operating the reactor shall be determined using unirradiated fuel elements or elements containing fission products in which the effect of xenon poisioning on total core reactivity has decayed to 0.05% delta k/k or less.

g. Critical experiments shall be performed under the supervision of the Director or other competent supervisory scientist licensed as a senior reactor operator. During the experimont there shall be present, in additon to this licensed supervisor, at least one other technically qualified person who shall act as an independent observer. Each step in the procedure shall be considered in advance by both persons, each calculation shall be checked by both persons, and no step shall be taken without the concurrence of both. A written record shall be made at the time of each fuel element addition or other core change which could significantly affect core reactivity.

h. The basic operating principles for the assembly and reloading of cores whose nuclear properties have been previously determined from critical experiments shall be as follows:

All core loading changes shall be performed under the supervision of a person having a senior operator's license. During the operation there shall be present in addition to the designated senior reactor operator at least one other technically qualified person who shall act as an observer.

The exact procedure to be followed for a particular reloading operation will be determined by the observer and the senior reactor operator in charge of the operation before the operation begins. Each step in the procedures shall be considered by both persons, and no step shall be taken without the concurrence of both.

2. Experiments

a. "Experiments" as used in this section shall be construed as any apparatus or device installed in the core region which is not a component of the core.

b. The Reactor Utilization Committee shall review and approve all experiments before initial performance at the facility. 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 Reactor Utilization Committee. The documentation shall include at least:

   (1) The purpose of the experiment,

   (2) A description of the experiment, and

   (3) An analysis of the possible hazards associated with the performance of the experiment.

c. All use of experimental facilities shall be approved by the Director of Operations.

d. The absolute value of the reactivity worth of any single independent experiment shall not exceed 0.006. If such experiments are connected or otherwise related so that their combined reactivity could be added to the core simultaneously, their combined reactivity shall not exceed 0.006.

e. The calculated reactivity worth of any single independent experiment not rigidly fixed in place shall not exceed 0.0008. If such experiments are connected or otherwise related so that their combined reactivity could be added to the core simultaneously, their combined reactivity worth shall not exceed 0.0008.

f. No experiment shall be installed in the reactor in such a manner that it could shadow the nuclear instrumentation system monitors and thereby give erroneous or unreliable information to the control system safety circuits.

g. No experiment shall be installed in the reactor in such a manner that it could fail so as to interfere with the insertion of a reactor control element.

h. No experiment shall be performed involving materials used in such a way that they might credibly result in an explosion.

i. No experiment shall be performed involving materials which could credibly contaminate the reactor pool causing corrosive action on the reactor components.

j. Experiments shall not be performed involving equipment whose failure could credibly result in fuel element damage.

k. There shall be no more than one vacant fuel element position within the periphery of the active section of the core.

3. Operations

a. Site

Control of access to the reactor facility shall be the responsibility of the Director of Operations.

b. Containment

(1) During any operation in which the control rods are withdrawn from the core containing fuel, the following conditions shall be satisfied:

a. Confinement building penetrations which are not designed and set to close automatically on actuation of the evacuation button shall be sealed, except that doors other than the truck door may be opened during reactor operation. If a door is to remain open, an individual from the reactor operations staff is continuously in attendance at the door.

b. The building clean-up system is operable.

(2) Requirements for Retest of Confinement

(a) Method of Retest

The building cleanup system shall be retested by pressing an evacuation button and observing that the following functions occur automatically:

1. Evacuation horn blows.

2. air conditoning and normal ventilation has turned off.

3. Dampers on all ventilating ducts leading to the outside have closed.

4. Building cleanup system-air scrubber and fresh air blower come on.

5. The negative differential pressure between the inside and outside of the building is at least 0.5 inches of water. This shall be determined by reading the differential manometer located in the control room.

(b) Frequency of Retest

The building cleanup system including the auxiliary electrical power system shall be retested at least weekly.

(3) The exhaust rate through the cleanup system shall not exceed 4500 cfm with not more than 1500 cfm coming from the reactor building and passing through the charcoal scrubber. The remaining air will be provided by a separate blower from an uncontaminated source. This shall create a pressure in the building which is equivalent to at least 0.5 inch of water below atmospheric pressure.

c. Primary Coolant System

(1) The minimum depth of water above the top of the active core shall be 23 feet.

(2) No piping shall be placed in the pool which could cause or fail so as to cause a siphon of the pool water to below the level of the ten inch coolant line penetrations.

(3) Makeup System

The effluent water of the primary coolant water makeup system shall be of a quality to insure compliance with K.3.c.(5) and (6) below.

(4) Cleanup System

The effluent water of the primary coolant water clean up system shall be of a quality to insure compliance with K.3.c.(5) and (6) below.

(5) The primary coolant shall be sampled at a minimum frequency of once per week and the samples analyzed for gross radioactivity, pH, and conductivity in accordance with written procedures. Corrective action shall be taken to avoid exceeding the limits listed below:

pH 5.5 to 7.5

conductivity 2 µmho/cm

(6) The radioactive materials contained in the pool water and in the primary coolant water shall be such that the radiation level one meter above the surface of the pool shall be less than 10 mrem/hr.

(7) During the forced circulation mode of operation, the primary coolant flow rate shall not be less than 1200 gpm. During determinations of reactor power by coolant heat balances, the coolant flow rate may be reduced to 600 gpm providing all other aspects of these Technical Specifications are met.

d. Secondary Cooling System

(1) The secondary coolant shall be sampled at a minimum frequency of once per week and the samples analyzed for pH in accordance with written procedures. Corrective action shall be taken to avoid exceeding the pH limit given below:

pH 5.5 to 9

(2) The concentration of radionuclides in the secondary water shall be determined at least once each day the reactor operates using forced convection cooling. The concentration shall be determined at least once per week when not being operated using forced convection cooling.

(3) If the radioactive materials contained in the secondary coolant exceed a radionuclide concentration in excess of the values in 10 CFR 20, Appendix B, Table I, Column II, above background, the reactor shall be shutdown and the condition corrected before operation using the secondary cooling system resumes.

(4) The secondary coolant system shall be placed in operation as required during power operation utilizing forced convection in order to maintain a primary coolant core outlet temperature of 125°F or below.

e. Reactor Core and Control Elements

(1) The reactor shall not contain in excess of 35 fuel elements. There shall be a minimum of four operable control elements.

(2) The limiting thermal and hydraulic core characteristics based on a 28 element, graphite reflected core are specified below:

(a)	Maximum Heat Flux	47,200 BTU/hr ft^2
(b)	Maximum Core Specific Power	1,120 watt/gm U^{235}
(c)	Maximum Fuel Surface Temperature	197°F
(d)	Coolant Velocity during Forced Convection Cooling	2.65 ft/sec, min.
(e)	Coolant Inlet Temperature	115°F max.
(f)	Average Coolant Temperature Rise	10°F max.
(g)	Primary System Bulk Outlet Coolant Temperature	125°F max.
(h)	Temperature Margin in Primary Coolant ($T_{sat}-T_{surf}$)	43°F
(i)	Number of Coolant Passes Through Core	1

(3) Principal Nuclear Characteristics of the Core

(a) Core and Control System Reactivity Worth

1. The reactor shall be subcritical by at least 1% $\Delta k/k$ from the cold, Xe-free, critical condition with the most reactive control element and the servo regulating element fully withdrawn.

2. The maximum worth of the servo regulating element shall be 0.7% $\Delta k/k$.

(b) Maximum Reactivity Addition Rate - $\Delta k/k/sec$

1.	By servo regulating element	maximum of 0.0002
2.	Manual by control element	maximum of 0.0002

(c) Reactivity Coefficients

1. Temperature coefficient — approximately $-0.5 \times 10^{-4}/^{o}C$ (calculated)

2. Void coefficient (core average) — approximately $-1.8 \times 10^{-3}/\%$ void (calculated)

(4) Principal Core Operating Limitations

(a) Maximum Pool Temperature Limitations

The pool water temperature shall not exceed 130°F.

(b) Reactivity Limitations

1. Excess Reactivity

The cold, clean excess reactivity for any core used in the reactor shall not exceed 0.047.

2. Minimum Shutdown Margin

All reactor cores used shall be such that they would be subcritical if any single control element and the servo regulating element were withdrawn.

(c) Reactivity Coefficient Limitation

The reactor power coefficent (as inferred by the control rod movements required to compensate for changes in power) shall be negative.

(d) Control Element Drive Performance Requirements

All control element drives shall meet the following specifications:

1. The control drive withdrawal rate shall not be more than 3.6 inches per minute.

2. For the electronic scram system, the time from initiation of a scram condition until control element release shall not exceed 100 milliseconds.

3. The time from initiation of a scram condition until the control element is fully inserted shall not exceed 900 milliseconds.

4. It shall be demonstrated at least every 3 months that the above specifications are met.

(e) <u>Servo Regulating Element Drive Performance Requirements</u>
If in use during operation, the servo regulating element drive shall meet the following specifications:

1. The drive withdrawal rate shall not be more than 78 inches per minute.

2. It shall be demonstrated at least once per month that the above specification is met.

(f) <u>Fission Density Limit</u>
The fission density limit for alloy, uranium aluminide, and uranium oxide fuel shall meet the following specifications:

1. The fission density limit shall be 0.5×10^{21} fissions/cc.

2. The fission density of all fuel elements which have burnup shall be calculated at least quarterly.

f. <u>Reactor Safety Systems</u>

(1) The reactor safety system shall be operable during all reactor operation. The safety system shall be checked out before each start-up and functionally tested for calibration at least monthly.

(2) It shall be permissible to continue operations with one or more of the safety system functions that produce only an alarm temporarily disabled providing that additional procedural controls are instituted to replace the lost safety system alarm function(s).

(3) The control element withdrawal interlocks and the servo system control interlocks shall be functionally tested at least once per month.

(4) During reactor startup or during mechanical changes that could affect core reactivity, the startup range neutron monitoring channel shall be operable and shall provide a neutron count rate of at least 3 counts per second with a signal to noise ratio at least 3 to 1.

(5) The linear level safety channels shall not read less than 15% of full scale when the reactor is operating at power levels above 1 watt.

(6) Following a reduction in power level, the operator shall adjust the servo power schedule to the new power level before switching to automatic operation.

(7) An alarm conditon from any one of the items listed in Section F.2.b. after working hours shall transmit coded information to a continuously manned central station in Providence, Rhode Island. The central station shall be provided with written instructions on the steps to be taken following an alarm.

g. Waste Disposal and Reactor Monitoring Systems

(1) The liquid waste retention tank discharge shall flow to a monitor station in the reactor building where the effluent shall be batch sampled and the gross activity per unit volume determined before release. All off-site releases shall be directly into Narragansett Bay.

(2) Gaseous radioactive waste shall be disposed of sing the reactor stack. Disposal limits shall conform to the following table. In this table, the MPC stated is for individual isotopes and mixtures contained in Column 1, Table II, Appendix B of 10 CFR 20.

	1	2
Type of Activity	Maximum Curies per second to k released	Curies per second to be released averaged over one year
Particulate Matter and Halogens with half-lives longer than 8 days	140 X MPC (uc/cc)	14 X MPC (uc/cc)
All other Radioactive Isotopes	10^5 X MPC (uc/cc)	10^4 X MPC (uc/cc)

(3) All radioactive liquid and solid wastes disposed of off-site shall be within the limits established by 10 CFR 20 or shall be removed from the site by a commercial licensed organization.

(4) The exhaust gas monitor shall be calibrated to alarm at an instantaneous release rate which instantaneously exceeds the limits stated in Column 2 for the annual average release rate. If the maximum permissible stack release rate stated in Column 1 is exceeded, the reactor shall immediately be placed in the shutdown mode of operation and the situation investigated.

(5) The area, primary and secondary coolant system and the exhaust gas monitors shall be in operation at all times when control elements or the servo regulating elements are withdrawn; however, indivdual area coolant system monitors may be taken out of service for maintenance and repair if replaced with portable radiation detection equipment. Adequate spare parts shall be on hand to allow necessary repairs to be made during the maintenance or calibration outages of the monitors.

(6) The area and the primary and secondary coolant system monitors shall be adjusted to alarm at a maximum reading of 2 mr/hr or 200% of the normal radiation levels in their area, whichever is larger.

(7) The door which controls entrance to the "maze" leading to the delay tank shall be locked with the key in the possession of the Director or a licensed senior operator. Entrance to the delay tank high radiation area shall require the presence of the Health Physicist or a licensed senior operator and the use of direct reading portable radiation monitoring equipment.

h. Fuel Storage

(1) New fuel shall be stored in egg crate boxes located in a security container. Access to the security container shall be restricted, through use of a lock, to the Director of Operations and the licensed senior reactor operators.

(2) Irradiated fuel, not in use in the reactor core, shall be stored in the criticality safe storage racks described in Section H. Only one fuel assembly may be inserted or moved from a storage rack at a time.

(3) Safety against inadvertent criticality shall be provided by limiting the number of fuel assemblies per rack to nine and then positively securing such racks at least 30 cm. apart, or by limiting the number of fuel assemblies to 18 per rack and then covering the two large faces of each rack with a sheet of aluminum covered cadmium.

4. Maintenance

(a) The electronic control and the process control system shall be checked for proper operation and calibration before each reactor start-up. If maintenance or recalibration is required, it shall be performed before reactor start-up proceeds.

(b) Maintenance shall be performed with the approval of the Director. Equipment and system maintenance records shall be kept to facilitate scheduling and completion of all necessary maintenance.

(c) Routine maintenance on all control and process system components shall be performed in accordance with written schedules and with written procedures.

INTRODUCTION

There are numerous Technical Specification changes required as a result of the use of the LEU fuel in the Rhode Island Nuclear Science Center reactor.

Parts A and B of the Safety Analysis Report touch on many of them. As a result of the Rhode Island Nuclear Science Center review process, additional changes which reflect current conditions or clarifications of some Technical Specification sections are also included in the final Technical Specification version. Appendix A is a copy of the Rhode Island Nuclear Science Center current Technical Specifications. Appendix B is a copy of the Technical Specifications with the changes included as a result of the SAR and review process. The double vertical lines adjacent to a section designates the section which has the proposed changes.

Implementation of the final approved Safety Analysis Report will be a difficult task for the Rhode Island Nuclear Science Center. Conditions outside the control of the licensee, such as key staff retirements, budget cuts, small operating staff etc., increase the difficulty and will curtail the operation of the facility during the conversion process. The Rhode Island Nuclear Science Center acknowledges the assistance of Argonne National Laboratory in the preparation of the Safety Analysis Report.

APPENDIX B

PROPOSED

RHODE ISLAND NUCLEAR SCIENCE CENTER

REACTOR TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

TABLE OF CONTENTS (CONTINUED)

A. Site

1. Location

The reactor shall be located at the Rhode Island Nuclear Science Center on three 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 University of Rhode Island is a state agency. 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 and 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.

2. Exclusion Area

Figure A.1 is a drawing of the Narragansett Bay Campus showing the three acre Nuclear Science Center site. The boundary of this area shall be posted with conspicuous signs to delineate the area. This three acre area shall be the exclusion area as defined in 10 CFR 100.

3. Restricted Area

Figure A.1 also shows the location of the reactor building on the three acre area. The reactor building and attached office laboratory wing shall be considered the restricted area as defined in 10 CFR 20.

4. Principal Activities

The principal activities carried on within the restricted and exclusion area shall be those associated with operation and utilization of the reactor. It shall be permissible to locate additional Nuclear Science Center or University of Rhode Island buildings within the exclusion area provided that these additional buildings are capable of timely evacuation and do not interfere with the operation of the reactor.

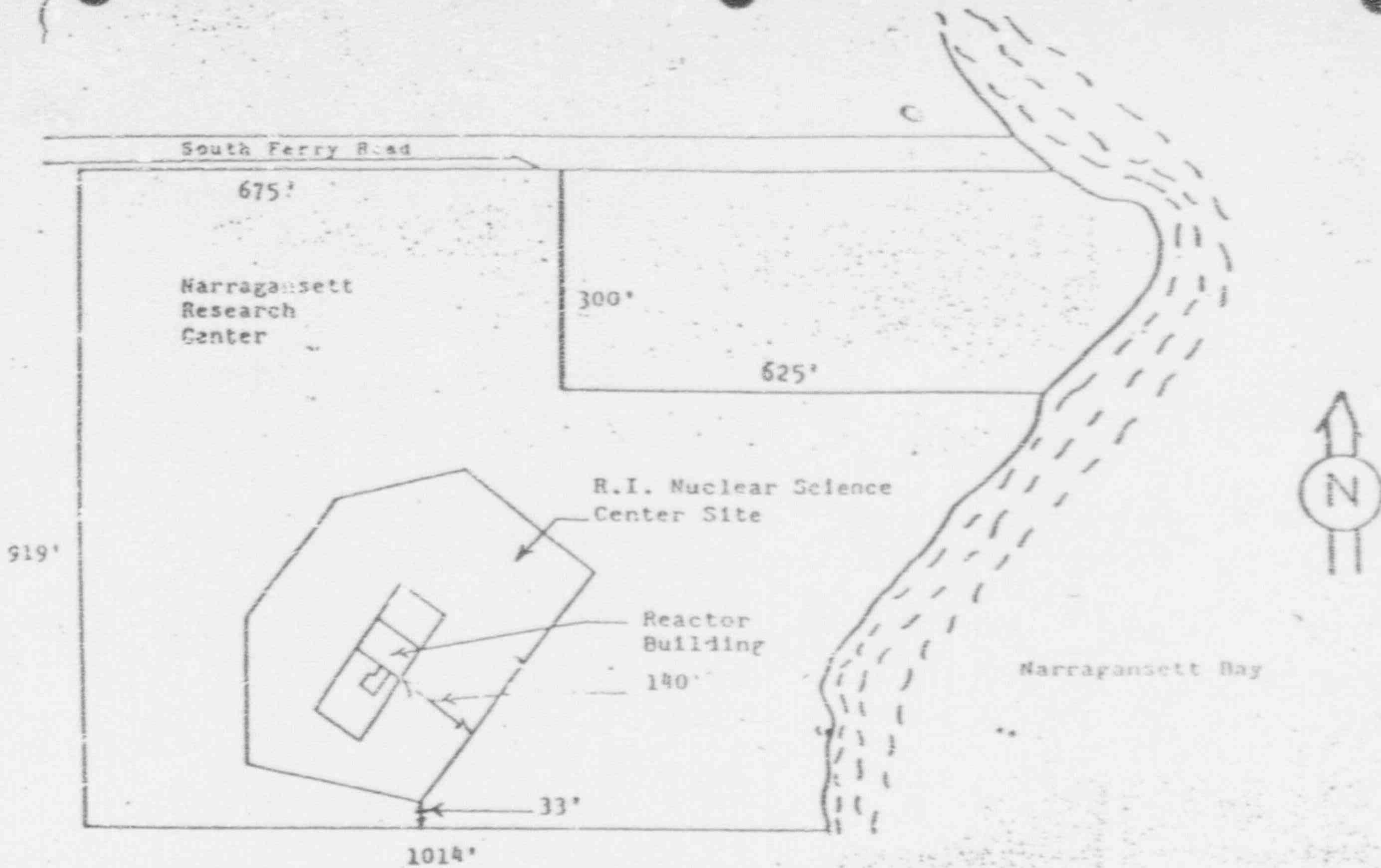


FIGURE A.1 PORTION OF NARRAGANSETT RESEARCH CENTER SHOWING LOCATION OF REACTOR BUILDING

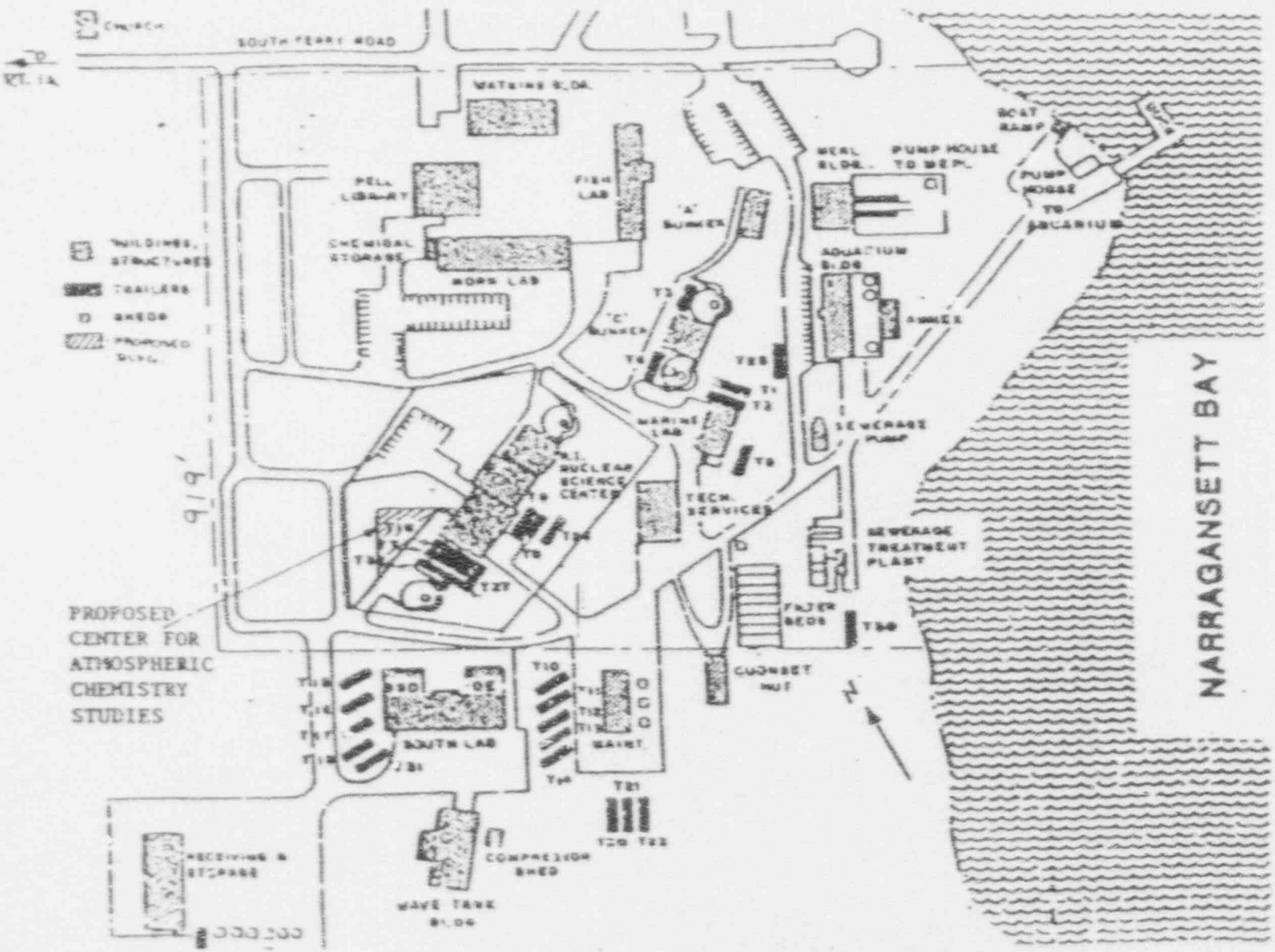


FIGURE A.1 PORTION OF NARRAGANSETT BAY CAMPUS OF URI SHOWING LOCATION OF REACTOR BUILDING

Amend #15 Oct, 1987

B. CONTAINMENT

1. 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 clean-up 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 and air conditioning system intakes 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, which is designed to exhaust at least 4000 cfm, operates in conjunction with additional exhaust blower(s) which provide an additional exhaust of at least 10000 cfm from non-reactor building sources and in conjunction with the air handling unit which takes air into the reactor building at less than 4000 cfm. The total exhaust rate through the stack is at least 14000 cfm. During normal operation, the building is at a pressure somewhat belong atmospheric. The control room air conditioner shall be a self-contained unit, thermostatically controlled, providing constant air temperature for the control room. If it is installed with a penetration through the wall of the reactor building, it shall have a damper at this penetration which closes when an evacuation button is pressed.

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.

C. REACTOR POOL AND PRIMARY COOLANT SYSTEM

1. General

The primary coolant system shall consist of the reactor pool, delay tank, heat exchanger, coolant pump, and the associated valves, piping, flow channels and sensors. During forced convection cooling, coolant water shall be supplied to the core by an aluminum line connected to the inlet flow channel which is on one side of the suspension frame. The coolant water shall flow from the inlet flow channel downward through the core to a plenum below the grid box. The coolant water shall then flow into the outlet flow channel on the opposite side of the suspension frame and then through a discharge line to the delay tank, coolant pump, heat exchanger and then return to the coolant inlet line.

2. Reactor Pool

The reactor pool shall be constructed of ordinary concrete with 1/4" thick 6061-T6 aluminum liner and shall have a volume of approximately 36,300 gal.

3. Shielding

The reactor pool and primary system shielding shall be adequate to meet the applicable personnel radiation protection requirements of 10 CFR 20.

4. Primary Coolant System

The primary coolant system shall conform to the following:

a. Heat Exchanger

The heat exchanger shall be designed to remove heat at the rate generated by the reactor at maximum licensed steady state power from the primary water and shall be designed to perform under the maximum primary system operating temperature and pressure. Replacement heat exchanger shell and tube bundles shall be constructed from stainless steel according to the requirements of Section III, Class C of the ASME Boiler and Pressure Vessel Code.

b. Primary Pump

Number of pumps	1
Type	Horizontal mounted, Centrifugal, Single Suction

Materials of construction	Wortnite
Rating	1500 gpm
Head	59 feet
Design Pressure	75 psig minimum
Design Temperature	150°F minimum
Motor Type	Drip proof, induction, 440 v, 3-phase, 60 cycle

c. Delay Tank

Number of tanks	1
Material of construction	Aluminum Association Alloy 5083 and 5086
Material Thickness	
Walls	0.25 inch
Dished Heads	0.375 inch
Capacity	3000 gal., minimum

d. Primary Recirculation Piping

Material and thickness	Sch. 40 Al. type 3003 aluminum
Size	8 and 10 inch
Design temperature	150°F, minimum
Design pressure	100 psig, minimum

e. Make-up System

A check valve shall be installed in the line between the potable water supply and the make-up and cleanup demineralizer to prevent entry of potentially contaminated water into the potable water supply.

Water source	Potable water from city main
Make-up demineralizer type	Mixed-bed single shell, regenerative
Make-up demineralizer capacity	
Normal	25 gpm
Emergency	50 gpm
Water softener capacity	
Normal	50 gpm

f. Cleanup System for Primary Coolant Water

Cleanup pump	
Capacity	40 gpm
Head	100 ft
Cleanup demineralizer	
Type	Mixed-bed, single shell, regenerative
Cleanup demineralizer capacity	
Normal	40 gpm
Emergency	50 gpm

D. SECONDARY COOLANT SYSTEM

The secondary coolant system shall carry the heat rejected from the primary coolant at the heat exchanger to the atmosphere at cooling towers. It shall be composed of the heat exchanger, cooling towers, pumps and associated valves, piping and sensors. In this system, water flows from the heat exchanger through a control valve to the cooling towers. From the cooling tower basins, the water is then pumped back to the heat exchanger.

E. EMERGENCY CORE COOLING SYSTEM

An emergency core cooling system shall be in place to provide a minimum of 4 GPM directly to the core grid box for a minimum duration of 6 hours.

E. REACTOR CORE AND CONTROL ELEMENTS

The reactor core and control elements shall have the following characteristics and nominal dimensions:

1. Principal Core Materials

Fuel matrix	U_3Si_2-Al dispersion
U-235 enrichment	Approximately 20%
Fuel clad	6061 aluminum
Fuel element side plates	6061 aluminum
End fittings	356-T6 or 6061 aluminum
Moderator	Water
Reflector-Graphite	AGOT grade (or equivalent graphite and/or water
Reflector-Beryllium	Beryllium-aluminum clad
Control elements	Mixture of B_4C and aluminum, clad with aluminum
Servo Element	Stainless steel 314

2. Fuel Elements

Plate width overall	2.81 inches
Active plate width	2.4 inches maximum
Plate length overall	25 inches
Active plate length	23.5 inches
Plate thickness	0.06 inch
Clad thickness	0.02 inch
Fuel matrix thickness	0.02 inch
Water gap between plates	0.1 inch

Number of plates per fuel element	22
U-235 per fuel element	275 grams, nominal
Overall fuel element dimensions	3 in x 3 in. x 40 in.

3. Reflector Elements - Graphite and Beryllium

Overall reflector element dimensions, nominal	3 in x 3 in. x 40 in.
Nominal clad thickness	.1 in.
Nominal graphite dimensions	2.8 in. x 2.8 in. x 28.7 in.
Nominal Beryllium dimensions	2.94 in. x 2.94 in. x 29 in

4. Control Elements

Width	10.6 in.
Thickness	0.38 in.
Overall length	54.1 in.
Active length	52.1 in.

5. Servo Regulating Element

Shape	Square stainless steel
Width	2.1 in.
Overall length	28.8 in.
Active	24.9 in.

6. Control Element Drive

Type	Electromechanical screw
Drive to safety element connection	Electromagnet
Stroke	32 in. maximum

7. Servo Regulating Element Drive

Type	Electromechanical screw
Drive to element connection	Lock screw (no scram)
Stroke	26 in. maximum
Position indication accuracy	± 0.02 in.

8. Neutron Sources

Start-up Source	
Number	2
Type	Plutonium-beryllium
Unit Source Strength	1×10^6 neutrons/sec. minimum
Maximum Power Level with Plutonium-beryllium sources installed	10 Kw
Operational Source	
Number	1
Type	Antimony-beryllium
Source Strength	2×10^6 neutrons/sec. minimum

F. REACTOR SAFETY SYSTEMS

1. Modes of Power Operation

There shall be two modes of power operation:

a. Power Operation - Natural Circulation (NC)

Power operation - NC shall be any reactor operation performed with the reactor cooling provided by natural circulation. The reactor power shall not exceed 0.1 MW during NC operation.

b. Power Operation - Forced Circulation (FC)

Power operation - FC shall by any reactor operation performed with reactor cooling provided by forced circulation. The reactor power shall not exceed 2 MW during FC operation.

2. Design Features

a. The Reactor Control System

The reactor safety system shall consist of sensing devices and associated circuits which automatically sound an alarm and/or produce a reactor scram. The systems shall be designed on the fail-safe principle (de-energizing shall cause a scram). Table F.1 and F.2 describe the arrangement and requirements of the safety system.

b. Process Instrumentation

Process instrumentation with readout in the control room shall be provided to permit measurement of the flow rate, temperature, and conductivity of the primary coolant and the flow rate of the secondary coolant. In addition, a second primary flow indicating device with readout in the control room shall be located between the reactor outlet plenum and the reactor outlet header.

After normal working hours, an independent protection system, separate from the system described in Section K.3.a, shall be used to monitor certain items in the reactor building and alarm in the event of an abnormal condition. The alarm channels provided are:

(1) A fire in the reactor room,

(2) A fire in a location other than the reactor room,

(3) A decrease of 2 inches in reactor pool water level,

(4) A power failure in the reactor building,

(5) An alarm condition from the radiation monitors reading out in the control room,

(6) An alarm condition from any other selected feature.

c. Master Switch

A key lock master switch shall be provided with three positions; "off", "test", and "on". These positions shall have the following functions:

(1) The "off" position shall de-energize the reactor control circuit

(2) The "test" position shall energize the reactor control circuit exclusive of the control blade magnets.

(3) The "on" position shall energize the reactor control circuit including the control blade magnets.

d. Power Level Selector Switch

A power level selector switch shall be provided with four positions: "0.1 MW", "1 MW", "2 MW", and "5 MW". These positions shall have the following functions:

(1) The "0.1 MW" position shall activate all safety system sensors except those indicated in Table F.1.

(2) The "1 MW" and "2 MW positions shall activate all safety system sensors.

(3) The "5 MW" position shall scram the reactor.

e. Control Element Withdrawal Interlocks

Interlocks shall prevent control rod withdrawal unless all of the following conditions exist:

(1) The master switch is in the "on" position,

(2) The safety system has been reset,

(3) The Log N amplifier switch is in the "operate" position,

(4) The startup channel neutron count rate is three counts per second or greater, and

(5) The start-up counter is not being withdrawn.

It shall not be possible to withdraw more than one control element at a time.

f. Servo System Control Interlock

Interlocks shall prevent switching to servo control unless the period as indicated by the Log N channel is thirty seconds or greater. The Servo control system shall be designed so that immediately following a scram the Servo control shall automatically return to the manual mode of operation.

TABLE F.1 - REACTOR SAFETY SYSTEM

Sensor or Trip Device	No. of Switches or Sensors	Trip Set Point	Alarm Set Point
Short Period	1	3 sec. min.	7 sec. min.
High Neutron Flux	2	Max. of 120% of full scale with a 2.4 MW max.	110% max.
High Temperature of Primary Coolant Entering Core During Forced Convection Cooling*			113°F max.
High Temperature of Primary Coolant Leaving Core During Forced Convection Cooling*		125°F max.	123°F max.
Low Flow Rate of Primary Coolant*	1	1580 gpm, min.	1650 gpm, min.
Low Pool Water Level	1	2" max.decrease	2" max.decrease
Seismic Disturbance	1	IV on Modified Mercalli Scale max.	
High Pool Temp	1	125°F	120°F
Bridge Misalignment*	1	X	X
Coolant Gates Open*	1 per gate *	X	X
Neutron Detector High Voltage Failure in Linear Level Safety Channels	1 per power supply	Decrease of 50 volts max.	
Manual Scram (Switch at bridge and on console)	2	X	X
High Conductivity of Primary Coolant	1		Equivalent to 2μmho/cm at 25°C, max.
Safety Blade Disengaged	1		X
Log N - Period Amplifier Failure	1	X	X
Regulating Rod at Either Limit of Travel	1		X
Low Flow Rate of Secondary Coolant*	1		800 gpm, min.
Bridge Movement	1	X	X
No Flow Thermal Column*	1	X	X

*These functions are bypassed when the Power Level Selector Switch is in the "0.1 MW" position.

TABLE F.2

REACTOR NUCLEAR INSTRUMENTATION

Channel	Detector	Sensitivity	Range	Information to Operator	Information to Logic Element (Scram)	Information to Servo System	Recorded Information
Start-up	Retractable gas filled B-10 filled proportional	Neutrons- approximately 12 counts/nv	Source level to full power	Neutron Flux	None	None	Relative power level on log scale
Log N	Fixed fission counter	Neutrons- approximately .7 cps/nv	Source level to $3x10^6$ watts	Power level Period	Period scram	None	Power level log scale and period
Linear level safety	Compensated ion chamber	Neutrons- approximately $4x10^{-14}$amp/nv	1 watt to $3x10^6$ watts	Power level	Level scram	Power level	Power level linear scale (either (channel)
Linear level safety	Compensated ion chamber	Neutrons- approximately $4x10^{-14}$amp/nv	1 watt to $3x10^6$ watts	Power level	Level scram	None	

G. WASTE DISPOSAL AND FACILITY MONITORING SYSTEMS

1. Waste Disposal Systems Design Features

a. Liquid Radioactive Waste Disposal System

All liquid waste (except sanitary waste) from the reactor building shall flow to retention tanks. These tanks shall be located either underground with a dirt cover or in a locked room(s) in the reactor building.

b. Gaseous Radioactive Waste Disposal System

All gaseous radioactive waste from the beam ports, thermal column, pneumatic irradiation system and all other radioactive gas exhaust points associated with the reactor itself shall be collected in a manifold and discharged to the reactor stack through an absolute filter, blower and damper.

c. Solid Radioactive Waste Storage

Solid Radioactive wastes shall either be stored in radioactive waste storage containers located within the reactor building or removed from the site by a commercial licensed organization.

2. Area and Exhaust Gas Monitor Design Features

a. Three fixed gamma monitors employing suitable detectors shall be employed in the reactor building. Each of these shall have the following characteristics:

1) A range consistent with the expected radiation levels in the area to be monitored (0.01 to 10 mr/hr, 0.1 to 100 mr/hr, or 1 to 1,000 mr/hr).

2) A radiation dose rate output indicated in the control room.

3) An adjustable high radiation alarm which shall be annunciated in the control room.

4) The three fixed gamma monitors shall be located to detect radiation as follows: At the pool biological shield between a beam port and the thermal column, above the storage container for new fuel elements, and at the reactor bridge.

b. A gamma monitor shall be provided near the primary coolant system, and an additional one shall be provided near the secondary coolant system for use in determining the presence of abnormally high concentrations of radioactivity in these systems. The characteristics of these monitors shall be as stated in a. above.

c. Six additional direct reading area monitors employing Geiger tube detectors shall be provided to monitor the pneumatic system receiver stations, the beam port areas, and other areas as required. Each of these shall have the following characteristics:

1) A range consistent with expected radiation levels in the area being monitored (0 to 10 mr/hr or 0 to 50 mr/hr).

2) A radiation dose rate output at the instrument.

3) An adjustable high radiation alarm to alarm at the instrument and create both an audible and visual signal.

d. A stack exhaust gas monitor system shall be provided which draws a representative sample of air from the exhaust gas. The monitor with indicators and alarms in the control room, shall have the following characteristics:

1) A beta particulate monitor with an alarm.

2) A gas monitor incorporating a scintillation detector with high level alarm and a minimum detectability level for an Argon-41 concentration in air of $10^{-6}\mu c/cc$. The monitor shall have a range of at least four decades.

3. <u>Other Radiation Monitoring Equipment</u>

a. Portable survey instruments for measuring beta-gamma dose rates in the range from .01 mr/hr to 250 r/hr shall be available at the facility. Portable instruments for measuring fast and thermal neutron fluxes in the range from 1 n/cm^2-sec to 25,000 n/cm^2-sec shall also be available to the facility.

b. Reactor excursion monitors shall be placed in the facility for measuring gamma and neutron doses in the event of an accident.

c. A radiation monitor shall be provided to monitor all persons leaving the reactor room for beta-gamma contamination.

4. High Radiation Area

During reactor operation, the dose rate from the delay tank may be in excess of 100 millirem per hour. On three sides, the tank shall be shielded. On the fourth side, the tank is shielded using a "maze" so that access to the tank is possible through a door equipped with a lock.

H. FUEL STORAGE

1. New Fuel Storage

New fuel shall be stored in a security container in "egg crate" boxes. Sheet cadmium at least 0.020 inches thick shall be fastened around the outside of the boxes in the region which contains the fuel. The number of fuel elements which can be placed in each box shall not exceed three. For all conditions of moderation possible at the site K_{eff} shall be less than 0.8.

2. Irradiated Fuel Storage

Two types of irradiated fuel element storage racks shall be provided. One type of rack shall contain spaces for nine fuel assemblies and shall have approximate over-all dimensions of 35.5 in. wide by 26 in. high by 6.25 in. thick, and shall be fixed to the pool wall. At least two of these racks shall be provided. The second type of rack shall consist of two of the nine fuel assembly racks described above attached together with a minimum space between the center lines of fuel assemblies in adjacent racks of 12 inches. This 18 fuel assembly rack shall be covered on the two 35.5 x 26 in. outside faces with a neutron absorbing material. At least one 18 fuel assembly rack shall be provided, and the rack may be moved within the pool. The fuel storage racks may also be used to store core components other than fuel assemblies. The irradiated fuel storage racks shall have a maximum K_{eff} of 0.8 for all conditions of moderation possible at the site. Storage spaces shall be provided for at least 36 fuel assemblies.

I. EXPERIMENTAL FACILITIES

The permanent experimental facilities shall consist of the following:

1. Thermal column.

2. Beam ports; two 8 inch dia, and four 6 dia.

3. A six inch diameter through port.

4. Radiation baskets.

5. A two-tube pneumatic tube system.

6. Dry gamma cave.

J. ADMINISTRATIVE AND PROCEDURAL SAFEGUARDS

1. Organization

The Rhode Island Atomic Energy Commission (RIAEC) shall have the responsibility for the safe operation of the reactor. The RIAEC shall appoint a Director of Operations and a Reactor Utilization Committee consisting of a minimum of five members, as follows:

(1) The Director of Operations

(2) The Reactor Facility Health Physicist

(3) A qualified representative from the faculty of Brown University

(4) A qualified representative from the faculty of Providence College

(5) A qualified representative from the faculty of the University of Rhode Island.

A qualified alternate may serve in lieu of one of the above.

The Director and Health Physicist are not eligible for chairmanship of the Committee. The Reactor Utilization Committee shall have the following functions:

a. Review proposals for the use of the reactor considering the suitability of the reactor for the proposed use and the safety factors involved.

b. Approve or disapprove proposed use of the reactor.

c. Review at least annually the operating and emergency procedures and the overall radiation safety aspects of the facility.

The Reactor Utilization Committee shall maintain a written record of its findings regarding the above.

2. Qualifications of Personnel

a. The Director of Operations shall have at least a bachelors degree in one of the physical sciences or engineering, and he shall be trained in reactor technology and be a licensed senior operator.

b. The staff Health Physicist shall be professionally trained and shall have at least a bachelors degree in one of the physical or biological sciences or engineering. He shall have experience such as may have been gained through employment in a responsible technical position in the field of health physics.

c. The reactor operators and senior operators shall be licensed in accordance with the provisions of 10 CFR 55.

d. In the event of temporary vacancy in the position of Director of Operations or the Health Physicist, the functions of that position shall be assumed by qualified alternates appointed by the RIAEC.

3. Responsibilities of Personnel

a. Director

(1) 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.

(2) 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 Reactor Utilization Committee.

b. Senior Reactor Operators

(1) A licensed senior reactor operator 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.

(2) The reactor operations which affect core reactivity shall not be performed without the senior operator on duty or readily available on call. The senior 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 operator as well as the time he assumes the duty shall be entered in the reactor log. When the senior operator is relieved, he 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 operator being relieved of his duty shall insure that all pertinent information is logged. The senior operator assuming duty shall check the log for information or instructions.

c. Reactor Operators

(1) The responsible senior operator shall designate for his shift a licensed operator (hereafter called "operator") who shall have primary responsibility under the senior 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 operator.

(2) 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 assumed duty.

(3) 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 to leave the control room during such an operation, he 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.

(4) The operator, under the senior operator on duty, shall be responsible for the operation of the reactor according to the approved operating schedule.

(5) 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 judgement 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.

d. Health Physicist

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

4. Written Instructions and Procedures

Detailed written operating instructions and procedures shall be prepared for all normal operations and maintenance and for emergencies. These procedures shall be reviewed and approved by qualified personnel before use. Each member of the staff shall be familiar with those procedures and instructions for which he has responsibility.

5. Site Emergency Plans

The Rhode Island Nuclear Science Center shall have available the services of other state agencies for dealing with certain types of emergencies. The RIAEC shall enter into an agreement with the Rhode Island Civil Defense Agency whereby the Civil Defense Agency will maintain an emergency monitoring and communications vehicle which they shall make available to the Nuclear Science Center in the event of an emergency involving release of fission products or other radioactive isotopes to the atmosphere. The emergency vehicle shall contain equipment such as portable radiation monitors, respirators, and a particulate air sampler. Communications using the statewide emergency network shall be available.

Personnel of the Civil Defense Agency and of local fire departments shall have received training from the Civil Defense Training Officer in the use of certain radiological instruments. Future training shall be augmented by including orientation on the reactor facility.

K. OPERATING LIMITATIONS

1. General

The following administrative controls shall be employed to assure the safe operation of the facility:

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

b. The reactor control and safety system must be turned on and functioning properly and an appropriate neutron source must be in the core during any change which can affect core reactivity.

c. During operations which could affect core reactivity, a licensed operator shall be stationed in the control room. Communications between the control room and the senior reactor operator directing the operation shall be maintained.

d. The operator shall not attempt to start up the reactor following an automatic scram or unexplained power decrease until the senior operator has determined the cause of the scram or power decrease and has authorized a start-up.

e. The reactivity of all core loadings to be utilized in operating the reactor shall be determined using unirradiated fuel elements or elements containing fission products in which the effect of xenon poisoning on total core reactivity has decayed to 0.05% delta k/k or less.

f. Critical experiments shall be performed under the supervision of the Director or other competent supervisory scientist licensed as a senior reactor operator. During the experiment there shall be present, in addition to this licensed supervisor, at least one other technically qualified person who shall act as an independent observer. Each step in the procedure shall be considered in advance by both persons, each calculation shall be checked by both persons, and no step shall be taken without the concurrence of both. A written record shall be made at the time of each fuel element addition or other core change which could significantly affect core reactivity.

g. The basic operating principles for the assembly and reloading of cores whose nuclear properties have been previously determined from critical experiments shall be as follows:

All core loading changes shall be performed under the supervision of a person having a senior operator's license. During the operation there shall be present in addition to the designated senior reactor operator at least one other technically qualified person who shall act as an observer.

The exact procedure to be followed for a particular reloading operation will be determined by the observer and the senior reactor operator in charge of the operation before the operation begins. Each step in the procedures shall be considered by both persons, and no step shall be taken without the concurrence of both.

2. Experiments

a. "Experiments" as used in this section shall be construed as any apparatus or device installed in the core region which is not a component of the core.

b. The Reactor Utilization Committee shall review and approve all experiments before initial performance at the facility. 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 Reactor Utilization Committee. The documentation shall include at least:

(1) The purpose of the experiment,

(2) A description of the experiment, and

(3) An analysis of the possible hazards associated with the performance of the experiment.

c. All use of experimental facilities shall be approved by the Director of Operations.

d. The absolute value of the reactivity worth of any single independent experiment shall not exceed 0.006. If such experiments are connected or otherwise related so that their combined reactivity could be added to the core simultaneously, their combined reactivity shall not exceed 0.006.

e. The calculated reactivity worth of any single independent experiment not rigidly fixed in place shall not exceed 0.0008. If such experiments are connected or otherwise related so that their combined reactivity could be added to the core simultaneously, their combined reactivity worth shall not exceed 0.0008.

f. No experiment shall be installed in the reactor in such a manner that it could shadow the nuclear instrumentation system monitors and thereby give erroneous or unreliable information to the control system safety circuits.

g. No experiment shall be installed in the reactor in such a manner that it could fail so as to interfere with the insertion of a reactor control element.

h. No experiment shall be performed involving materials used in such a way that they might credibly result in an explosion.

i. No experiment shall be performed involving materials which could credibly contaminate the reactor pool causing corrosive action on the reactor components.

j. Experiments shall not be performed involving equipment whose failure could credibly result in fuel element damage.

k. There shall be no more than one vacant fuel element position within the periphery of the active section of the core.

3. Operations

a. Site

Control of access to the reactor facility shall be the responsibility of the Director of Operations.

b. Containment

(1) During any operation in which the control rods are withdrawn from the core containing fuel, the following conditions shall be satisfied:

a. Confinement building penetrations which are not designed and set to close automatically on actuation of the evacuation button shall be sealed, except that doors other than the truck door may be opened during reactor operation. If a door is to remain open, an individual from the reactor operations staff is continuously in attendance at the door.

b. The building clean-up system is operable.

(2) Requirements for Retest of Confinement

(a) Method of Retest

The building cleanup system shall be retested by pressing an evacuation button and observing that the following functions occur automatically:

1. Evacuation horn blows.

2. air conditioning and normal ventilation has turned off.

3. Dampers on all ventilating ducts leading to the outside have closed.

4. Building cleanup system-air scrubber and basement chem lab blower come on.

5. The negative differential pressure between the inside and outside of the building is at least 0.5 inches of water. This shall be determined by reading the differential magna helic gauge located in the control room.

(b) Frequency of Retest

The building cleanup system including the auxiliary electrical power system shall be retested at least weekly.

(3) The exhaust rate through the cleanup system shall not exceed 4500 cfm with not more than 1500 cfm coming from the reactor building and passing through the charcoal scrubber. The remaining air will be provided by a separate blower from an uncontaminated source. This shall create a pressure in the building which is equivalent to at least 0.5 inch of water below atmospheric pressure.

c. Primary Coolant System

(1) The minimum depth of water above the top of the active core shall be 23 feet.

(2) No piping shall be placed in the pool which could cause or fail so as to cause a siphon of the pool water to below the level of the ten inch coolant line penetrations.

(3) Makeup System

The effluent water of the primary coolant water makeup system shall be of a quality to insure compliance with K.3.c.(5) and (6) below.

(4) Cleanup System

The effluent water of the primary coolant water clean up system shall be of a quality to insure compliance with K.3.c.(5) and (6) below.

(5) The primary coolant shall be sampled at a minimum frequency of once per week and the samples analyzed for gross radioactivity, pH, and conductivity in accordance with written procedures. Corrective action shall be taken to avoid exceeding the limits listed below:

pH	5.5 to 7.5
conductivity	2 μmho/cm

(6) The radioactive materials contained in the pool water and in the primary coolant water shall be such that the radiation level one meter above the surface of the pool shall be less than 10 mrem/hr.

(7) During the forced circulation mode of operation, the primary coolant flow rate shall not be less than 1580 gpm.

d. Secondary Cooling System

(1) The secondary coolant shall be sampled at a minimum frequency of once per week and the samples analyzed for pH in accordance with written procedures. Corrective action shall be taken to avoid exceeding the pH limit given below:

pH	5.5 to 9

(2) The concentration of radionuclides in the secondary water shall be determined at least once each day the reactor operates using forced convection cooling. The concentration shall be determined at least once per week when not being operated using forced convection cooling.

(3) If the radioactive materials contained in the secondary coolant exceed a radionuclide concentration in excess of the values in 10 CFR 20, Appendix B, Table I, Column II, above background, the reactor shall be shutdown and the condition corrected before operation using the secondary cooling system resumes.

(4) The secondary coolant system shall be placed in operation as required during power operation utilizing forced convection in order to maintain a primary coolant core outlet temperature of 125°F or below.

e. Reactor Core and Control Elements

(1) The reactor shall not contain in excess of 35 fuel elements. There shall be a minimum of four operable control elements.

(2) The limiting thermal and hydraulic core characteristics based on a 14 element, graphite and beryllium reflected core are specified below:

(a)	Maximum Heat Flux	.424 MW/M^2
(b)	Maximum Core Specific Power	519.48 W/gU^{235}
(c)	Maximum Fuel Surface Temperature	110°C
(d)	Coolant Velocity during Forced Convection Cooling	1.48 M/sec
(e)	Coolant Inlet Temperature	115°F max.
(f)	Average Coolant Temperature Rise	10°F max.
(g)	Primary System Bulk Outlet Coolant Temperature	125°F max.
(h)	Temperature Margin in Primary Coolant ($T_{sat}-T_{surf}$)	5.8°C
(i)	Number of Coolant Passes Through Core	1

(3) Principal Nuclear Characteristics of the Core

(a) Core and Control System Reactivity Worth

1. The reactor shall be subcritical by at least 1% $\Delta k/k$ from the cold, Xe-free, critical condition with the most reactive control element and the servo regulating element fully withdrawn.

2. The maximum worth of the servo regulating element shall be 0.7% $\Delta k/k$.

(b) Maximum Reactivity Addition Rate - $\Delta k/k/sec$

1.	By servo regulating element	maximum of 0.0002
2.	Manual by control element	maximum of 0.0002

(c) Reactivity Coefficients

1. Temperature coefficient — approximately $.82 \times 10^{-4}/^{o}C$ (calculated) density only

2. Void coefficient (core average) — approximately $-2.7 \times 10^{-3}/\%$ void (calculated)

(4) Principal Core Operating Limitations

(a) Maximum Pool Temperature Limitations

The pool water temperature shall not exceed 125°F. The pool water temp shall be monitored with readout in the Control Room. A trip and alarm shall be included in the system.

(b) Reactivity Limitations

1. Excess Reactivity

The cold, clean excess reactivity for any core used in the reactor shall not exceed 0.047.

2. Minimum Shutdown Margin

All reactor cores used shall be such that they would be subcritical if any single control element and the servo regulating element were withdrawn.

(c) Reactivity Coefficient Limitation

The reactor power coefficie (as inferred by the control rod movements required to compensate for changes in power) shall be negative.

(d) Control Element Drive Performance Requirements

All control element drives shall meet the following specifications:

1. The control drive withdrawal rate shall not be more than 3.6 inches per minute.

2. For the electronic scram system, the time from initiation of a scram condition until control element release shall not exceed 100 milliseconds.

3. The time from initiation of a scram condition until the control element is fully inserted shall not exceed 900 milliseconds.

4. It shall be demonstrated at least every 3 months that the above specifications are met.

(e) Servo Regulating Element Drive Performance Requirements

If in use during operation, the servo regulating element drive shall meet the following specifications:

1. The drive withdrawal rate shall not be more than 78 inches per minute.

2. It shall be demonstrated at least once per month that the above specification is met.

f. Reactor Safety Systems

(1) The reactor safety system shall be operable during all reactor operation. The safety system shall be checked out before each start-up and functionally tested for calibration at least monthly.

(2) It shall be permissible to continue operations with one or more of the safety system functions that produce only an alarm temporarily disabled providing that additional procedural controls are instituted to replace the lost safety system alarm function(s).

(3) The control element withdrawal interlocks and the servo system control interlocks shall be functionally tested at least once per month.

(4) During reactor startup or during mechanical changes that could affect core reactivity, the startup range neutron monitoring channel shall be operable and shall provide a neutron count rate of at least 3 counts per second with a signal to noise ratio at least 3 to 1.

(5) The linear level safety channels shall not read less than 15% of full scale when the reactor is operating at power levels above 1 watt

(6) Following a reduction in power level, the operator shall adjust the servo power schedule to the new power level before switching to automatic operation.

(7) An alarm condition from any one of the items listed in Section F.2.b. after working hours shall transmit coded information to a continuously manned central station in Providence, Rhode Island. The central station shall be provided with written instructions on the steps to be taken following an alarm.

9. Waste Disposal and Reactor Monitoring Systems

(1) The liquid waste retention tank discharge shall flow to a monitor station in the reactor building where the effluent shall be batch sampled and the gross activity per unit volume determined before release. All off-site releases shall be directly into the municipal sewer system.

(2) Gaseous radioactive waste shall be disposed of using the reactor stack. Disposal limits shall conform to the following table. In this table, the MPC stated is for individual isotopes and mixtures contained in Column 1, Table II, Appendix B of 10 CFR 20

	1	2
Type of Activity	Maximum Curies per second to be released	Curies per second to be released averaged over one year
Particulate Matter and Halogens with half-lives longer than 8 days	140 X MPC (uc/cc)	14 X MPC (uc/cc)
All other Radioactive Isotopes	10^5 X MPC (uc/cc)	10^4 X MPC (uc/cc)

(3) All radioactive liquid and solid wastes disposed of off-site shall be within the limits established by 10 CFR 20 or shall be removed from the site by a commercial licensed organization.

(4) The exhaust gas monitor shall be calibrated to alarm at an instantaneous release rate which instantaneously exceeds the limits stated in Column 2 for the annual average release rate. If the maximum permissible stack release rate stated in Column 1 is exceeded, the reactor shall immediately be placed in the shutdown mode of operation and the situation investigated.

(5) The area, primary and secondary coolant system and the exhaust gas monitors shall be in operation at all times when control elements or the servo regulating elements are withdrawn; however, individual area coolant system monitors may be taken out of service for maintenance and repair if replaced with portable radiation detection equipment. Adequate spare parts shall be on hand to allow necessary repairs to be made during the maintenance or calibration outages of the monitors.

(6) The area and the primary and secondary coolant system monitors shall be adjusted to alarm at a maximum reading of 2 mr/hr or 200% of the normal radiation levels in their area, whichever is larger.

(7) The door which controls entrance to the "maze" leading to the delay tank shall be locked with the key in the possession of the Director or a licensed senior operator. Entrance to the delay tank high radiation area shall require the presence of the Health Physicist or a licensed senior operator and the use of direct reading portable radiation monitoring equipment.

h. Fuel Storage

(1) New fuel shall be stored in egg crate boxes located in a security container. Access to the security container shall be restricted, through use of a lock, to the Director of Operations and the licensed senior reactor operators.

(2) Irradiated fuel, not in use in the reactor core, shall be stored in the criticality safe storage racks described in Section H. Only one fuel assembly may be inserted or moved from a storage rack at a time.

(3) Safety against inadvertent criticality shall be provided by limiting the number of fuel assemblies per rack to nine and then positively securing such racks at least 30 cm. apart, or by limiting the number of fuel assemblies to 18 per rack and then covering the two large faces of each rack with a sheet of aluminum covered cadmium.

4. Maintenance

(a) The electronic control and the process control system shall be checked for proper operation and calibration before each reactor start-up. If maintenance or recalibration is required, it shall be performed before reactor start-up proceeds.

(b) Maintenance shall be performed with the approval of the Director. Equipment and system maintenance records shall be kept to facilitate scheduling and completion of all necessary maintenance.

(c) Routine maintenance on all control and process system components shall be performed in accordance with written schedules and with written procedures.