~~CONFIDENTIAL~~

LA-CP-87-2

JAN 18 1987

Case File 908

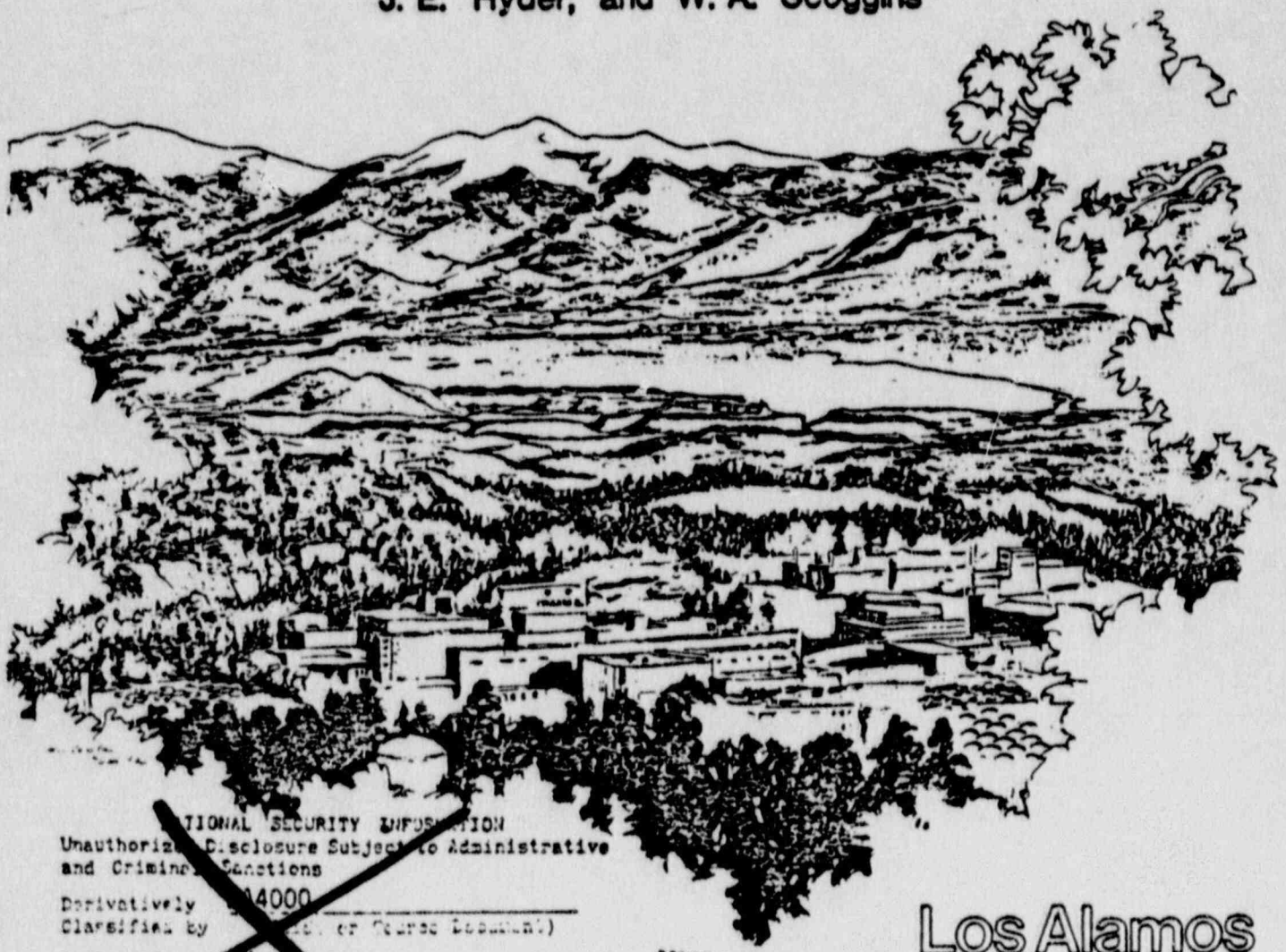
UNCLASSIFIED

NONPOWER REACTOR SABOTAGE STUDY (U)

by

W. D. Zerwekh, A. E. Pope,

J. E. Hyder, and W. A. Scoggins



UNCLASSIFIED

Los Alamos
Los Alamos National Laboratory
Los Alamos, New Mexico 87545

~~CONFIDENTIAL~~

c14

TABLE OF CONTENTS

TABLE OF CONTENTS

UNCLASSIFIED

NONPOWER REACTOR SABOTAGE STUDY

by

W. D. Zerekh, A. E. Pope, J. E. Hyder, and W. A. Scoggins

ABSTRACT

(U) The nine largest Nuclear Regulatory Commission (NRC)-licensed nonpower reactor (those having a thermal power level of 2 MW and greater) facilities were examined to identify actions necessary for sabotage and to estimate the potential radiological consequences of successful sabotage. Fault-tree analysis techniques were used to develop a set of sabotage scenarios that would yield the maximum radiological consequence, but we did not attempt to evaluate the probability of success. Building transport and atmospheric dispersion models also were developed and applied to each facility. Potential thyroid and whole-body exposures were calculated as a function of distance resulting from 1% core melt or 1% core burn of each of the reactors. These calculations can be extrapolated linearly to predict the minimum fraction of core melt necessary to produce an exposure in an uncontrolled area of 25 rem whole-body or 300 rem thyroid.

1.0 INTRODUCTION

(U) The nine reactors studied were all nonpower test, research and training reactors (TRTRs) ranging in power from 2 to 20 MWth. Four of the units were tank-type reactors with MTR plate-type, highly enriched fuel. The remaining were open-pool reactors. Three used MTR plate-type, high-enriched fuel; one used MTR low-enriched fuel; and one used PULSTAR low-enriched fuel pins. The facilities' operating schedules varied significantly.

(C)

(c)

(c)

(U) 2.0 DEFINITION OF PROBLEM UNCLASSIFIED

(C)

(U) The increased frequency of terrorist attacks around the world has raised the interest in the potential vulnerability of TRTRs to sabotage. Therefore, Los Alamos has been requested to assess these vulnerabilities.

(U) 2.1 Nonpower Reactor Vulnerabilities

(U)

(U) 2.2 Selected Case Studies

(U) The nine largest NRC-licensed nonpower reactor (those having a thermal power level of 2 MW and greater) facilities were selected for this study.

(C)

(U) The initial analyses were followed by a study of the Massachusetts Institute of Technology Reactor (MITR), a tank-type 5-MW unit, and the Rhode Island Nuclear Science Center Reactor (NSCR), a 2-MW open-pool facility. Later, the State University of New York (SUNY) Buffalo Materials Research Center (BMRC) Reactor, a 2-MW open-pool unit using 6% enriched PULSTAR-type fuel, and the CINTICHEM, Inc. Reactor, a 5-MW open-pool high-enriched MTR-fuel unit, were reviewed. The study was concluded with examinations of the University of Michigan Ford Nuclear Reactor (FNR), a 2-MW open-pool low-enriched fuel facility, and the University of Virginia Reactor (UVAR), a 2-MW open-pool high-enriched fuel unit.

(C)

(U) 2.3 Study Conservatisms

(U) The various sabotage scenarios were selected to emphasize the potential radiological consequences. Thus, this study is a technical evaluation of required actions, resources, and possible facility vulnerabilities. This study makes no attempt to address the credibility of the various scenarios. To analyze rationally the consequences, a number of conservative assumptions were necessary. The conservatisms can be grouped into four distinct areas: (1) potential core inventory, (2) saboteur knowledge and ability, (3) building release mechanisms and atmospheric conditions, and (4) potential exposures.

(c)

(c)

(c)

(c)

(c)

(c)

(U) 3.0 REACTOR DESCRIPTIONS

(U) All of the detail contained in the descriptions of the individual reactors and their major systems is readily available in the open literature. Most have been extracted directly from the facility SARs and are presented here as a basis for the facility analyses to follow.

(U) 3.1 National Bureau of Standards Reactor

(U) The NBSR is a highly enriched, tank-type, heavy-water-moderated-and-cooled test reactor consisting of 30 MTR-type curved-plate fuel elements arranged in a hexagonal array. The reactor coolant enters the bottom of the vessel through two concentric inlet pipes to plena under the fuel elements. The coolant then flows up through the fuel elements and returns down to outlet pipes that also penetrate the bottom of the reactor vessel. The heat from the primary system is removed through a pair of heat exchangers and a light-water secondary coolant system with a cooling tower. Table I shows the NBSR facility parameters used in the analysis.[5,6] Figures 1 and 2 show the vessel arrangement.

(U) 3.1.1. Fuel Elements. Each of the fuel elements is a rectangular array of 17 curved fuel plates approximately 2-1/2 by 3-1/2 in. (6 by 9 cm) in cross section and have two 11-in. (28-cm) fueled sections separated by a 7-in. (18-cm) central unfueled section. The fuel is a uranium-aluminum mixture containing 0.63 oz (18 g) of ^{235}U (9[illegible]% enriched) per plate; the uranium is divided equally between the upper and lower fueled sections. When fully loaded, the reactor contains 13.2 lb (~6 kg) of fuel. Average fuel burnup is ~55%. Four fuel elements are replaced every 30 days; each element spends ~7-1/2 months in the core.

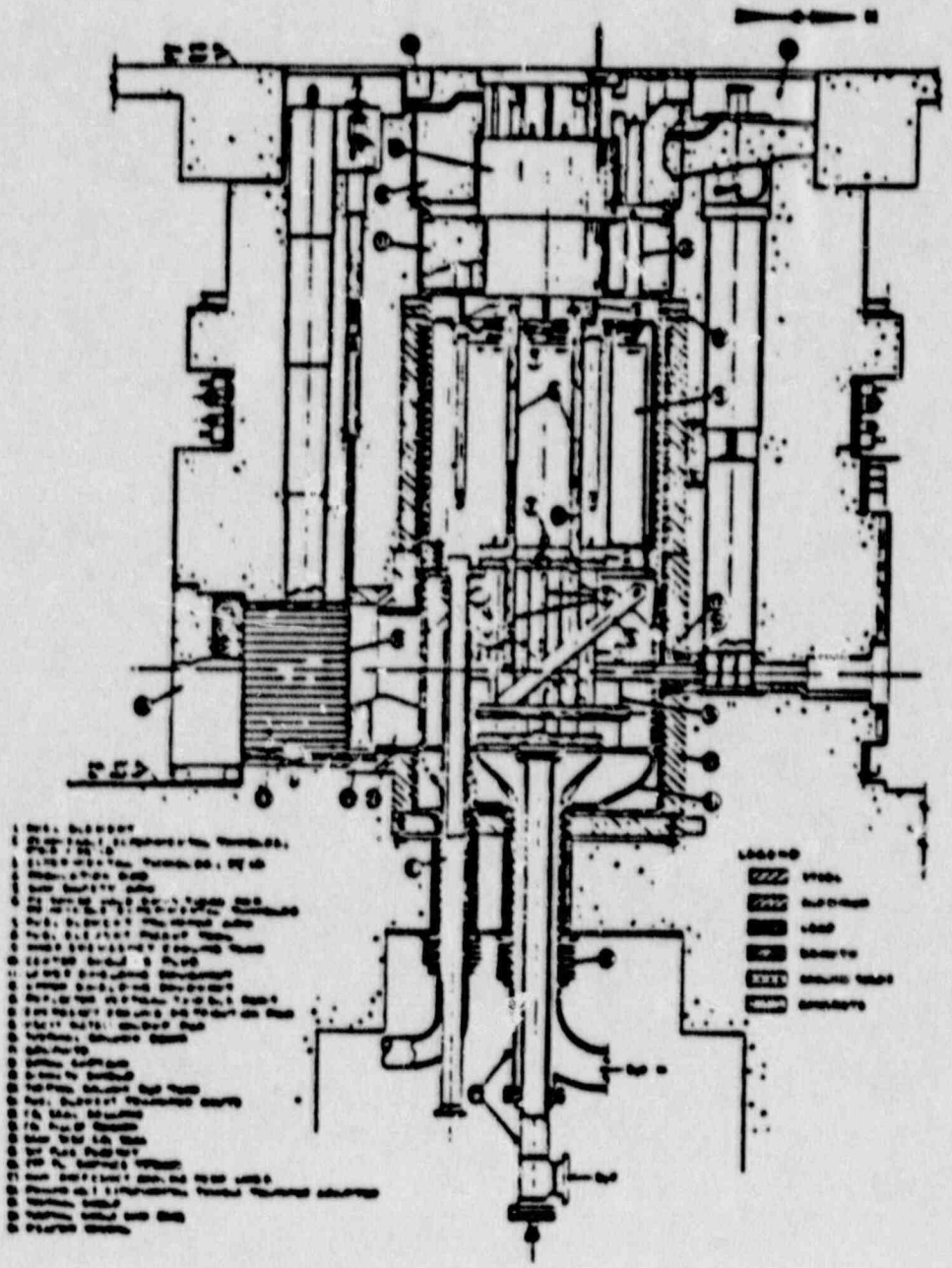
(U) 3.1.2. Reactivity Control System. The reactor is controlled by four shim-safety semaphore-type blades and one aluminum regulating rod. The safety blades are composed of flattened cadmium tubes clad with aluminum and filled with helium. The drive shafts penetrate the reactor vessel and biological shield and connect the shim blades to the drives. The shim drives are large springs that are compressed by ball nuts and screw jacks when the shim blades are raised. The shim-safety blade is raised and lowered as the nut rides up and down the screw, which is driven by an electric motor through an electromagnetic clutch. A reactor trip signal deenergizes the electromagnetic clutches, releasing the springs and driving the blades into the core. The regulating rod uses a reversible two-phase electric motor and lead screw drive unit and has no safety

TABLE I

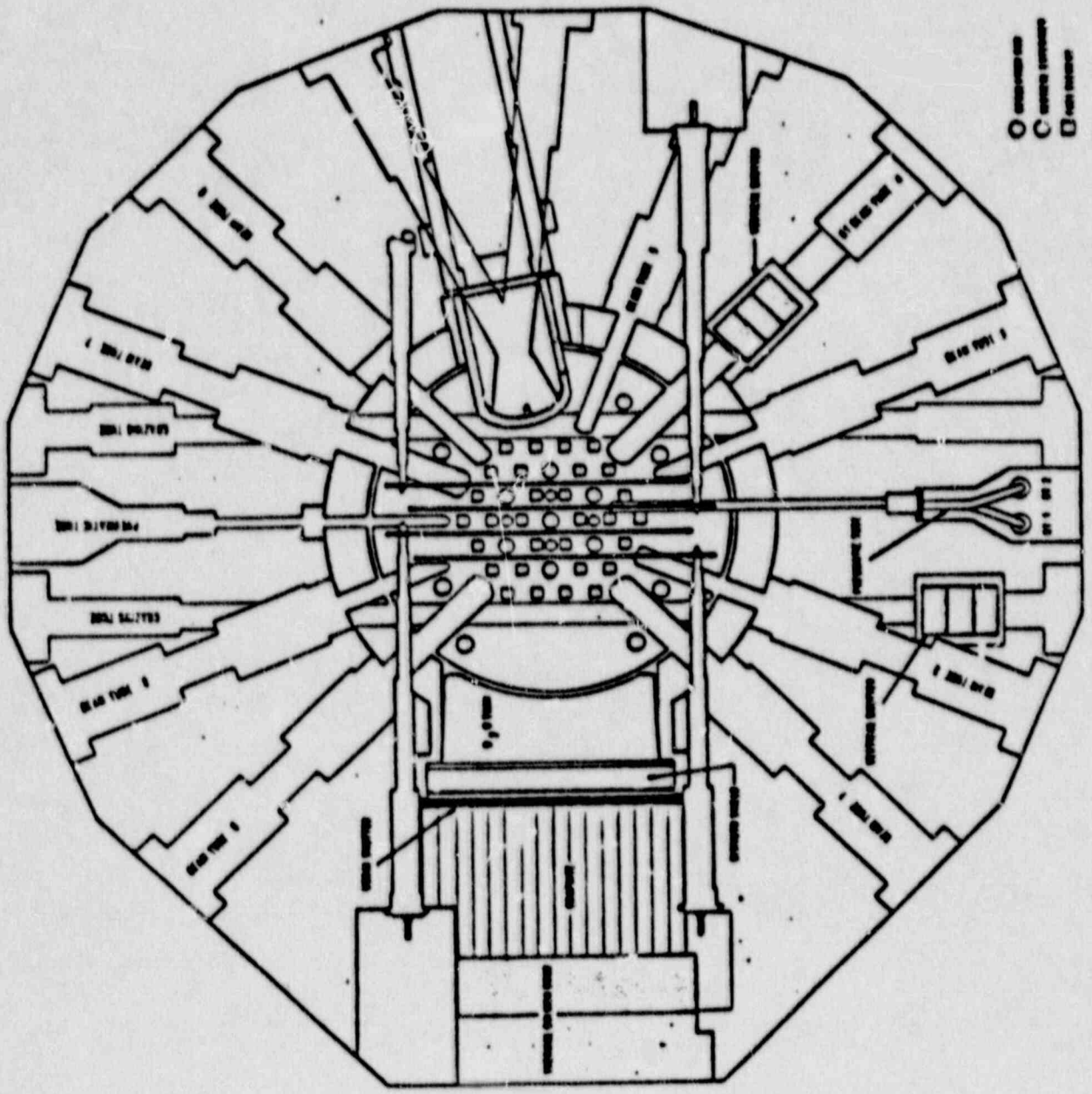
(U) NBSR FACILITY PARAMETERS

Power Level	20 MW

(C)



(U) Fig. 1. NBSR vertical cross section.



(U) Fig. 2. NBSR horizontal cross section.

shutdown function. In the unlikely event that the shim-safety blades cannot be inserted, the reactor can be shut down by dumping the D_2O top reflector down to a level ~1 in. (2.5 cm) above the reactor core.

(U) 3.1.3 Reactor Vessel Internal Arrangement. The core is contained in a 16-ft (4.9-m)-high, 7-ft (2.1-m)-diam aluminum tank. Several experimental irradiation facilities (including radial and through beam tubes, a thermal column, pneumatic facilities, and experiment thimbles) are located around the periphery of the core. The beam tube thimbles are aimed at the unfueled center section, which allows them to be exposed to a high ratio of thermal to fast neutron flux.

(U) 3.1.4. Emergency Core Cooling System. The NBSR emergency core cooling system (ECCS) has three backup systems that ensure continued core cooling in the event of a loss of the D_2O coolant/moderator. These backup systems are (1) the inner emergency cooling tank, (2) the emergency cooling tank, and (3) a tray that surrounds the lower half of the core. The inner emergency cooling tank is inside the reactor vessel and drains automatically on the loss of coolant, dumping to a distribution header on the top grid plate. This tank has an 1100-gal (415R-L) water capacity and can supply emergency coolant for ~30 min. The emergency cooling tank is near the ceiling of the reactor room; it has a 3000-gal (11 340-L) capacity and must be opened manually to supply either the core plenum or the inner emergency cooling tank. The tray surrounding the bottom half of the core provides cooling to the outside of the elements until the decay heat boils away the water captured by the tray.

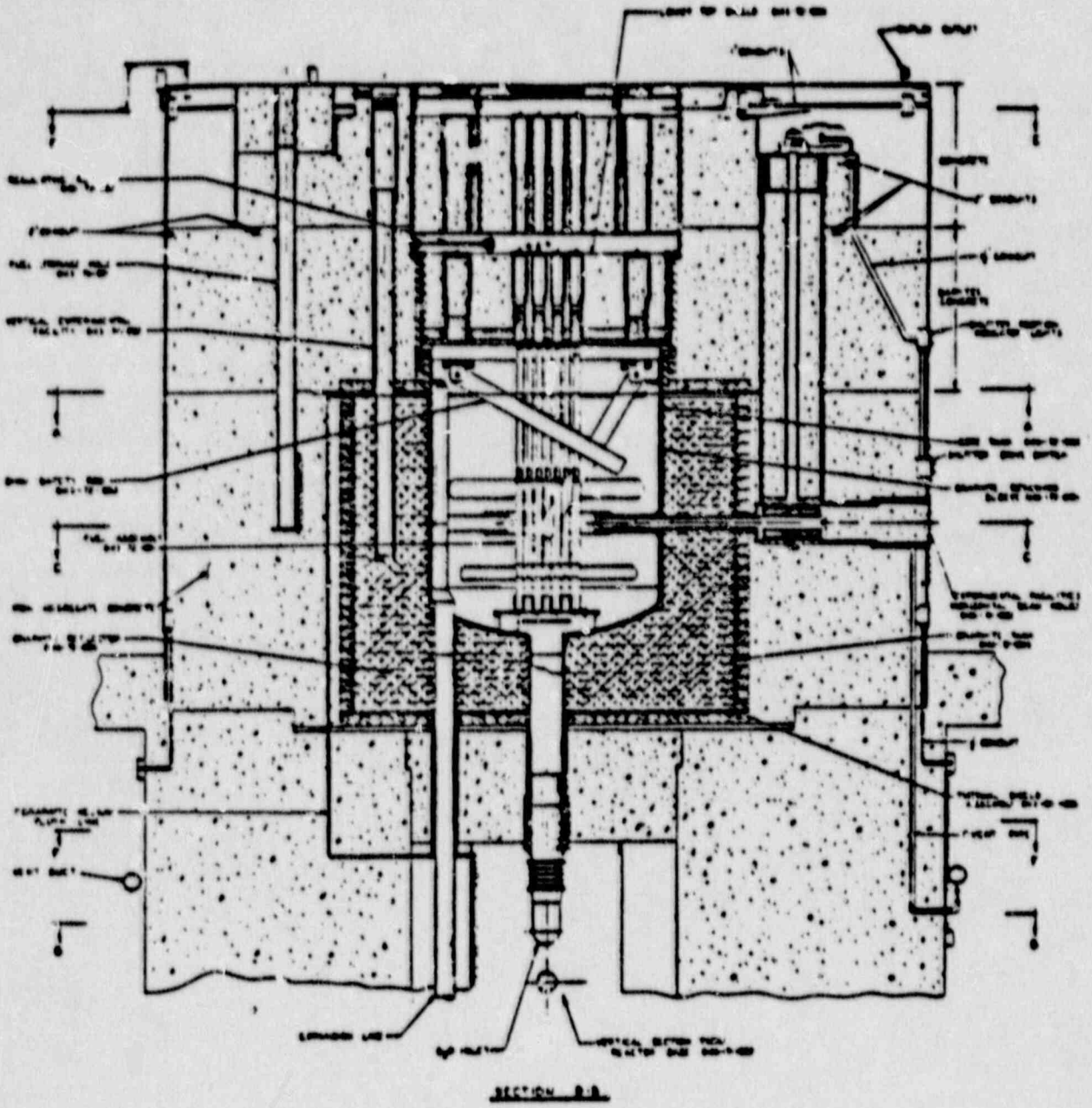
(U) 3.2 Georgia Tech Research Reactor

(U) The GTRR, like the NBSR, is a highly enriched tank-type, heavy-water-moderated-and-cooled research reactor. In addition, it has a graphite reflector outside of the reactor vessel. The core consists of a hexagonal array of 17 MTR-type elements. As in the NBSR, the coolant enters the bottom of the vessel to a plenum under the lower core plate. However, there is only a single plenum in the GTRR. The coolant flows up through the elements and returns down around the core, exiting at the bottom as at the NBSR. The rest of the primary and secondary systems function quite similarly to the NBSR. Table II shows the GTRR facility parameters.[5,7] Figures 3 and 4 show the reactor vessel assembly.

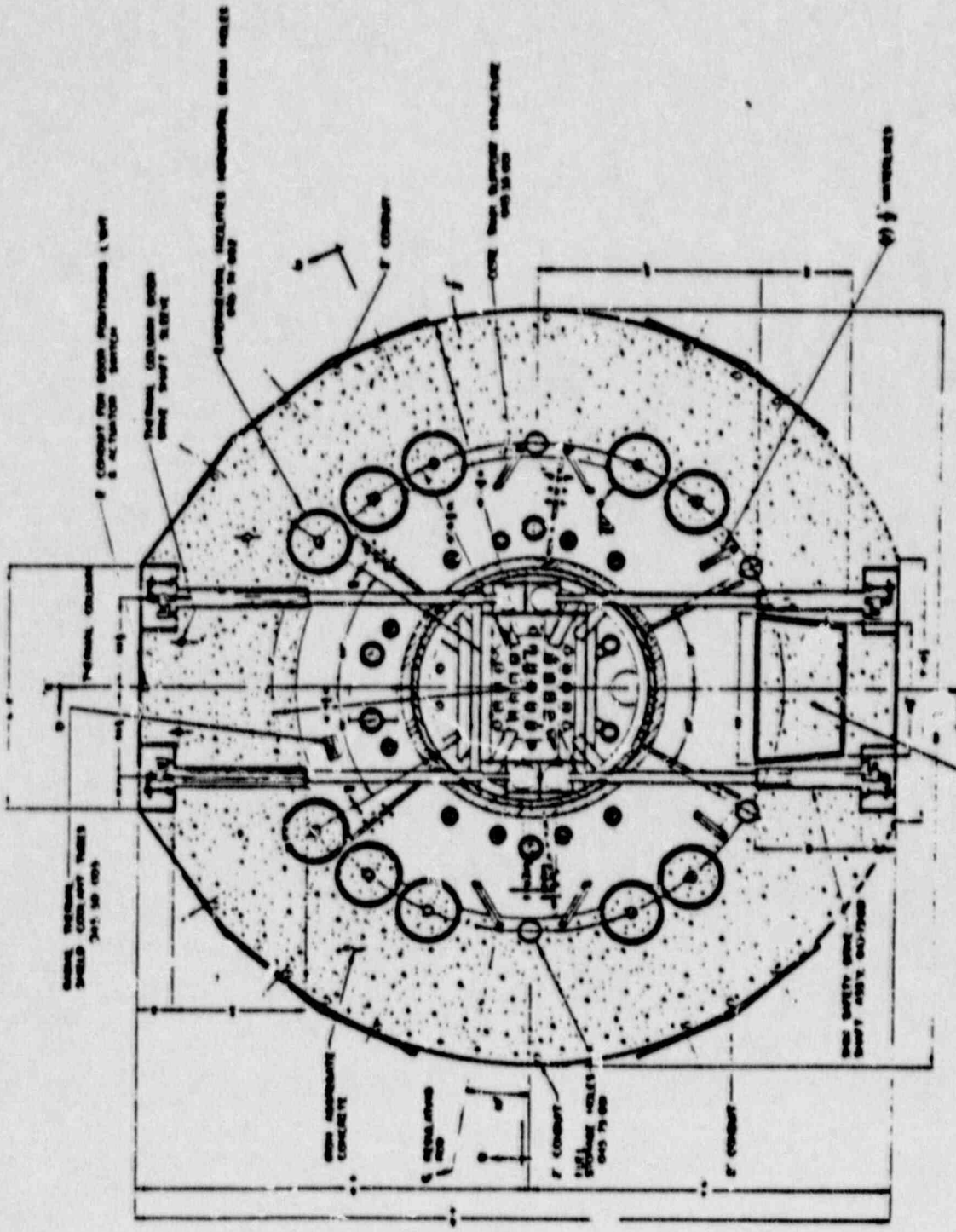
TABLE 11

(U) GTRR FACILITY PARAMETERS

Power Level	5 MW
(C) [redacted]	
Moderator and Coolant	Heavy water (D_2O)
Reflector	Heavy water (D_2O) and graphite
Coolant Flow	1800 gal/min (113.6 L/s)
Fuel Type	MTR plate with unfueled side plates
Enrichment	93% ^{235}U
Number of Elements	19
Arrangement	Hexagonal array on 7-in. (17.8-cm) centers
Control rods	Four semaphore-type, aluminum-clad cadmium shim-safety blades; one aluminum regulating rod
Shielding	Lead and steel thermal shield; concrete biological shield
Vessel Size	6.0 ft (1.83 m) in diameter by 10 ft 4 in. (3.15 m) high
Vessel Volume	1.1×10^3 gal (4.2×10^3 L)
Reactor Building	
Construction	Steel shell with concrete liner
Volume	2.6×10^5 ft^3 (7.4×10^3 m^3)
Leak Rate	54.2 x ft^3/h (4.3×10^4 m^3/s)
Stack Height	76 ft (23.2 m)
Stack Exhaust Rate	4.0×10^3 ft^3/m (1.9 m^3/s)
Location	Georgia Institute of Technology Campus, Atlanta, Georgia



(U) Fig. 3. GTRR vertical cross section.



(U) Fig. 4. GTRR horizontal cross section.

(U) 3.2.1. Fuel Elements. Each element is a rectangular array of 16 curved fuel plates approximately 2-3/4 by 3 by 36 in. long (7 by 7.5 by 220 cm long) with a 25-in. (63.5-cm) fueled section. Each element contains 6.63 oz (188 g) of 93 enriched ^{235}U. At the normal beginning of the cycle, core loading is 6.6 oz (3.0 kg) ^{235}U; average burnup is 29.5 . Two elements are replaced every 90 MW-days, which would be ~18 days at full power, followed by two shifts for refueling. This is the fuel cycle assumed by this report, even though the GTRR only operates a small fraction of the time.

(U) 3.2.2. Reactivity Control System. The reactor is controlled by four shim-safety semaphore-type blades and one aluminum regulating rod. The blades are similar in design and operation to those used at the NBSR.

(U) As in the NBSR, the GTRR can be shut down by dumping the D_2O top reflector to ~1 in. (~2.5 cm) above core level, which adds sufficient negative reactivity to overcome the unlikely event that the shim-safety rods fail to insert.

(U) 3.2.3. Reactor Vessel Internal Arrangement. The reactor core is contained in a 10-ft 4-in. (3.15 m)-high, 6-ft(1.83-m)-diam aluminum tank. There are several experimental irradiation facilities, including thermal columns, beam tubes, experimental thimbles, and pneumatic facilities, located around the periphery of the core. The core permits only up to 19 fuel assemblies to be arranged in a hexagonal configuration.

(U) 3.2.4. Emergency Core Cooling System. The GTRR ECCS is a 300-gal (1135-L) D_2O tank located in the reactor room above the reactor tank. It provides an automatic introduction of coolant to the reactor tank upon the loss of reactor coolant or the loss of electrical power. It can supply emergency coolant for approximately 30 min.

(U) 3.3 University of Missouri Research Reactor

(U) The MURR is an enclosed-pool-type, beryllium- and graphite-moderated, light-water-cooled, flux-trap reactor with a licensed power level of 10 MW. The design includes a pressurized tank located within a pool that contains the fueled region of the reactor. The core is an annular right circular cylinder that is 5.5 in. (14 cm) in inside diameter, 11.6 in. (29.5 cm) in outside diameter, and 24 in. (61 cm) in length; the core contains eight fuel elements arranged vertically around the annulus. The primary loop operates at 70 psig (584 kPa) and is designed for reactor operation at 10 MW. An annular gap outside the pressure vessel contains the control elements and is surrounded by a

beryllium reflector assembly. An additional outer reflector composed of graphite canned in aluminum surrounds the beryllium reflector assembly.

(U) The reactor coolant piping enters the pool through the bottom and connects to the top of the vessel above the core region. The vessel discharge piping connects below the core region and is routed above the core through an invert loop before exiting at the bottom of the pool. An anti-siphon system connects with the piping at the top of the invert loop, and two siphon break valves are arranged in parallel on the anti-siphon line. This arrangement provides a captive volume to prevent uncovering the core after a primary loop rupture outside the pool.

(U) The heat from the core is transferred to the secondary loop through a pair of heat exchangers arranged in parallel. The secondary coolant system discharges the heat to a cooling tower.

(U) Isolation valves located in the inlet and the outlet of the reactor coolant lines in the pipe trench below the reactor operating floor can be used to isolate the reactor during significant decreases in pressure or flow and to activate the reactor in-pool convective cooling loop. The reactor convective cooling loop consists of a valve, an in-pool heat exchanger, and the necessary piping to connect to the primary loop. This system serves to remove core decay heat through the in-pool cooling system following a loss of normal coolant flow. The pool is cooled by a separate system that removes heat from the reflector and the bulk pool water.

(U) Table III lists the MURR facility parameters used[5,8], and Figs. 5 and 6 show the vessel assembly.

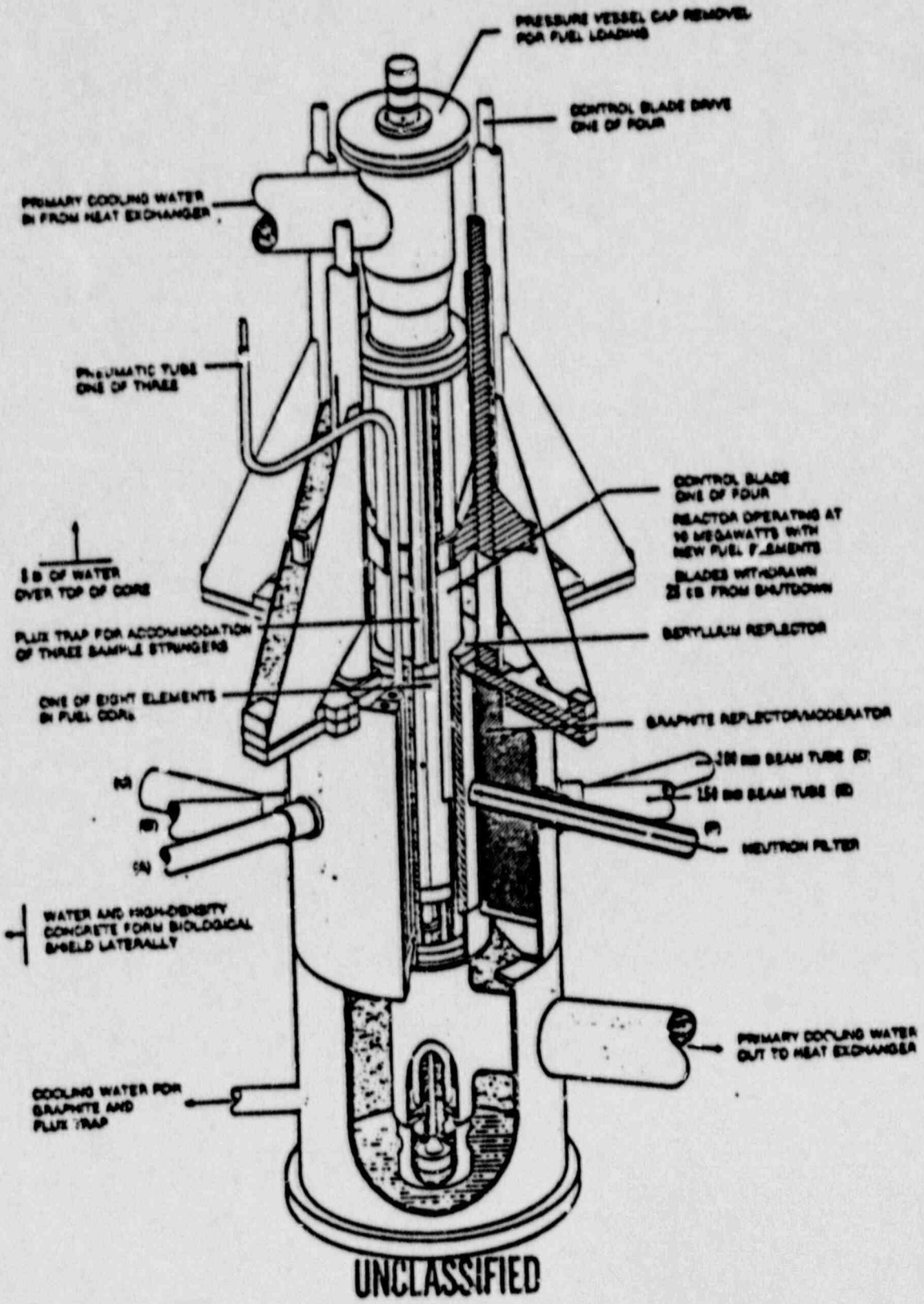
(U) 3.3.1. Fuel Elements. Each of the MURR pie-shaped fuel elements has an overall length of 32.5 in. (82.6 cm), contains a 24-in. (61-cm) active region, has a 3.28-in. (8.33-cm) radial dimension, and has 24 curved fuel-bearing plates of progressive width. There are eight assemblies in the MURR core; thus, each element spans a 45° segment of the annulus. The fuel is a uranium-aluminum mixture containing 27.5 oz (780 g) of ^{235}U (93.15% enriched) per fuel element. When fully loaded, the reactor contains 12.3 lb (~5.6 kg) of ^{235}U. Average fuel burnup is ~29%. The equivalent of one element is depleted every 3 weeks.

(U) 3.3.2. Reactivity Control System. The reactor is controlled by four shim-safety blades and one stainless-steel regulating rod. The shim-safety blades are constructed of boral containing a boron carbide and aluminum neutron-absorbing mixture. The regulating blade is stainless steel. The control and

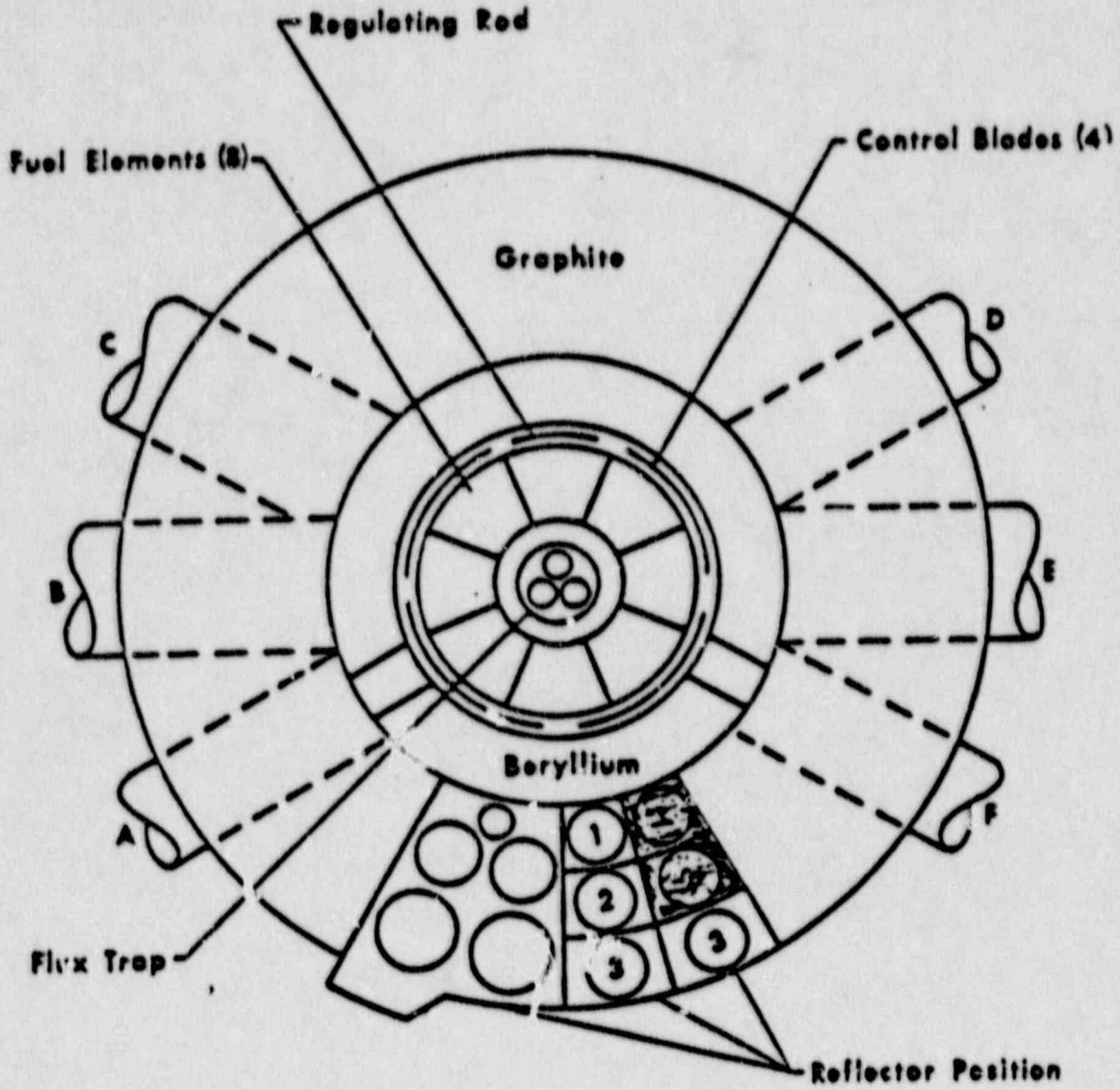
TABLE III

(U) MURR FACILITY PARAMETERS

Power level	10 MW
(C) [redacted]	
Moderator and coolant	Light water
Reflector	Beryllium and graphite
Coolant flow	3600 gal/min (227 L/s) pressurized loop; 800--1200 gal/min (57--76 L/s) pool
Fuel type	Pie-shaped elements using curved plates of progressive widths
Enrichment	93% ^{235}U
Number of elements	8
Arrangement	Cylindrical annulus
Control rods	Four curved-plate, aluminum-clad boral safety rods; one curved-plate, stainless-steel regulating rod.
Shielding	Light-water thermal shield; concrete biological shield
Vessel size	12 in. (32 cm) in diameter with a 5-in. (12.7-cm)-diam axially concentric flux trap
Pool Volume	1.7 x 10^4 gal (6.4 x 10^4 L)
Reactor Building	Reinforced concrete
Construction Volume	2.4 x 10^5 ft^3 (6.7 x 10^3 m^3)
Leak Rate	915 ft^3/h (7.2 x 10^{-3} m^3/s)
Stack Height	57 ft (17.4 m)
Stack Exhaust Rate	1.5 x 10^3 ft^3/m (0.8 m^3/s)
Location	University of Missouri Campus, Research Park, Columbia, Missouri



(U) Fig. 5. MURR vertical section.



(U) Fig. 6. MURR horizontal cross section.

regulating rods operate in the gap between the reactor pressure vessel and the beryllium reflector. The control blade drive mechanisms are on the bridge structure directly over the pool surface. The shim-safety drives are electrically driven ball-screw mechanisms working through an electromagnetic clutch. A reactor scram signal deenergizes the clutch, allowing the safety blades to fall by gravity. The regulating rod uses a reversible two-phase electric drive motor and has no safety shutdown function. An automatic shim control circuit adjusts the regulating rod and assists it with the compensation for various cyclic poison effects. When the regulating rod reaches the 20% withdrawn position, the automatic shim control circuit will insert the shim-safety rods until the regulating rod reaches its 60% withdrawn position, allowing the regulating rod to continue compensating for reactivity changes.

(U) 3.3.3. Reactor Vessel Internal Arrangement. The core is contained in a pressure vessel having a 12-in. (32-cm) inside diameter. A flux trap with a 5-in. (12.7-cm) diameter penetrates the center of the core axially symmetric with the vessel, forming an annular core region. All reactor components other than the fuel (that is, control rods, reflectors, and irradiation facilities) lie outside the vessel in the pool.

(U) 3.3.4. Emergency Core Cooling System. Two main systems ensure core cooling during any single-failure accident condition. One is the anti-siphon system; the other is the in-pool convective loop. The anti-siphon system prevents uncovering of the core after a coolant loop rupture outside of the pool. The anti-siphon line is connected to the top of the core coolant discharge line invert loop, providing a siphon break. Two parallel redundant valves in this line open on a loss of primary loop pressure. Loop isolation valves located just outside of the biological shield in the pipe chase also close on a loss of primary pressure to assist in mitigating a loss-of-coolant situation.

(U) The in-pool convective loop is a heat exchanger located above the core that provides an emergency heat sink on loss of flow through the primary loop. During normal operation, the discharge line of the heat exchanger is closed by a redundant pair of valves, which open on a loss of flow. The top of the heat exchanger is vented to a holdup tank to prevent the natural convection flow from air-locking.

(U) 3.4. Massachusetts Institute of Technology Reactor

(U) The MITR is a highly enriched, tank-type, light-water-moderated-and-cooled, heavy-water-and-graphite-reflected research reactor consisting of 26 rhomboid elements in a close-packed hexagonal array. Six control blades are arranged around the core, one on each face of the hexagon.

(U) The reactor is contained in two tanks. The inner tank, which closely surrounds the reactor core, contains the light-water coolant/moderator. This water is pumped down between the inner tank and the core and returns up through the core, providing cooling. The outer tank contains heavy water that serves as a neutron reflector.

(U) Heat from the primary system is transferred through heat exchangers to a secondary system with a cooling tower. Table IV lists the MITR facility parameters used[5,9], and Figs. 7 and 8 show the arrangement of the two vessels.

(U) 3.4.1. Fuel Elements. Each of the 26 fuel elements is a rhomboid array of 15 flat plates with a perpendicular distance across them of 2.375 in. (6.03 cm). The vertex angles of the rhombus are 60° and 120° so that the elements can be close-packed in a hexagonal array. The elements are 26.25 in. (66.68 cm) long and have a fueled section approximately 22.5 in. (57.2 cm) long. The elements are symmetric on three axes and therefore can be rotated 180° or flipped during refueling to maximize burnup. Each element initially contains 1.1 lb (506 g) of ^{235}U, and average burnup exceeds 40%.

(U) 3.4.2. Reactivity Control System. The reactor is controlled by six shim blades and one regulating rod. One shim blade is positioned against each face of the hexagonal core array, and the regulating rod is located at one apex of the hexagon.

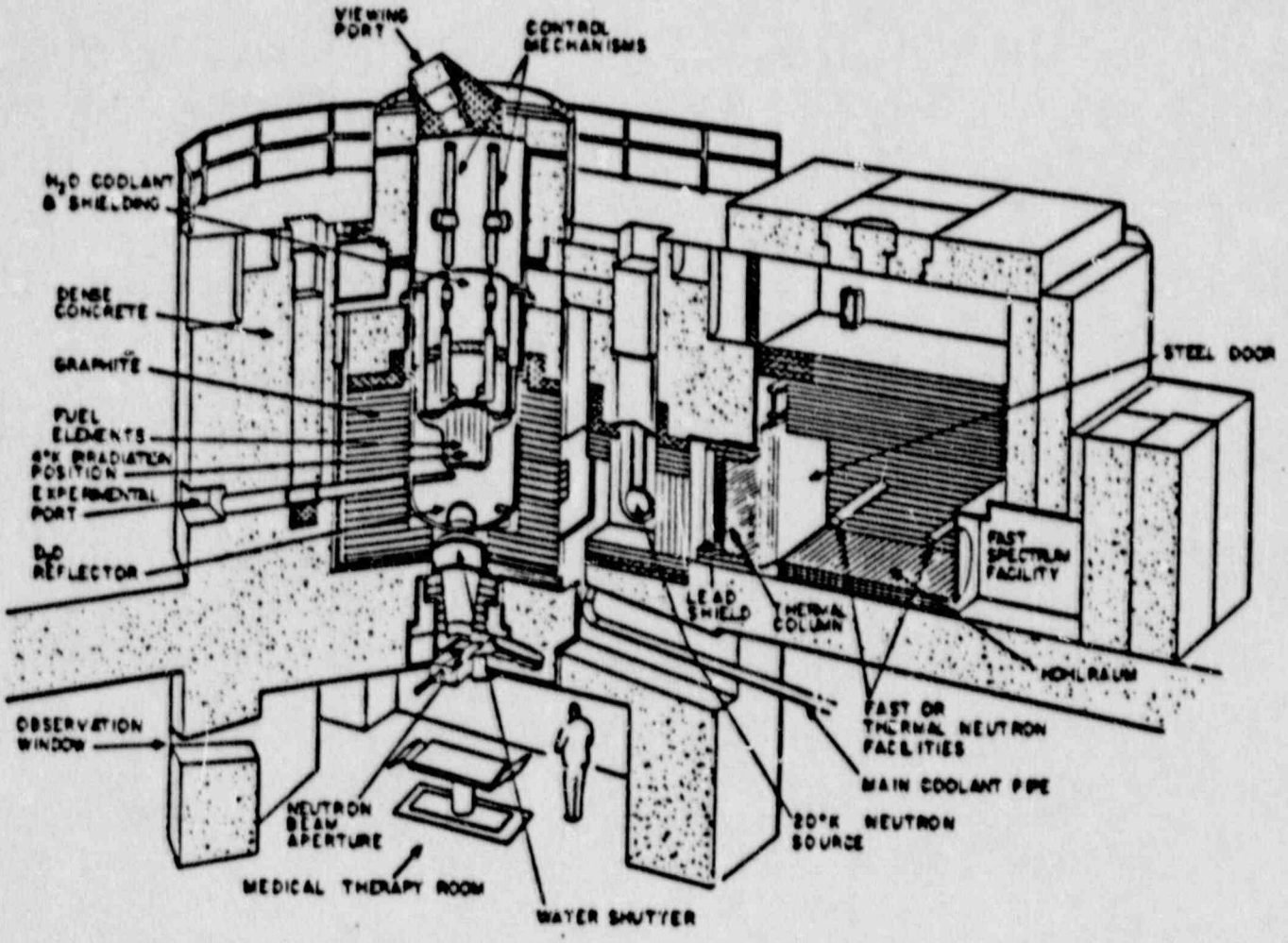
(U) Each shim blade consists of two 0.125-in. (0.318-cm)-thick sheets of 1.1% boron stainless steel 30.625 in. (77.79 cm) long by 7.0 in (17.8 cm) wide. The sheets are separated by spacers that allow a 0.050-in. (0.127-cm) gap between them for cooling and protection from warpage during irradiation. The blades are coupled to the drive through an electromagnet to allow scramming.

(U) The regulating rod is cylindrical with an absorber section 24.875 in. (63.18 cm) long and 0.875 in. (2.22 cm) in diameter. It has an aluminum core, a 0.040-in. (0.10-cm)-thick cadmium layer, and an outer aluminum cladding. It is attached directly to the drive mechanism (without an electromagnet), and it has no scram function.

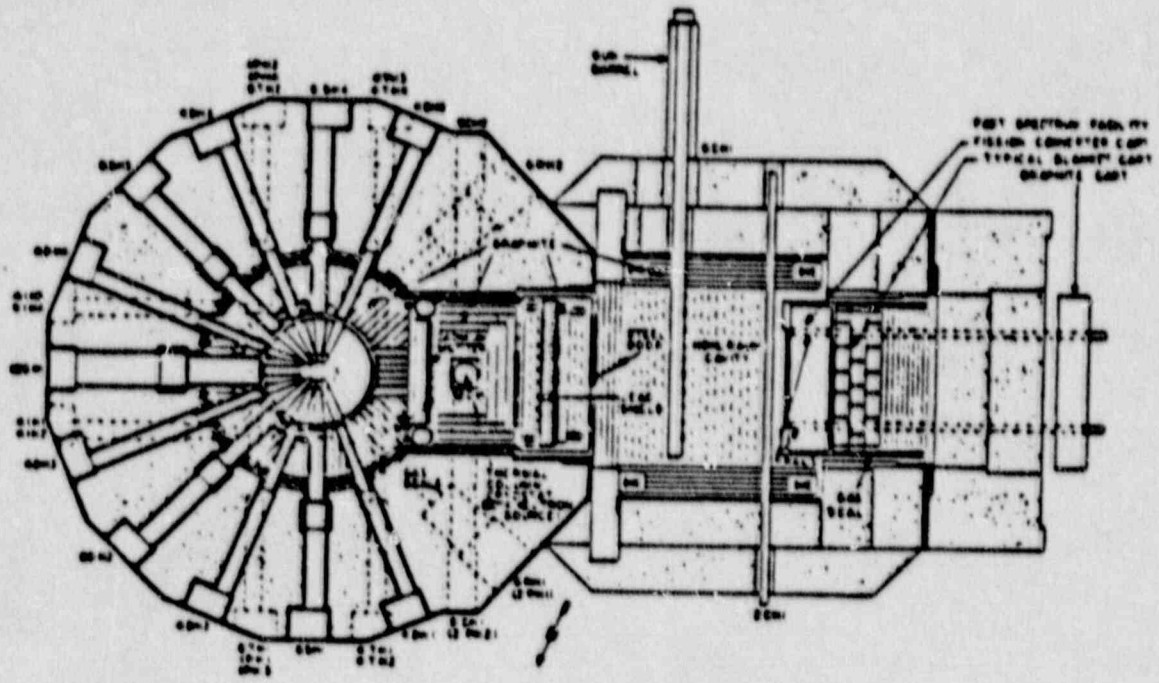
TABLE IV

(U) MITR FACILITY PARAMETERS

Power Level	5 MW
(C)	
Moderator and Coolant	Light water
Reflector	Heavy water and graphite
Coolant Flow	2000 gal/min (126 L/s)
Fuel Type	MTR plate-type in rhomboid array
Enrichment	93% ^{235}U
Number of Elements	26
Arrangement	Close-packed hexagonal array
Control Rods	Six flat shim blades, one round regulating rod
Vessel Size	
Core Tank	12.75 ft (3.80 m) high with an upper diameter of 45 in. (1.14 m) and a lower diameter of 20 in. (0.51 m)
Reflector Tank	10 ft (3.1 m) high with a diameter of 4 ft (1.2 m)
Core Tank Volume	800 gal (3×10^3 L)
Shielding	Lead and steel thermal shield; concrete biological shield
Reactor Building	
Construction	Steel shell with concrete liner
Volume	2.0×10^5 ft^3 (5.7×10^3 m^3)
Leak Rate	161 ft^3/h (1.3×10^{-3} m^3/s)
Stack Height	150 ft (46 m)
Stack Exhaust Flow Rate	4×10^3 ft^3/m (1.9 m^3/s)
Location	MIT campus, Cambridge, Massachusetts



(U) Fig. 7. Vertical section of MITR.



(U) Fig. 8. Horizontal cross section of MITR

(U) 3.4.3. Reactor Vessel Internal Arrangement. The core is contained in a complex-shaped inner tank that changes in diameter at approximately the top of the active fuel from an inside diameter of 20 in. (50.8 cm) surrounding the core to a diameter of approximately 45 in. (114 cm) above the core. The overall height is approximately 8.5 ft (2.6 m). This tank separates the light-water coolant and moderator from the heavy water reflector. The core shroud and core support housing assembly are inside this tank and separate the core inlet flow from the core outlet flow.

(U) The coolant enters the inner tank near the top through an 8-in. (20-cm) pipe and is directed down between the shroud and the inner tank to a plenum area under the core. The flow then is forced up through the core to the outlet pipe that terminates inside the shroud.

(U) The heavy-water reflector that surrounds the inner tank is contained in an outer vessel approximately 4 ft (1.2 m) in diameter and 10 ft (3.1 m) high. The 14 reentrant beam tubes penetrate the lower portion of this tank and are positioned to take advantage of the thermal neutron flux peak in the reflector beneath the core itself.

(U) 3.4.4. Emergency Core Cooling System. There are two basic methods of emergency cooling: internal recirculation and the spray system. There is a captive volume of ~500 gal (1750 L) in the inner vessel, which is sufficient to remove decay heat, precluding fuel melt.

(U) Two sets of ball check valves enhance the cooling effect of this captive volume of water. The first set of valves is the anti-siphon valves. These two valves penetrate the top of the shroud, preventing a pipe rupture outside the reactor tank from siphoning the captive volume of fluid out of the vessel. During normal operation, these valves are held shut by the differential pressure across the core caused by the primary coolant flow; upon loss of flow, the ball drops, opening a pathway through the top of the shroud, precluding any siphoning effect. The second set of valves (four each) penetrate the upper core support structure and provide a natural circulation return path from inside the shroud just above the core back to the downcomer region outside the core. As was the case with the anti-siphon valves, these valves are held closed by the core differential pressure during normal operation and open by gravity on loss of flow.

(U) If, through some vessel leakage, the water in the vessel is lost, two core spray lines can be supplied from either the City of Cambridge water system or from a recycle system that itself has multiple supplies.

(U) 3.5 Rhode Island Nuclear Science Center Reactor

(U) The Rhode Island NSCR is a highly enriched pool-type light-water reactor consisting of 30 square elements in a rectangular array positioned in an aluminum frame. Primary coolant piping attached to this frame provides cooling for the core. The piping penetrates the pool and transports the heat generated by the core to heat exchangers, where the heat is transferred to the secondary cooling system. This system transports the heat to cooling towers outside the building. Table V shows NSCR facility parameters used[5,10]; Fig. 9 shows the facility arrangement.

(U) 3.5.1. Fuel Elements. Each of the 30 fuel elements is a square array consisting of 18 plates 3 in. (1.62 cm) wide, 24 in. (61 cm) long, and 0.060 in. (0.15 cm) thick. The fuel is a 93% enriched uranium-aluminum alloy containing 4.4 oz (124 g) of ^{235}U per element. The elements are left in the core until the ^{235}U content drops to approximately 3.8 oz (108 g).

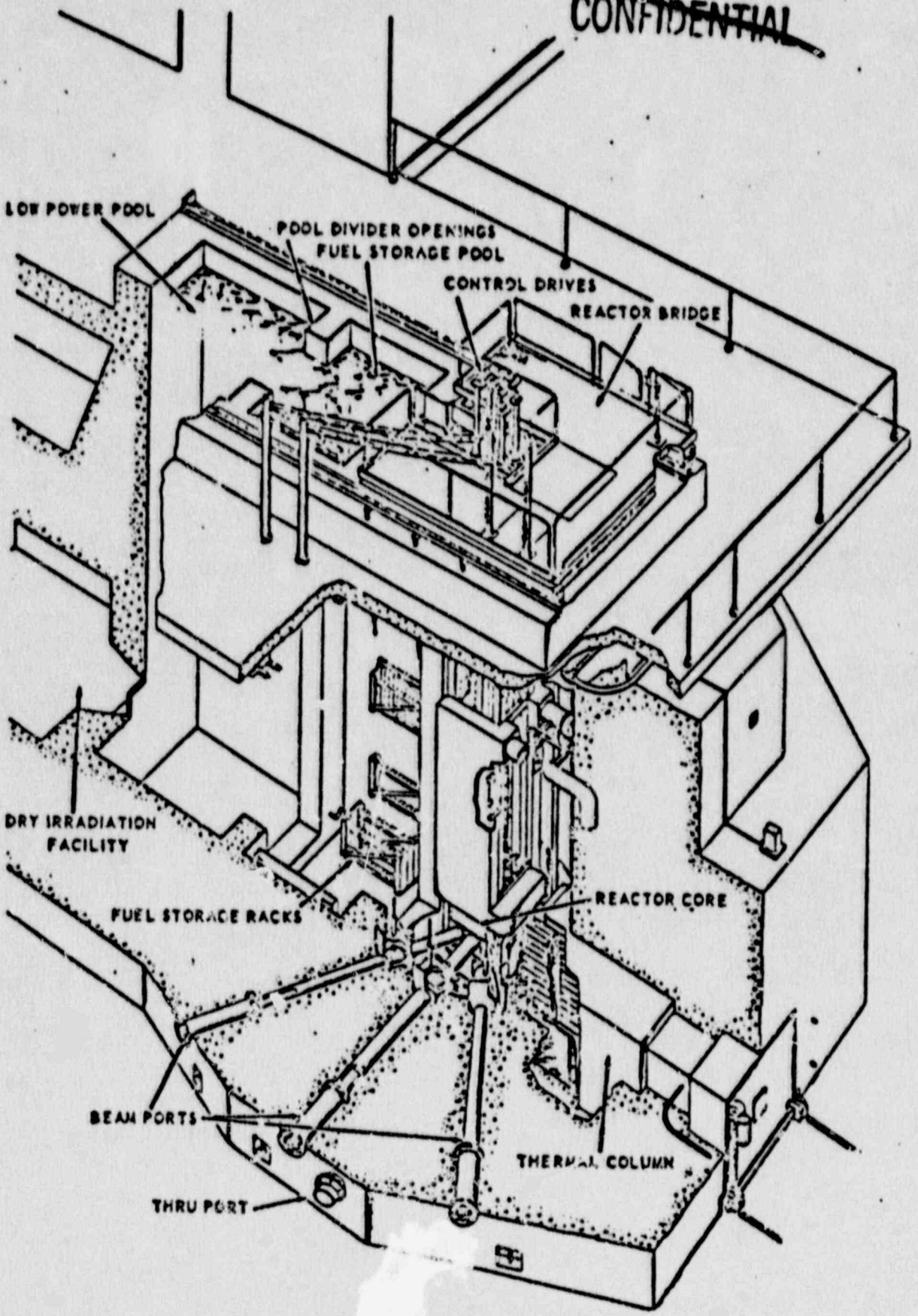
(U) 3.5.2. Reactivity Control System. The reactor is controlled by four control blades and one servo element. All are constructed from a mixture of B_4C and aluminum (Boral) and clad with aluminum. The control blades are 10.6 in. (26.9 cm) wide, 0.38 in. (0.97 cm) thick, and 54.1 in. (137.4 cm) long, with an active length of 52.1 in. (132.3 cm). All are driven by linear electromechanical screw drives. However, the control blades are coupled to the drives through electomagnets, which provide for scram action, but the servo element (having no scram action) is coupled directly.

(U) 3.5.3. Reactor Pool Arrangement. The reactor is suspended from a bridge near the bottom of a pool that is ~28 ft (8.5 m) deep and contains 36 300 gal (137 200 L) of high purity light water. The horizontal cross-section of the pool is roughly rectangular and consists of three sections. The first section is the stall area, which contains the primary coolant piping and flow baffles that provide forced cooling to the core. This area also contains one tangential and six radial beam ports. The center section of pool contains the fuel racks for storing spent fuel, and the third section is an open pool area that is separated from the dry irradiation facility by a thin aluminum window that is an integral portion of the pool liner. In this region, the core can be operated at low power (<100 kWth) with natural circulation so that it can be used as an irradiation source for experiments in the dry irradiation facility.

TABLE V

(U) RHODE ISLAND NSCR FACILITY PARAMETERS

Power Level	2 MW
(C)	
Moderator and Coolant	Light water
Reflector	Graphite
Coolant Flow	1500 gal/min (95 L/s)
Fuel Type	MTR plate-type
Enrichment	93% ^{235}U
Number of Elements	30
Arrangement	7 by 9 rectangular array; outer ring and five inner locations unfueled
Control Rods	Four flat control blades; one square regulating rod
Shielding	Pool water and concrete
Pool Volume	36 300 gal (137 200 L)
Reactor Building	
Construction	Reinforced concrete
Volume	2.1×10^5 ft^3 (5.9×10^3 m^3)
Leak Rate	2.6×10^4 ft^3/h (2.0×10^{-1} m^3/s)
Stack Height	115 ft (35 m)
Stack Exhaust Flow Rate	4.0×10^3 ft^3/m (1.9 m^3/s)
Location	Narragansett Research Center Narragansett, Rhode Island



(U) Fig. 9. Sectional view RI NSCR.

(U) 3.5.4. Emergency Core Cooling System. Emergency cooling is provided by the large inventory of water in the pool. However, if this water were to drain, sufficient water is retained in the aluminum box surrounding the core for enough time to prevent core melt.

(U) 3.6 Buffalo Materials Research Center Reactor

(U) The BMRC Reactor is a fixed-core, pool-type research reactor using light water as the moderator and coolant and using solid pin-type fuel assemblies. The reactor core is supported on a plenum structure that rests on the pool floor. Primary coolant piping attached to the plenum provides cooling for the core. This primary piping penetrates the pool wall and transports the heat generated by the core to a heat exchanger, where it is transferred to the secondary cooling system. This secondary system transports the heat to a cooling tower outside the building, where the heat is dumped to the environment. Table VI shows the BMRC Reactor facility parameters used;[5,11] Figs. 10 and 11 show the facility arrangements.

(U) 3.6.1. Fuel Elements. The BMRC reactor uses a type of fuel commonly referred to as PULSTAR. It consists of sintered UO_2 pellets in a pin geometry, which is similar to current light-water power reactor fuels. Each assembly consists of 25 fuel-bearing pins. The pins are thin-walled 0.02-in. (0.051-mm) Zircaloy-2 tubes filled with sintered UO_2 pellets with welded Zircaloy-2 end plugs. The uranium is enriched to 6% in the isotope ^{235}U. The UO_2 pellets are about 0.42 in. (1.07 cm) in diameter and about 0.60 in. (1.52 cm) long. A finished pin is 0.47 in. (1.19 cm) in diameter and 26 in. (66.0 cm) long with spacers brazed around the circumference (90° apart) of the pin near the ends and at the center.

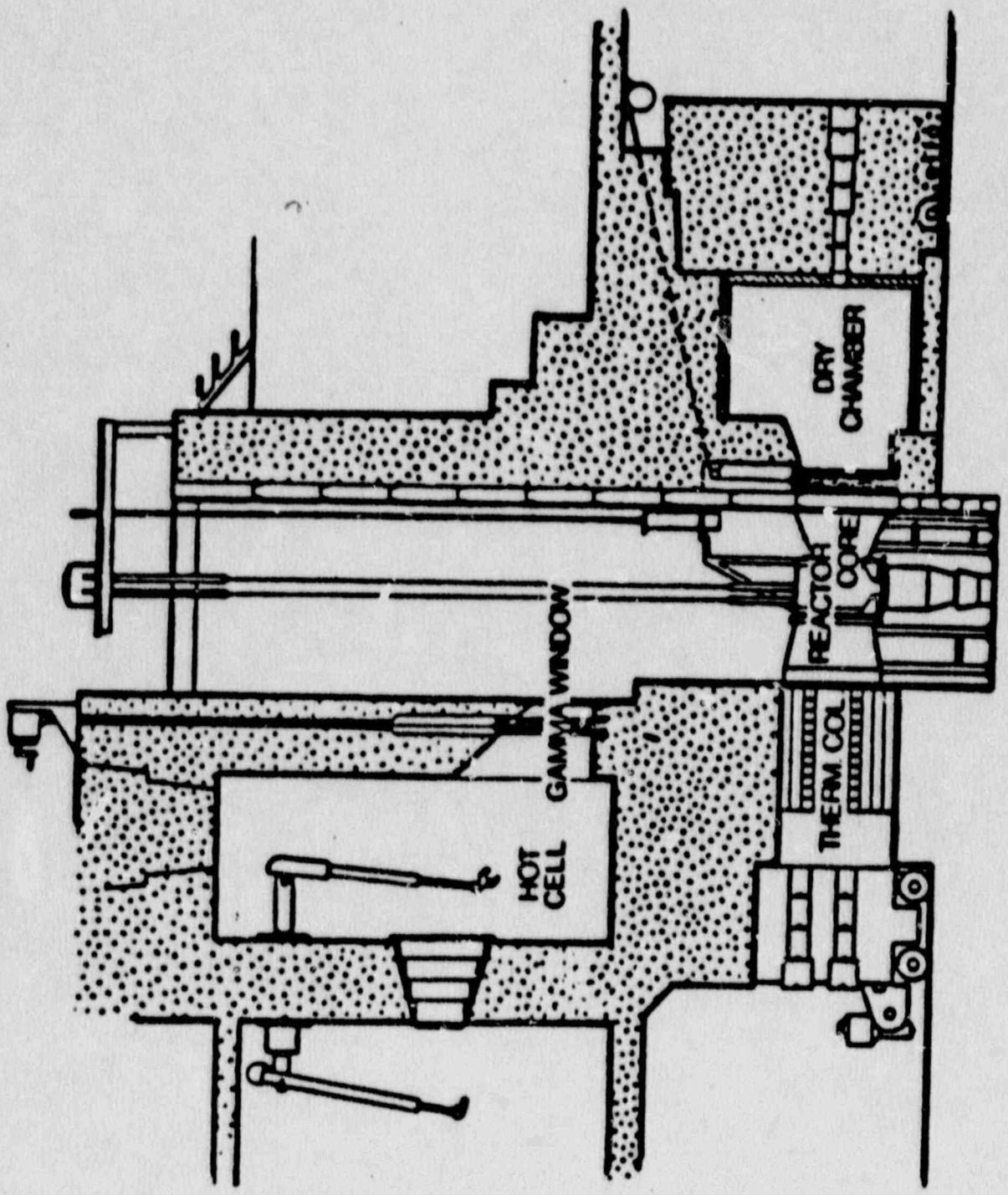
(U) Approximately 1.1 oz (30.7 g) ^{235}U [1.1 lb (513 g U)] is contained in each pin, making a total of 1.7 lb (768 g) ^{235}U [28.2 lb (12.8 kg U)] per assembly. The pins are fastened mechanically in groups of 25 with aluminum end fittings and are contained in a Zircaloy-2 box. A guide tube (nosepiece) machined to fit the grid plate is attached to the lower end of the fuel pin assembly. A horizontal rod is fastened between two sides of the box near the upper end of the fuel assembly and serves as a handle for insertion or removal of the assembly from the grid plate. The finished assembly is about 38 in. (96.5 cm) long with a cross section of about 2.7 in. by 3.2 in. (6.86 cm by 8.13 cm). The

TABLE VI

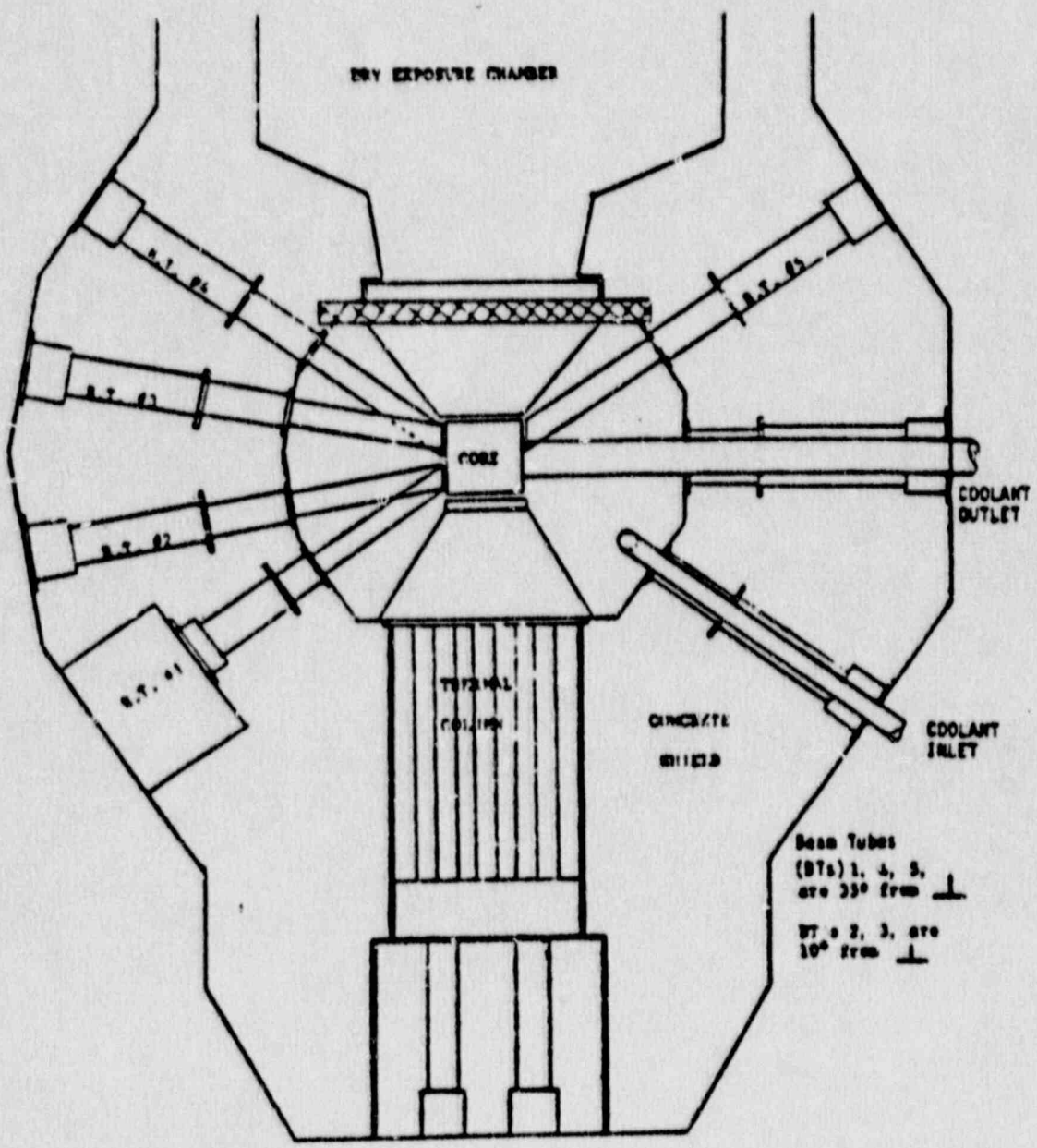
(U) BMRC REACTOR FACILITY PARAMETERS

(C)

Power Level	2 MW
Moderator and Coolant	Light water
Reflector	Graphite
Coolant Flow	1200 gal/min (76 L/s)
Fuel Type	PULSTAR 25 pins/element
Enrichment	6% ^{235}U
Number of Elements	22--30
Arrangement	6 by 6 array; one row usually filled with graphite
Control Rods	5 control-safety blades; 1 regulating blade
Shielding	Pool water and concrete
Pool Volume	14 600 gal (55 300 L)
Reactor Building	
Construction	Reinforced concrete
Volume	1.8×10^5 ft^3 (5.3×10^3 m^3)
Leak Rate	<420 ft^3/h (3.3×10^{-3} m^3/s)
Stack Height (Power Plant Stack)	167 ft (50.9 m)
Stack Exhaust Flow Rate	5.0×10^3 ft^3/m (2.36 m^3/s)
Location	State University of New York at Buffalo Campus, Buffalo, New York



(U) Fig. 10. Vertical cross section of BMRC Reactor.



(U) Fig. 11. Horizontal cross section of BMRC reactor.

nosepiece is inserted in a large hole in the grid plate that supports the entire fuel array. Both ends of the assembly are open so that cooling water can flow up or down around the fuel pins.

(U) 3.6.2. Reactivity Control System. The BMRC reactor is controlled by five control-safety blades and one regulating blade. All are made of a nickel-plated silver/indium/cadmium alloy. These blades are 0.18 in. (0.46 cm) thick by 5 in. (12.7 cm) wide and 24 in. (61 cm) long. All are driven by a linear electromechanical screw driver. The control-safety blades are coupled to the drives through electromagnets, which allows for scram action, but the regulating blade, having no scram function, is coupled directly.

(U) 3.6.3. Reactor Pool Arrangement. The reactor core is located near the bottom of a water-filled, aluminum-lined, reinforced-concrete pool that is roughly hexagonal in cross section and 14 ft (4.27 m) wide at the top. This pool contains ~ 14 600 gal (55 300 L) of high purity light water, and the top of the reactor core is 20 ft (6.1 m) below the water surface. The pool has an aluminum liner that is 0.25 in. (6.35 mm) thick. A sealant is used to prevent corrosion of the liner that might otherwise occur because of contact with the concrete shield walls. This liner is penetrated as follows.

(U) (1) Five 6-in. (15.2-cm)-round beam-tube ports radiate from the core around the lower tank section.

(U) (2) One pneumatic conveyor system enters near the top of the upper tank and terminates in the reflector region above the beam tubes.

(U) (3) The primary coolant exits the tank through a penetration that formerly housed a 12-in. (30.5-cm)-square beam port.

(U) (4) The primary coolant returns to the pool through a penetration that formerly housed a 6-in. (15.2-cm)-round beam port.

(U) (5) A pass-through canal (tube) provides a passage between the upper portion of the tank and the hot cell.

(U) (5) Eight emergency pool fill nozzles are located in the lower tank section just below the step.

(U) Storage cylinders for fuel assemblies are arranged around the upper section of the lower tank on all faces except the back wall, where a dry chamber nosepiece is located. In addition, a rack for eight used elements is located on the tank wall common with the hot cell to provide an experimental gamma irradiation facility.

(U) The grid plate is a 5-in. (12.7-cm)-thick aluminum plate bolted to a plenum chamber. Thirty-six holes capable of accommodating the nosepieces of the fuel assemblies are arranged in a 6-by-6 pattern on the plate. Small holes are located between the nosepiece holes to provide additional passages for water flow past the sides of the fuel assemblies. Holes not required for fuel assemblies or in-core experiments are plugged to confine the coolant flow to core assemblies and experiment positions. Small pins set in the grid plate mate with holes in the nosepiece shoulders to position the assemblies axially.

(U) The plenum chamber, which channels the coolant flow to the discharge pipe during forced convection cooling, is supported by four legs. The aluminum superstructure above the core provides a support for the neutron detection chambers.

(U) 3.6.4. Emergency Core Cooling System. Emergency cooling is provided by the large inventory of water in the pool. However, if this water were to drain instantaneously, the licensee has calculated a maximum fuel pin temperature that is >1832°F (1000°C) below the melting point of the Zircaloy-2 cladding. An emergency pool-filling system is available for adding city water directly to the pool through headers arranged to spray the core. The system is operated manually by opening a normally closed solenoid valve installed in a line between the city water system and the pool.

(U) 3.7 CINTICHEM, Inc. Reactor

(U) The CINTICHEM nuclear reactor is a pool-type research reactor licensed to operate at thermal power levels up to and including 5 MW. The reactor is a light-water-moderated, -cooled, and -shielded, water-and-graphite-reflected, solid-fuel reactor. It is typical of a number of NRC-licensed reactors based on the Oak Ridge National Laboratory Bulk Shielding Reactor (BSR) design and has been in operation since 1961. Its principal use is for the production of radiochemicals and radiopharmaceuticals for use in medical therapy, research, and commerce.

(U) The primary cooling system consists of demineralized water plus heat exchangers and pumps. Heat generated in the pool water by the reactor is transferred to the heat exchangers, where it subsequently is removed by the secondary cooling system. The secondary cooling system in turn transfers its heat to the atmosphere through a cooling tower. During forced cooling, pool water flows down in the pool or stall (depending on the core position used), through the

reactor core grid plate and plenum at a flow rate of ~2200 gal/min (140 L/s), and then to a holdup tank. The water then is drawn from the holdup tank by a main circulating pump and pumped through the shell sides of the two stainless-steel heat exchangers in series and back into the pool.

(U) Table VII lists the CINTICHEM reactor facility parameters used;[5,12] Figs. 12 and 13 show the facility arrangements.

(U) 3.7.1. Fuel Elements. Each standard fuel element assembly is composed of four major components: the unfueled aluminum side plates, the fuel plates, the lower end fitting, and the fuel handle. The two side plates keep the fuel plates in an approximately 3- by 3-in. (7.6- by 7.6-cm) assembly. A horizontal rod fastened between the side plates near the upper end of the fuel assembly serves as a handle for the insertion or withdrawal of the assembly from the grid plate. A standard fuel element assembly has 16 fueled plates. The fuel plates are 24-5/8 in. (9.7 cm) long and made of enriched (93%) uranium-aluminum alloy fuel "meat" sandwiched between high-purity aluminum cladding. Each fuel plate is formed into a convex shape to minimize thermal stress and is fastened to the side plates by swaging. An end fitting that is machined to fit into the grid plate is attached to the lower end of the fuel plate assembly. The nominal fuel content of the element is 6.9 oz (196 g) ^{235}U.

(U) Six of the elements composing the reactor core are special control-rod fuel assemblies. These assemblies contain 3.4 oz (98 g) of ^{235}U in nine fueled plates. These control rod-fuel assemblies contain a centrally located slot into which the reactor control rods are inserted. Assembled at the top of each of these elements is a shock absorber that cushions the fall of the control rods when they are dropped.

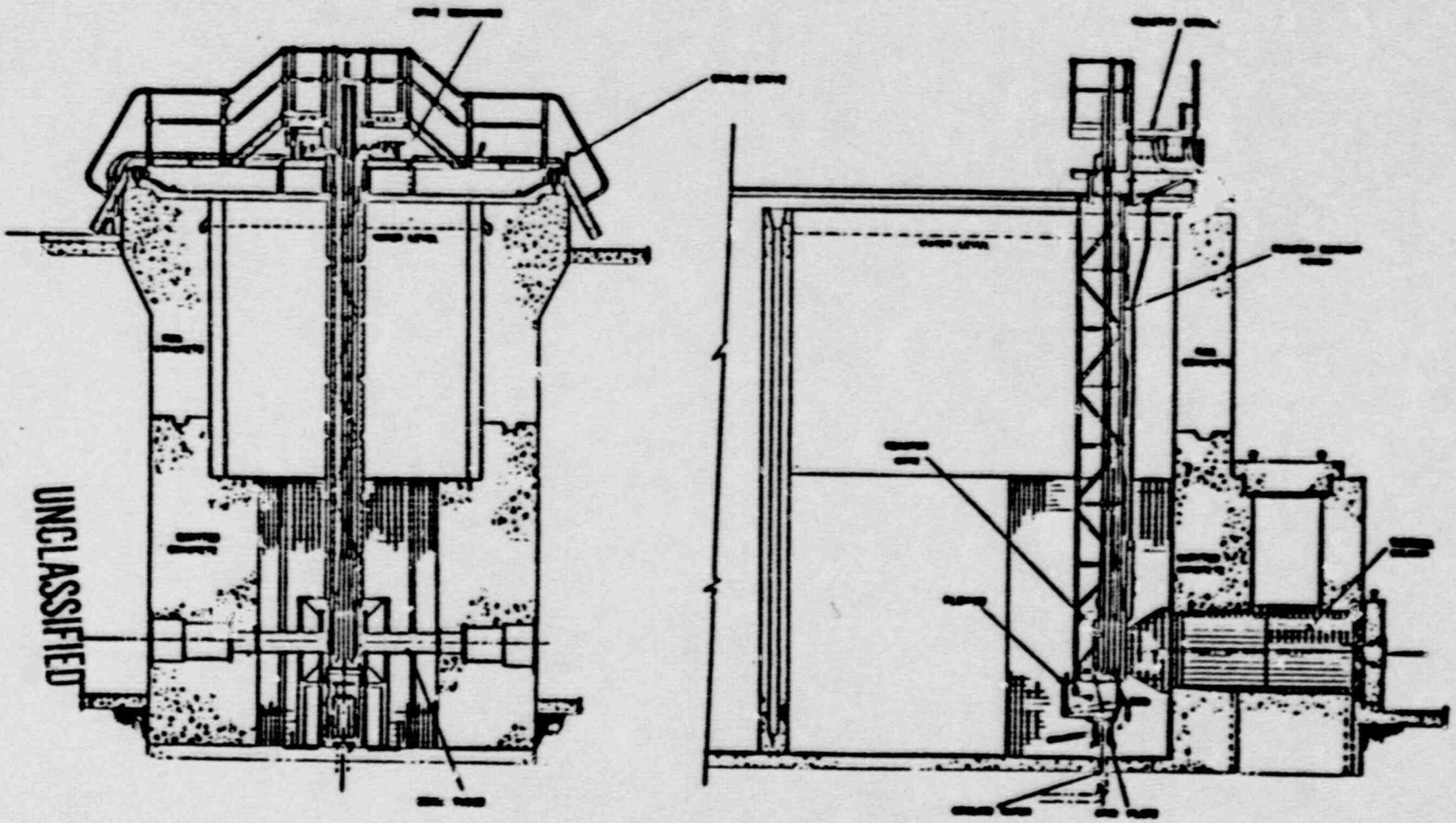
(U) 3.7.2. Reactivity Control System. The reactor control system is typical of those used for pool-type research reactors. The reactor is controlled by five thermal neutron-absorbing silver/indium/cadmium control rods and one stainless-steel regulating rod. These are 0.85 in. (2.16 cm) thick, 2.23 in. (5.66 cm) wide, and 24.5 in. (62.23 cm) long. The control rods provide coarse adjustment of the neutron flux level, and the regulating rod provides fine adjustment. The five control rods can be operated manually and can be scrammed automatically. The regulating rod may be operated manually or automatically in response to power level demand settings.

(U) Each control rod is coupled to the drive mechanism shaft by electro-magnets. Scramming or quick insertion is accomplished by de-energizing the

TABLE VII

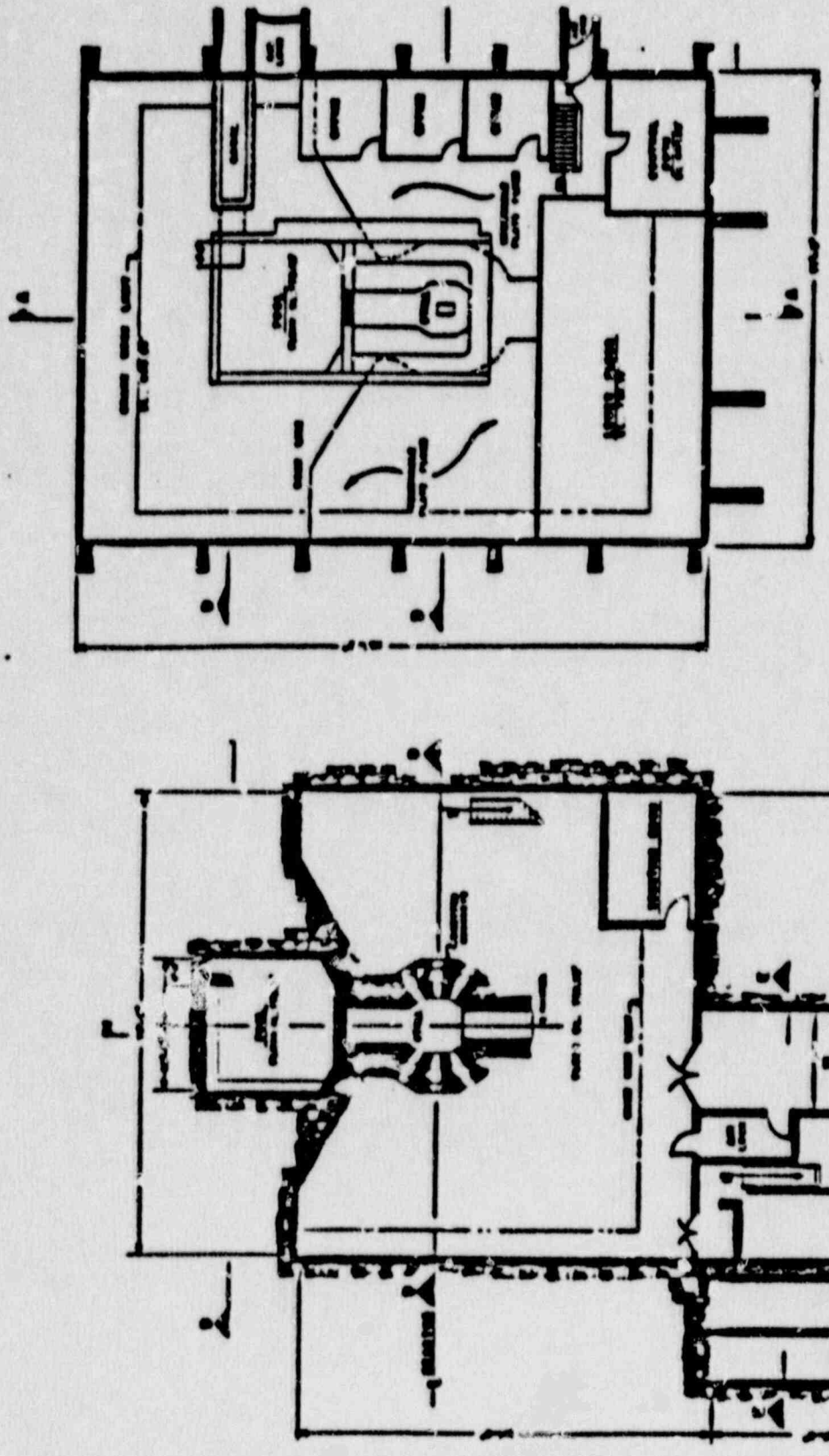
(U) CINTICHEM REACTOR FACILITY PARAMETERS

Power Level	5 MW
(C)	
Moderator and Coolant	Light water
Reflector	Water and graphite
Coolant Flow	2200 gal/min (140 L/s)
Fuel Type	MTR plate
Enrichment	93% ^{235}U
Number of Elements	31 standard, 6 control
Arrangement	6 by 9 rectangular array
Control Rods	5 Ag/In/Cd safety rods; 1 stainless-steel regulating rod
Shielding	Pool water and concrete
Pool Volume	120 000 gal (4.54×10^5 L)
Reactor Building Construction Volume Leak Rate	 Reinforced concrete 2.85×10^5 ft^3 (8.1×10^3 m^3) 1.2×10^4 ft^3/h (9.4×10^{-2} m^3/s)
Stack Height (above main floor of Reactor Building)	188 ft (57 m)
Stack Exhaust Flow Rate (from Reactor Building	2.0×10^4 ft^3/m (9.4 m^3/s)
Location	Sterling Forest near Tuxedo, New York



(U) Fig. 12. Vertical sections through CINTICHEM reactor pool.

~~CONFIDENTIAL~~



(U) Fig. 13. CINTICHEM reactor building plan.

electromagnet. The force of gravity separates the control rod from the magnet, and the rod falls into the core. The time from scram initiation to full insertion is equal to or less than 800 ms.

(U) The regulating rod assembly consists of a stainless-steel rod fastened to a long extension attached to the drive mechanism. The regulating rod provides fine control of the reactor. The position of the regulating rod normally is servo-controlled to maintain constant reactor power. The rod also can be inserted or withdrawn manually but has no scram function.

(U) 3.7.3. Reactor Pool Arrangement. The reactor core is composed of MTR-type fuel assemblies and control-rod fuel assemblies, with built-in control-rod guides, inserted in the grid plate. The elements may be arranged in a variety of lattice patterns depending on experimental requirements.

(U) The reactor core is suspended from a movable bridge and immersed in a 49- by 23- by 32-ft (14.9- by 7.0- by 9.2-m)-high pool of demineralized water. The combined pool and stall volume totals 120 000 gal (454 000 L).

(U) The pool is divided into two sections separated by a 4-ft (1.2 m)-wide opening that can be closed by a removable watertight gate. The narrower stall section contains the fixed experimental facilities, such as the beam tubes and thermal column. The open end of the pool permits bulk irradiations and provides storage space for irradiated fuel and experiments.

(U) The stall area shielding consists of a 5.8-ft (1.77-cm)-thick magnetite concrete wall extending to a height of 15 ft (4.6 m) above the pool floor. The wall thickness is reduced to 3 ft (0.9 m) at the top of the stall. The first 4 ft (1.2 m) of the wall above the step is magnetite concrete, and the remainder is regular concrete.

(U) All sections of the pool and stall areas that are in contact with the reactor water are coated for ease of decontamination and to prevent interaction of the reactor water with the concrete. Areas normally exposed to high radiation are coated with glazed ceramic tile.

(U) The core support bridge, which is movable and on rails mounted on top of the pool walls, is constructed of structural steel; the central section incorporates a superstructure to allow for reactor control mechanisms and electrical equipment to be mounted. The bridge is moved by manual rotation of a crank handle for positioning the reactor in either the pool or stall. A locking device prevents accidental or unauthorized movement of the bridge.

(U) The core support tower, which is suspended from the bridge, is a structural aluminum frame. The walls of the pool contain six aluminum storage racks. The design is such that a critical array (with or without the core being considered) cannot be achieved with elements stored in the racks.

(U) 3.7.4. Emergency Core Cooling System. A core spray system is available for emergencies. It is operated manually after indication of a low water level and consists of piping and spray nozzles located above the core. If primary coolant is lost and the core is uncovered, the reactor core may be cooled by one of two spray nozzles--one discharging into the pool and the other discharging into the stall areas. Water is supplied to these nozzles from the municipal water system. If this should fail, a 100 000-gal (380 000-L) reservoir located on a hill above the facility is available to supply emergency reactor coolant. The valve that isolates this system must be operated manually to activate this water source.

(U) 3.8 University of Michigan Ford Nuclear Reactor

(U) The FNR is an open-pool type, heterogeneous research reactor that operates at a maximum licensed power level of 2 MW. The FNR converted from high-enriched uranium (HEU) to low-enriched uranium (LEU) fuel (~19.5% enrichment) in October 1984. Light demineralized water is used for moderation, cooling, and shielding. Graphite and heavy water (D_2O) are used for additional neutron reflection. The reactor is cooled by natural convection for power levels up to 100 kW and by forced convection for higher power operation. The reactor power is regulated by the insertion or withdrawal of three neutron-absorbing shim-safety rods and one regulating rod.

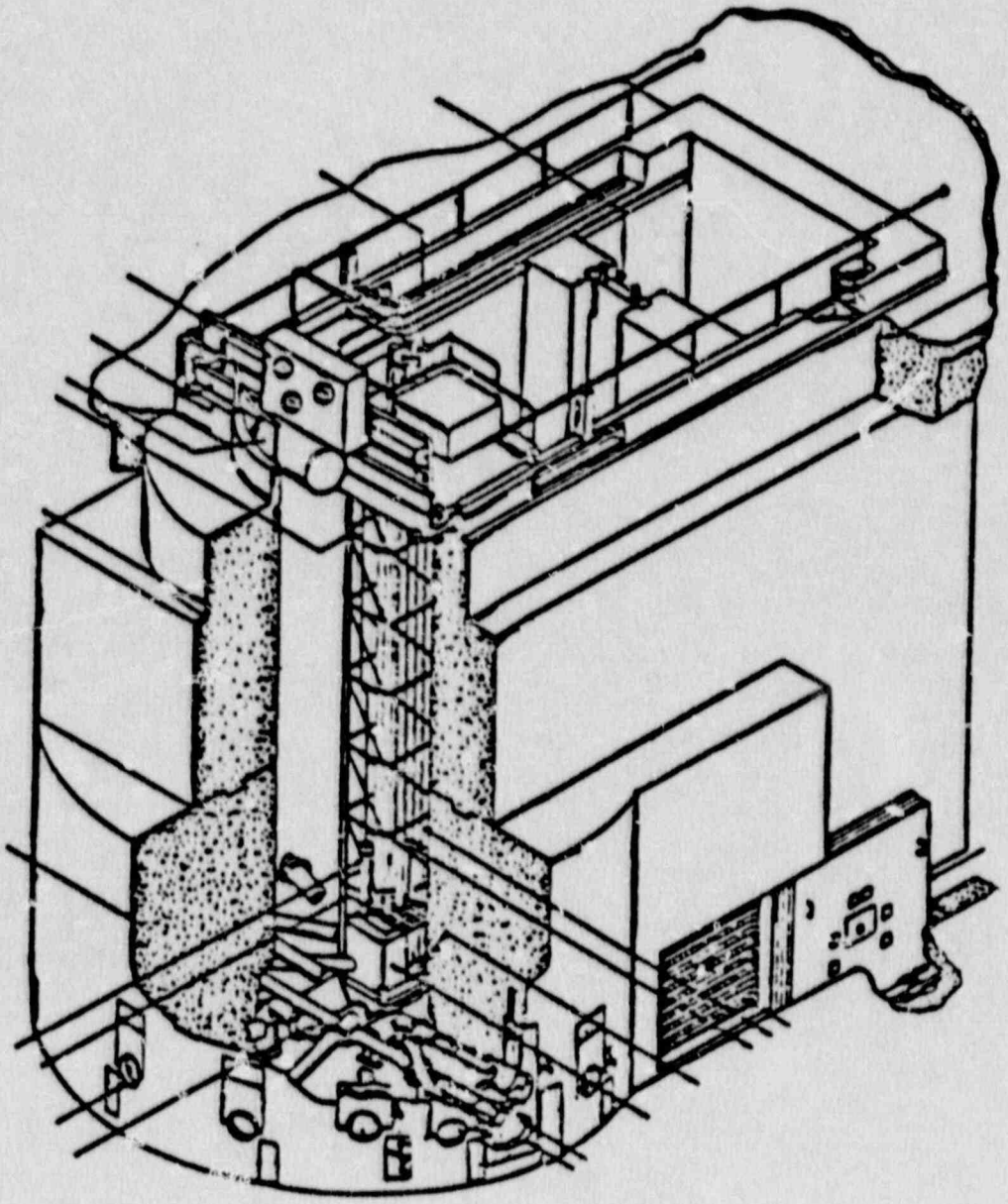
(U) The FNR is used principally for activation analysis, neutron irradiation studies, isotope production, neutron radiography, and training. The FNR operates an average of ~9000 MWh/yr. The parameters[5,13] for the facility that are used in this analysis are given in Table VIII. The physical arrangement of the reactor core, pool, and experimental facilities is shown in Figs. 14 and 15.

(U) 3.8.1. Fuel Elements. The FNR uses MTR-type fuel elements with overall dimensions of ~3.25 in. (8.26 cm) by 2.94 in. (7.47 cm) by 34.78 in. (88.34 cm). The fuel assemblies consist of curved plates containing uranium aluminide (UAl_x) or uranium oxide (U_3O_8) fuel enriched to less than 20% ^{235}U. Standard 18-plate aluminum-clad elements contain 5.9 oz (~167 g) ^{235}U. The shim-safety

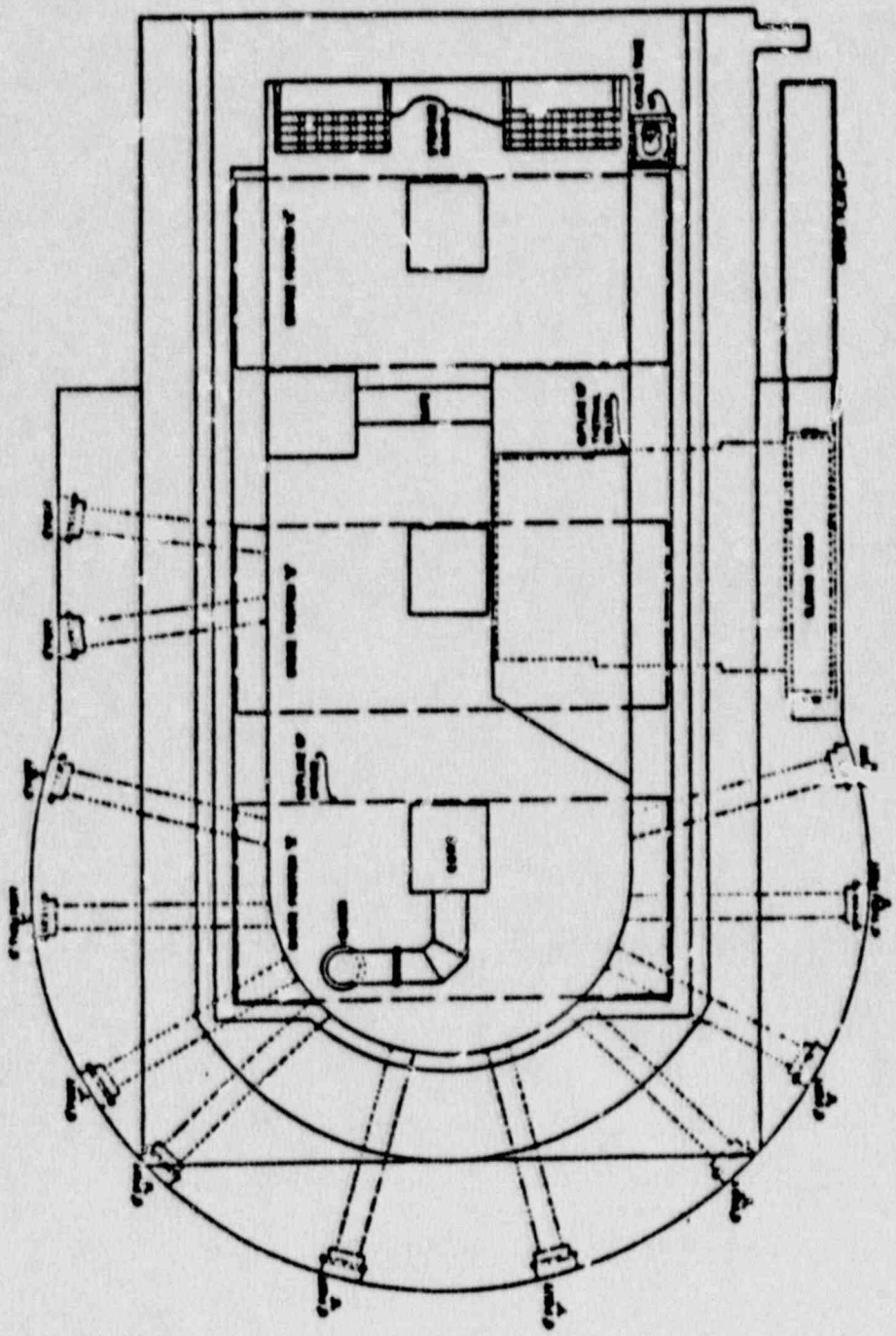
TABLE VIII

(U) UNIVERSITY OF MICHIGAN FORD NUCLEAR REACTOR FACILITY PARAMETERS

Power Level	2 MW
(C)	
Moderator and Coolant	Light water
Reflector	D_2O and graphite
Coolant Flow	900--1000 gal/min (57--63 L/s)
Fuel Type	MTR plate
Enrichment	19.5% ^{235}U
Number of Elements	35 to 44
Arrangement	6 by 8 rectangular array
Control Rods	3 shim-safety rods, 1 regulating rod
Pool Volume	40 500 gal (150 000 L)
Shielding	Pool water and concrete
Reactor Building	
Construction	Reinforced concrete
Volume	2.7×10^5 ft^3 (7.7×10^3 m^3)
Leak Rate	600 ft^3/h (4.7×10^{-3} m^3/s)
Stack Height	54 ft (16.5 m)
Stack Exhaust Flow Rate	
Reactor Building Stack	1.0×10^4 ft^3/min (4.7 m^3/s)
Reactor Building to PML 2	1.3×10^3 ft^3/min (0.6 m^3/s)
Location	University of Michigan, North Campus Ann Arbor, Michigan



(U) Fig. 14. Cutaway of FNR pool.



(U) Fig. 15. Horizontal cross section of FNR pool.

and regulating rods move in a control rod guide channel in a control rod fuel element; therefore, there are only nine fuel-bearing plates in such an element. These fuel elements can contain 3.0 oz (~84 g) ^{235}U. The active fueled length of the elements is ~24 in. (60.96 cm). The plates are made in a sandwich fashion with 0.015-in. (3.8-mm)-thick aluminum cladding surrounding a 0.03-in. (7.6-mm) layer of 42 wt% uranium of either uranium-aluminide or uranium oxide fuel.

(U) 3.8.2. Reactivity Control System. The FNR uses three boron-stainless-steel shim-safety rods and one stainless-steel regulating rod. All of the rods fit into the central gap in any of the control rod fuel elements and may be located in any of the 16 central core positions. The shim-safety rods are solid bars 1.0 in. (2.54 cm) by 3.0 in. (7.62 cm) by 24.0 (61.0 cm) long. The regulating rod is a hollow tube having the same dimensions as the shim-safety rods.

(U) The electromechanical drive assemblies for the shim-safety rods and the regulating rod consist of a motor and reduction gears that drive a rack and pinion and are coupled to an extension cable. The shim-safety rods are connected to the rack through an electromagnet and ferro-nickel armature. When electric power is lost or interrupted, the rods separate from the drive unit and fall by gravity, scramming the reactor.

(U) The shim-safety rods normally are withdrawn as a group (ganged) to a predetermined height in the core, but they also may be withdrawn individually. The shim-safety rods are used for coarse adjustment of reactor power; the regulating rod is used only for fine adjustments of power level. The regulating rod may be operated either manually or automatically. The regulating rod drive assembly does not have an electromagnet and therefore has no scram capability.

(U) 3.8.3. Reactor Pool Arrangement. The reactor core is suspended from a movable bridge mounted on rails that span the top of the concrete tank. It consists of 35 to 40 19.5%-enriched MTR-type curved-plate fuel elements, three shim-safety rods, and one regulating rod arranged in a rectangular configuration (6-by-8 fuel element array). A heavy water tank located directly behind the core provides a thermal neutron flux for the beam ports and also is used as a startup source. The core can be surrounded by graphite elements that serve as neutron reflectors.

(U) The aluminum suspension frame supports the grid plate, which contains the fuel elements, the ion chambers, the control and regulating rods, and the fission chamber guide tubes. The grid plate has 48 holes in a 6-by-8 rectangular

array for positioning the various core components. In addition, there are several 7/8-in. (2.22-cm)-diam holes drilled through the grid plate between the fuel element holes to provide additional cooling around the fuel elements. Water passes up through these holes during the natural convective cooling mode of reactor operation, and water is forced down through the holes during forced circulation. Additionally, several aluminum plugs are available to fill unused grid core positions and prevent the circulating water from bypassing the fuel elements. Special holddown mechanisms consisting of long extensions between the guide tubes of the control rod fuel elements and the control rod drive mechanisms are used to prevent any inadvertent withdrawal of the control rod fuel elements from the grid plate during reactor startups.

(U) The reactor is suspended ~20 ft (6.1 m) beneath the surface of a rectangular concrete pool that is 10 ft (3.05 m) wide by 20 ft (6.1 m) long by 27 ft (8.23 m) deep. The tank is lined with white ceramic tiles that prevent spalling and aid in visibility and decontamination. Additionally, a graphite-filled thermal column is located in the center of the west wall of the tank. The reactor is limited to 100-kW operation when it is near the thermal column position because there is only convective cooling in that position.

(U) Fuel storage racks are located along the north and south ends of the reactor pool. The pool is divided by two islands and an aluminum gate that can be used to isolate either half of the pool in the event of a leak. Also, a water-lock system located in the south end of the pool allows for the transfer of fuel, experiments, or samples from the reactor pool to a hot cave.

(U) Twelve aluminum beam ports penetrate the north end of the pool. Eight pneumatic irradiation tubes also penetrate the pool floor and terminate adjacent to the west face of the reactor core. Figure 15 shows a cross-section view of the reactor pool.

(U) 3.8.4. Emergency Core Cooling System. The large inventory of pool water is expected to provide adequate cooling of the reactor core under all accident conditions other than instantaneous loss of coolant while the reactor is at full power (2 MW).

(U) 3.9 University of Virginia Reactor

(U) The UVAR is an open-pool-type reactor using up to 7.7 lb (3.5 kg) of ^{235}U fuel enriched to approximately 93%. It is a light-water-moderated, graphite- or water-reflected reactor that currently is authorized to operate at steady-state power levels up to and including 2 MWth.

(U) The UVAR is used primarily for class instruction, student experiments, reactor operator training, research, and radioisotope production. The parameters[5,14] for the UVAR used in the analysis are given in Table IX. The general arrangement of the reactor pool is shown in Fig. 16.

(U) 3.9.1. Fuel Elements. The UVAR can operate with either flat-plate or curved-plate MTR-type fuel elements. The plates of both elements are a sandwich of aluminum cladding over a uranium-aluminum alloy "meat" approximately 0.02 in. (0.051 cm) thick and 23.5 in. (59.7 cm) long. The cladding is 0.015 in. (0.038 cm) thick except on the outer plates of the curved-plate elements, where it is 0.0225 in. (0.057 cm) thick. The overall dimensions for both types of fuel elements are approximately 34 in. (86.4 cm) long, 3 in. (7.62 cm) wide, and 3 in. (7.62 cm) thick.

(U) The standard flat-plate fuel element consists of 12 plates. The control rod element has the center six plates removed to allow space for inserting the rod. Partial elements also are available, and these have six fuel-bearing plates alternating with six nonfuel-bearing plates. Each standard flat-plate fuel element contains approximately 165 g of ^{235}U, and the control rod or partial element contains approximately one-half as much. A 0.211-in. (0.536-cm) space is provided between each flat plate for coolant flow.

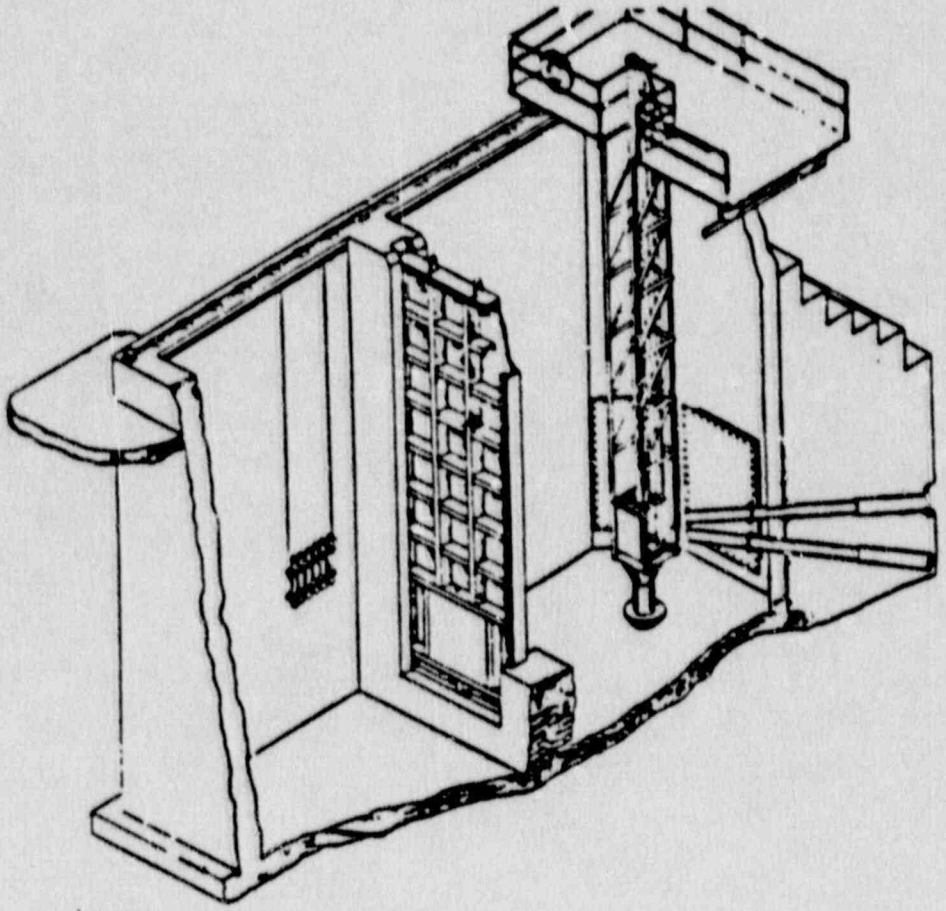
(U) Each standard curved-plate fuel element consists of 18 fuel-bearing plates, and the control rod element contains 9 fuel-bearing plates. A partial element contains nine fuel-bearing plates alternating with nine nonfuel-bearing plates. The standard curved-plate fuel element contains approximately 6.9 oz (195 g) of ^{235}U, and the control rod or partial element contains approximately 3.5 oz (98 g) of ^{235}U. The coolant gap in the curved-plate elements is 0.122 in. (0.31 cm).

(U) Although the UVAR may use either flat-plate or curved-plate fuel, the two types are not mixed in one core loading because this is an unreviewed question bearing on reactor safety.

TABLE IX ~~CONFIDENTIAL~~

(U) UVAR FACILITY PARAMETERS

Power Level	2 MW
(C)	
Moderator and Coolant	Light water
Reflector	Graphite and/or light water
Coolant Flow	900 gal/min (57 L/s)
Fuel Type	MTR plate
Enrichment	93% ^{235}U
Number of Elements	Maximum of 25 equivalent full elements
Arrangement	8 by 8 square array
Control Rods	3 shim-safety rods, 1 regulating rod
Pool Volume	75 000 gal (284 000 L)
Shielding	Pool water and concrete
Reactor Building	
Construction	Cinderblock and brick
Volume	8.3×10^4 ft^3 (2.4×10^3 m^3)
Leak Rate	2710 ft^3/min (1.28 m^3/s)
Stack Height	35 ft (10.7 m)
Stack Exhaust Flow Rate	7.6×10^3 ft^3/m (3.6 m^3/s)
Location	University of Virginia Campus Charlottesville, Virginia



(U) Fig. 16. Cutaway of the UVAR pool.

(U) 3.9.2. Reactivity Control System. The power level in the UVAR is controlled by three shim-safety rods and one regulating rod. All four rods are of the bayonet type, which fit into a central gap provided in special control rod fuel elements as discussed in the previous section. The rods and their fuel elements can be located in any core position.

(U) The shim-safety rods are made of boron-stainless steel clad with aluminum. The absorbing section, which is approximately 1.5% boron by volume, is 24.8 in. (63.0 cm) long and has a 2.25- by 0.875-in. (5.72- by 2.22-cm) cross section with semicircular ends. Each shim-safety rod is worth approximately 3% Δk/k, and each is moved in and out of the core by an individual electromechanical system. The drive mechanism, which is supported by the bridge, consists of an electric motor and lead-screw drive. The rod containing the absorber section is suspended from the drive mechanism by an electromagnet. During normal operation, these rods are driven either in or out at a rate of 3.7 in./min (0.16 cm/s). When a scram signal is received, the magnets are deenergized and the shim-safety rods drop freely into the core. The rods are inserted fully in less than 1 s.

(U) The regulating rod, which has the same dimensions as the shim-safety rods, is stainless steel with an aluminum cladding. Its reactivity worth is approximately 0.5% Δk/k. The regulating rod has a similar drive mechanism, but the rod is permanently fixed to it. The rod travels at a speed of approximately 24 in./min (1.0 cm/s) in either direction and does not drop on a scram signal.

(U) 3.9.3. Reactor Pool Arrangement. The reactor pool is 32 ft (9.75 m) long (north-south), 12 ft (3.66 m) wide, and 26.3 ft (8.02 m) deep and holds about 75 000 gal (281 000 L) of water. The core is suspended in the pool by an aluminum framework attached to a movable bridge. The bridge moves in a north-south direction on rails positioned along the east and west sides of the pool. The bridge is restrained so the reactor cannot be brought closer than 4 ft (1.22 m) from the pool walls. The reactor's vertical position is fixed; the bottom of the core is 4.5 ft (1.37 m) above the pool floor. With this core elevation, the top of the active fuel region is 19.75 ft (6.02 m) below the surface of the water when the pool is full.

(U) For reasonable lengths of operating time at low power levels, the heat capacity of the pool is sufficient to permit operation at power without the use of an external heat exchanger. However, power levels of several hundred kilowatts require dissipating the heat to an external heat dump. Forced convection cooling of the core is necessary to operate near and above 1 MW. The UVAR has

a coolant outlet header at the south end of the pool that can be raised to contact the bottom of the core plenum. For operation at more than 200 kW, the core is located above this header and the primary coolant is pumped downward through the fuel and then through the external heat exchanger and back to the pool. For power levels of 200 kW or less, the reactor core may be operated with natural convective cooling at the north end of the pool or at the south end of the pool with the header disengaged.

(U) The fuel elements are held in place by a grid plate containing an 8-by-8 array of holes for positioning the fuel elements and experimental apparatus. The minimum critical loading with water as the reflector is a 4-by-5 array of fuel elements. Graphite elements also can be loaded around the core to act as a reflector. In this configuration, the minimum critical loading is a 4-by-4 fuel array surrounded on all four sides by two rows of graphite elements.

(U) Not all of the positions in the grid plate are filled by either fuel or graphite elements for many core configurations. For these loadings, plugs are fitted into any empty holes so that the cooling water passes down through the fuel elements rather than through the open holes when forced circulation is used. The grid plate also contains a series of small holes interspaced between the positioning holes to provide cooling flow between the elements.

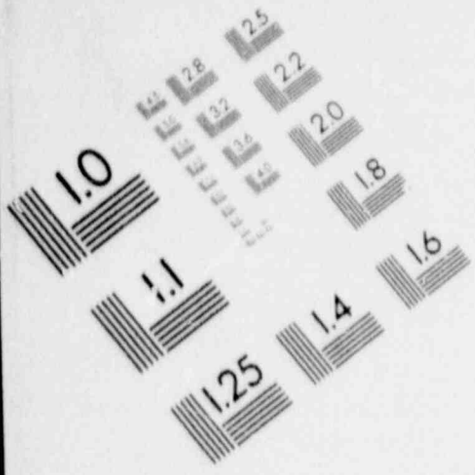
(U) The south wall of the reactor pool is penetrated by two large access facilities measuring 5 ft (1.52 m) wide by 5 ft (1.52) high. When not in use, these experimental facilities are filled with concrete and lead bricks backed by a dolly-mounted, stepped concrete block. Each facility is closed off from the pool by a gasketed aluminum plate.

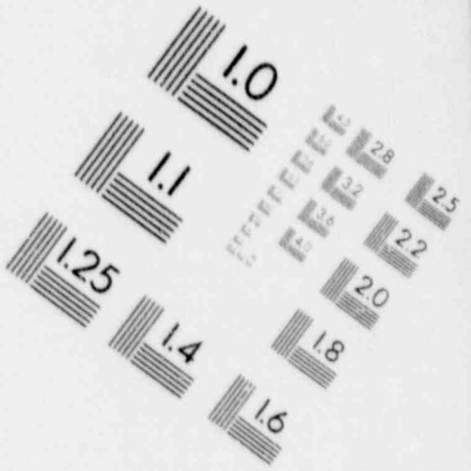
(U) A thermal column can be incorporated into one of these large access facilities by replacing most of the shielding with graphite blocks. At the present time, one of these large access facilities is equipped with a small penetration to a tangential exposure chamber so that target radiations can be studied without the interference of direct core radiation.

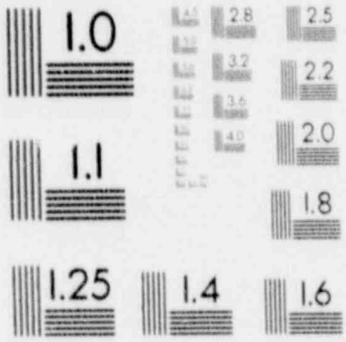
(U) In addition, two 8-in. (20.3-cm) beam ports penetrate the concrete shield on the south side of the pool. When not in use, they are filled with concrete plugs with an offset in diameter to reduce radiation streaming. The door to each port contains a 3-in. (7.62-cm) lead shield. Aluminum ports extend the beam ports to the reactor face. These ports normally are filled with water but can be drained individually when in use. A blind-flange aluminum plate separates the aluminum port extension from the concrete shield penetration. When

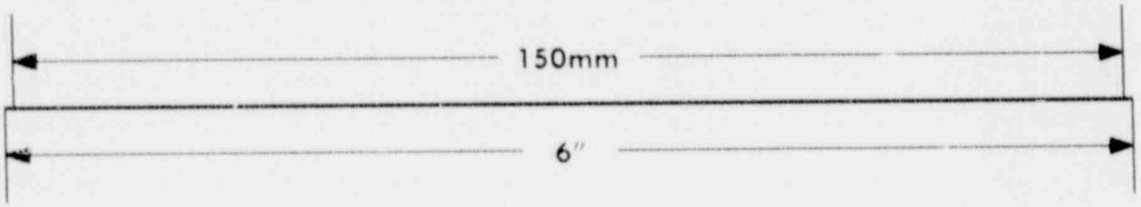
1

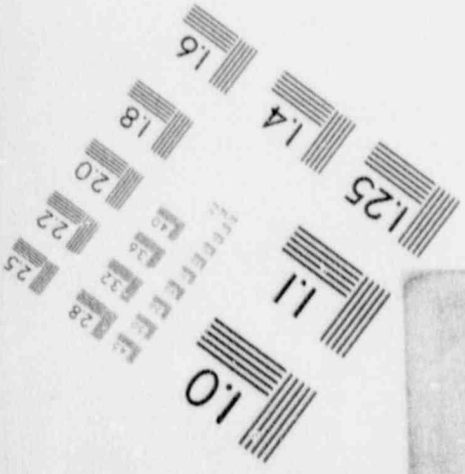
IMAGE EVALUATION TEST TARGET (MT-3)

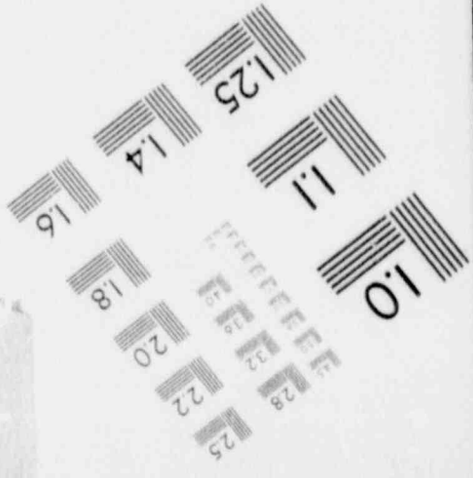






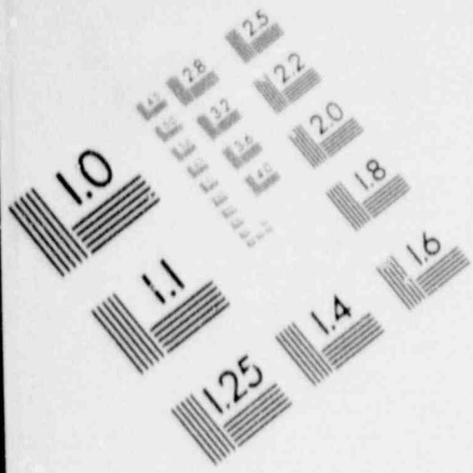


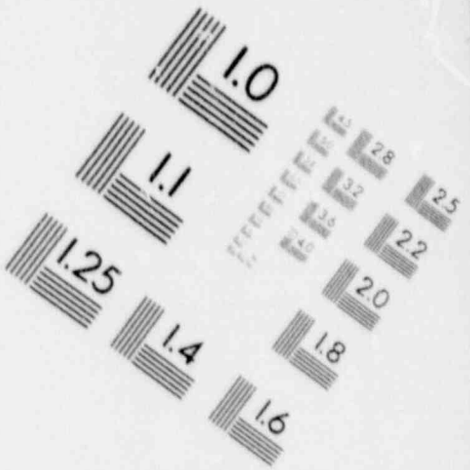


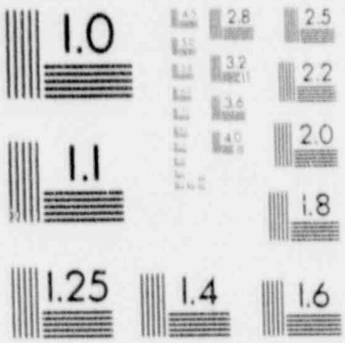


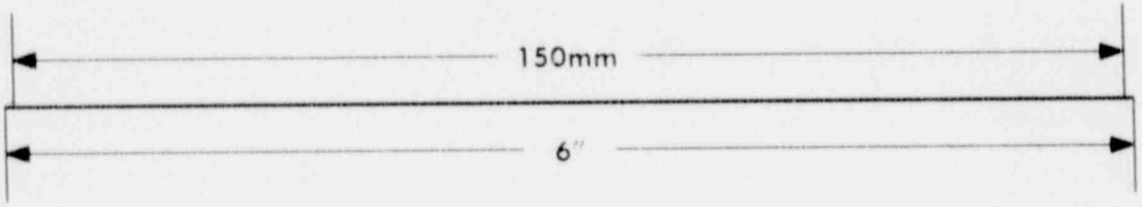
1

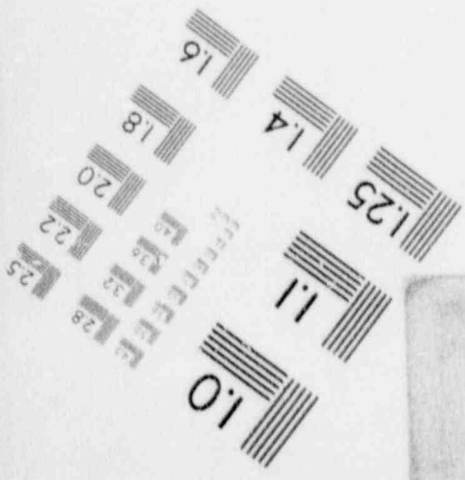
IMAGE EVALUATION TEST TARGET (MT-3)

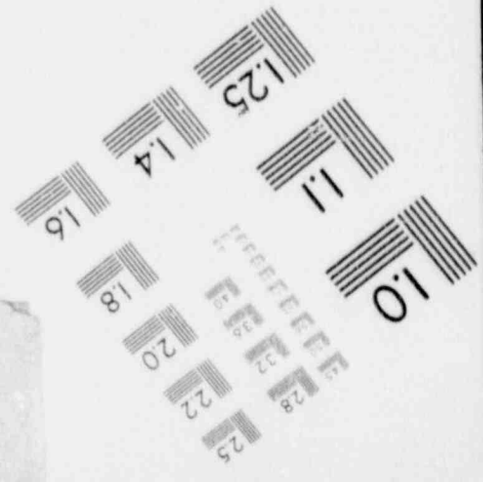


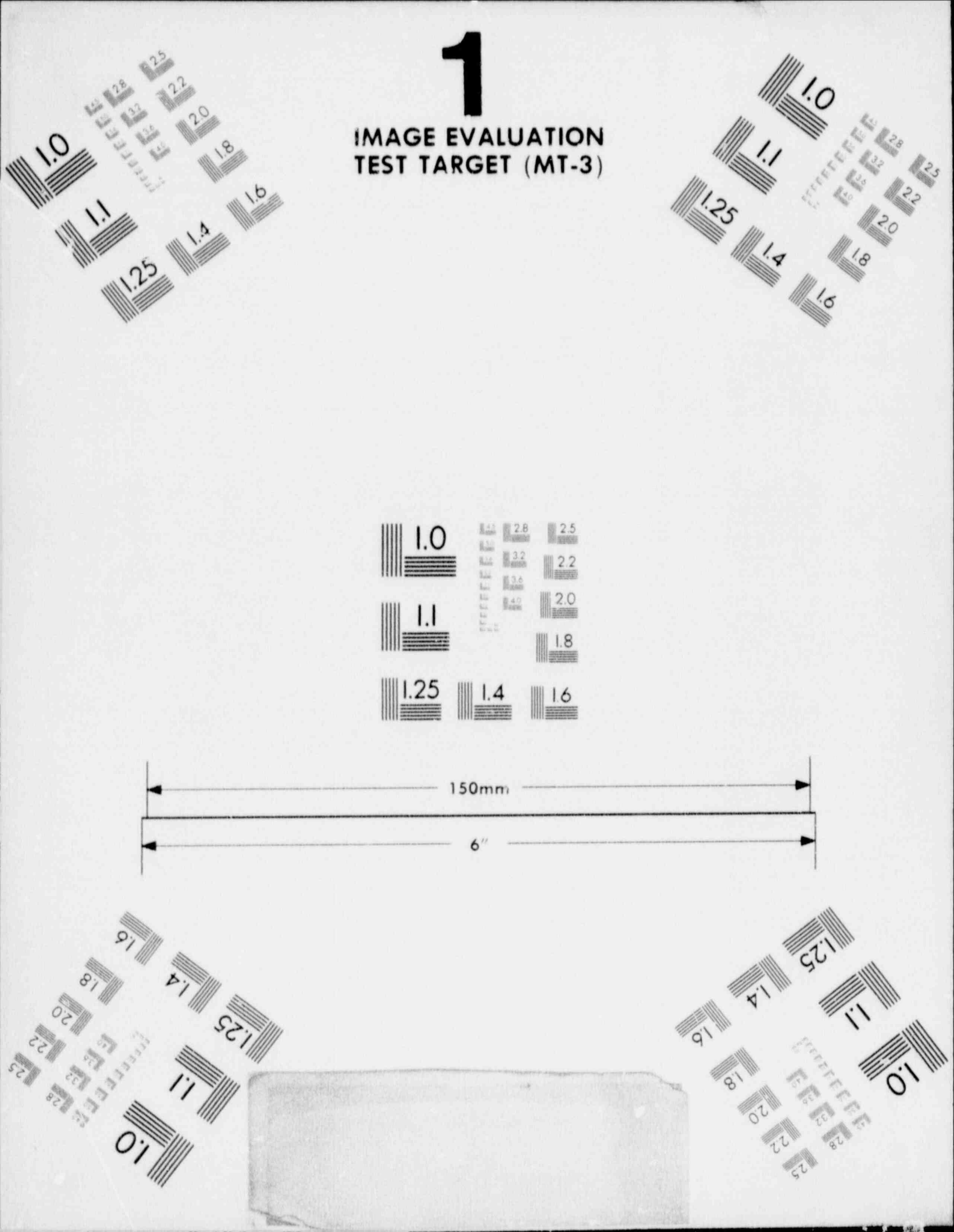






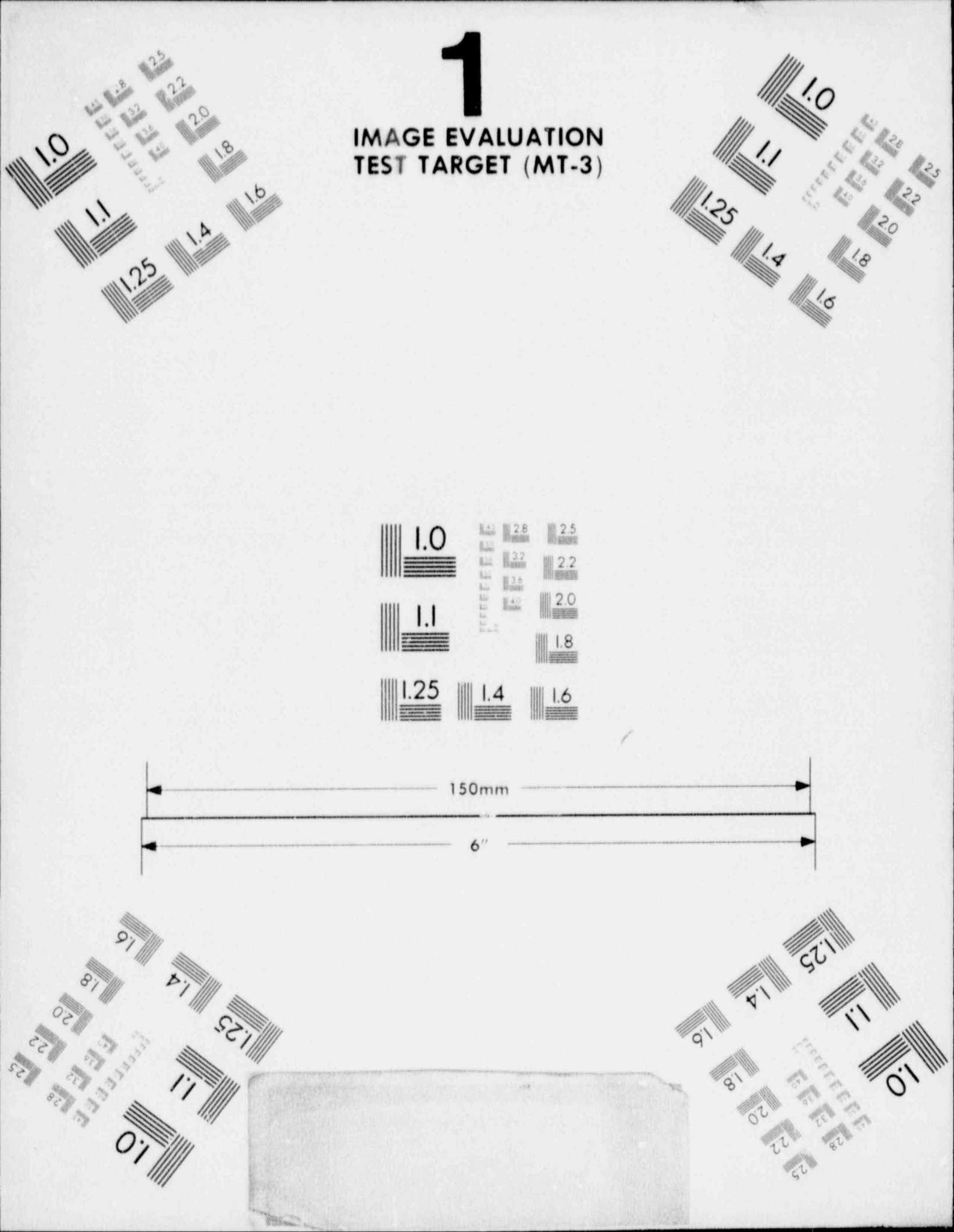





1
IMAGE EVALUATION
TEST TARGET (MT-3)
1.0
1.1
1.25
1.4
1.6
1.8
2.0
2.2
2.5
2.8
3.2
3.6
4.0
150mm
6"

1

IMAGE EVALUATION TEST TARGET (MT-3)



these beam ports are used, external shield walls or beam stops must be installed to control radiation levels in the experimental area.

(U) 3.9.4. Emergency Core Cooling System. A core spray system is available to provide cooling to the fuel as protection against melting in the event of a loss-of-coolant accident (LOCA). There are two completely independent systems, each with a pair of spray headers and its own emergency water storage tank. Each system is designed to deliver an average spray flow of 10 gal/min (0.63 L/s) over the core for at least 30 min and not less than 7.5 gal/min (0.47 L/s) for an additional 1 h, which will reduce the core temperature to a point that cooling by air convection is adequate to prevent fuel melt.

(U) Each system has a 1500-gal (5.68×10^3-L) emergency water storage tank mounted on the wall inside the pool. Recirculating water from the demineralizer is returned to each of these tanks. An overflow from each tank is located about 2 in. (5.1 cm) above the highest operating level of the pool water. Accordingly, there always will be a slight head (2 in. or 5.1 cm) of water in the tank and a small flow of water through the headers. This ensures that the tank is always full and that stagnant water and resultant corrosion do not occur in the spray headers. The entire system is made of aluminum and stainless steel to inhibit corrosion. Thus, the system is always ready for an immediate supply of core spray water in case water is lost from the pool. There are no moving parts that can fail, and there are no automatic electronic or mechanical devices that are required to function.

(U) 4.0 FAULT-TREE ANALYSIS

(U) Using fault-tree analysis techniques, we created a master logic diagram to identify possible radiological releases from the building (Fig. 17). Each of the events on the diagram was developed to identify the size of risk involved. We have summarized the events that are generally applicable to all research reactors and then examined events applicable to specific sites. The section numbers below correspond to events on the diagram.

(c)

(c)

(c)

(c)

(c)

(c)

(c)

(c)
(c)

(U) 5.0 FACILITY ANALYSES

(U) Below are the analyses of potential events that might result in the release of signiricant quantities of radioactive material from the individual reactor cores.

(U) (C) 5.1 National Bureau of Standards Reactor Sabotage Scenarios

(c)

(c)

(c)

(c)

(c)

(S) 5.2 Georgia Tech Research Reactor Sabotage Scenarios

(c)

(c)

(c)

(c)

(c)

(C)

(U) 5.3 University of Missouri Research Reactor Sabotage Scenarios

(C)

(C)

(c)

(c)

(c)

(U) 5.4 Massachusetts Institute of Technology Reactor Sabotoge Scenarios
(C)

(C

(C

(c)

(c)

(c)

(c)

(U) 5.5 Rhode Island Nuclear Science Center Reactor Sabotage Scenarios

(c)

(c)

(c)

(c)

(c

(c

(U) 5.6 Buffalo Materials Research Center Reactor Sabotage Scenarios

(C)

(C

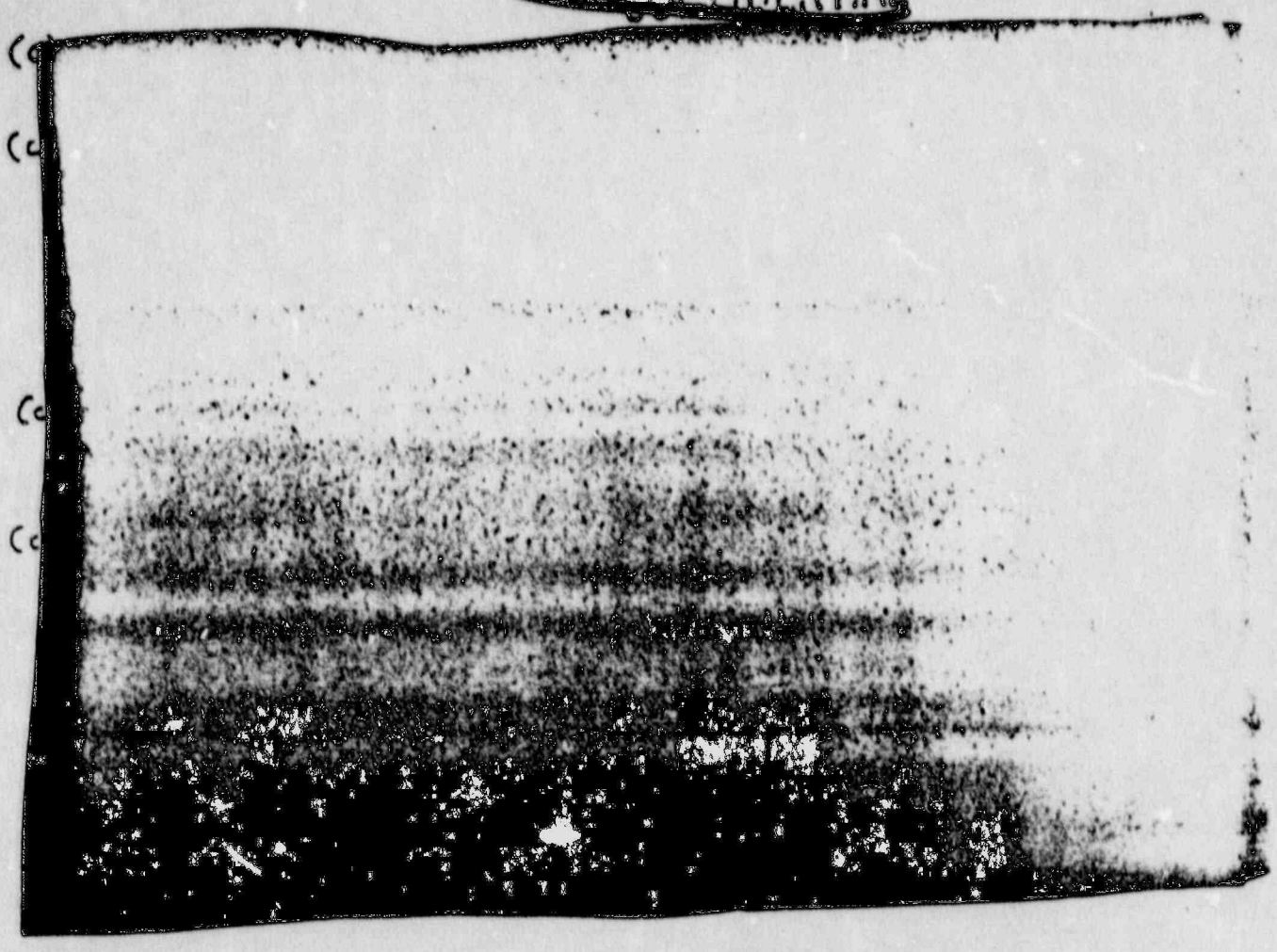
(C

(

(U) 5.7 CINTICHEM Inc. Reactor Sabotage Scenarios

(C)

(C



(U) 5.8 University of Michigan Ford Nuclear Reactor Sabotage Scenarios

(C)

(c)

(c)

(U) 5.9 University of Virginia Reactor Sabotage Scenarios

(U) This reactor also is an open-pool, metallic-fueled arrangement.

(c)

(c)

(c)

(c

(U) 6.0 RADIONUCLIDE RELEASE CONSEQUENCES

(C)

[redacted]

(U) 6.1 Source Terms

(U) The source terms for the individual reactors were calculated using the CINDER computer code.[3] CINDER calculated the significant radionuclide contribution resulting from the fission-product build-up based on the fuel irradiation histories of each reactor.

(C) [redacted]

(U) The model based on Lewis[17] provides for calculating a time-dependent concentration that is released into the building wake cavity. The model reduces to a differential expression described by

$$\frac{dX}{dt} = \frac{R}{V} - \frac{FX}{V} \quad . \tag{1}$$

where

R = release rate from the core (Ci/s),
V = volume (m^3) (total transport path volume),
X = concentration (Ci/m^3), and
F = flow rate (m^3/s) [based on a leak rate in an unventilated volume (<10% volume/h)].

Note that

$R_s = FX$ when R_s is in units of curies per second ($Ci/m^3 \cdot m^3/s$),

where

R_s = release rate from building (Ci/s) .

Equation (1) reduces to

$$X = \frac{R}{F}\left(1 - e^{-Ft/V}\right) \quad . \tag{2}$$

To get the time-averaged value of the concentration, Eq. (2) gives us

$$\langle X \rangle = \frac{R}{F} \frac{1}{t_2 - t_1} \int_{t_1}^{t_2} \left(1 - e^{-Ft/V}\right) dt \quad . \tag{3}$$

If we let $t_1 = 0$ be the initiation of radionuclide release to the building volume, then $\langle\ \rangle$ reduces to

$$\langle X \rangle = \frac{R}{F} \frac{1}{t_2} \int_0^{t_2} (1 - e^{-Ft/V}) \, dt \quad . \tag{4}$$

Evaluated at 0, Eq. (4) gives

$$\langle X \rangle = \frac{R}{F} + \frac{R}{Ft_2} \frac{V}{F} (-1 + e^{-Ft_2/V}) \quad . \tag{5}$$

If we replace R with our source term $R_s = F\chi_o$, then we get Eq. (6).

$$\langle X \rangle = \chi_o \left[1 - \frac{V}{Ft_2} (1 - e^{-Ft_2/V}) \right] \quad . \tag{6}$$

where χ_o is the building concentration at $t = o$

Equation (6) provides the radionuclide flow rate and average concentration out to the building wake cavity.

(U) For each reactor, the significant radionuclides for the whole-body exposures were determined using the following equation.

$$S_i = \frac{Y_i \cdot RF_i \cdot DCF_i}{\Sigma_i (Y_i \cdot RF_i \cdot DCF_i)} \quad , \tag{7}$$

where

S_i = significance index for the i^{th} radionuclide,
Y_i = curie yield for the i^{th} radionuclide,
RF_i = appropriate release fraction for the i^{th} radionuclide, and
DCF_i = appropriate dose conversion factor for the i^{th} radionuclide.

(C)

(c)

(c

(

TABLE X [redacted]

MOST SIGNIFICANT RADIONUCLIDES FOR EXTERNAL WHOLE-BODY DOSE (U)

(U) Isotope	(U) Half-Life	(C)	(C)	(U) Decay Constant s^{-1}	(U) Dose Conversion Factor[C] (rem/s per Ci/m^3)
^{87}Kr	76.300 min			1.51×10^{-4}	1.5×10^{-1}
^{88}Kr	02.840 h			6.78×10^{-5}	3.7×10^{-1}
^{89}Kr	03.160 min			3.64×10^{-3}	3.5×10^{-1}
^{131}I	08.040 days			9.98×10^{-7}	6.7×10^{-2}
^{132}I	02.300 h			8.37×10^{-5}	4.0×10^{-1}
^{133}I	20.800 h			9.26×10^{-6}	1.1×10^{-1}
^{134}I	52.600 min			2.20×10^{-4}	4.6×10^{-1}
^{135}I	06.610 h			2.91×10^{-5}	2.9×10^{-1}
$^{135\ m}Xe$	15.360 min			7.38×10^{-4}	7.5×10^{-2}
^{137}Xe	03.830 min			3.02×10^{-3}	3.2×10^{-2}
^{138}Xe	14.130 min			8.15×10^{-4}	2.1×10^{-1}
^{103}Ru	39.400 days			2.04×10^{-7}	8.2×10^{-2}
^{134}Te	42.000 min			2.76×10^{-4}	1.5×10^{-1}

(C)

(C)

(U) [C]D. C. Kocher, "Dose Rate Conversion Factors for External Exposures to Photon and Electron Radiation from Radionuclides Occuring in Routine Releases from Nuclear Fuel Cycle Facilities," Health Physics 38, 543 (1980).

UNCLASSIFIED

UNCLASSIFIED

[redacted]

TABLE XI

MOST SIGNIFICANT RADIONUCLIDES FOR THYROID DOSE (c)

(c) Isotope	(c) Half-Life	(c) Decay Constant λ (s^{-1})	(c) Dose Conversion Factor[c] (rem/Ci inhaled)
^{131}I	08.04 days	9.98×10^{-7}	1.1×10^{6}
^{133}I	20.80 h	9.26×10^{-6}	1.8×10^{5}
^{134}I	52.60 min	2.20×10^{-4}	1.1×10^{3}
^{135}I	06.61 h	2.91×10^{-5}	4.4×10^{4}
^{132}Te	78.00 h	2.46×10^{-6}	2.2×10^{-5}

(c) UNCLASSIFIED

(c) UNCLASSIFIED

(c) [c]Letter to Jack Corley from J. W. Healy, HSE-DO, Los Alamos National Laboratory, May 25, 1984, that used ICRP-30 methodology to develop simple conversion factors.

(U) Following are the hypothetical operating schedules to predict the maximum achievable core inventory for the individual reactors. These were used in determining the input parameters required by CINDER code.

(C)

(C)

(C)

(C)

TABLE XII

(c) MOST SIGNIFICANT RADIONUCLIDES FOR EXTERNAL WHOLE-BODY DOSE FROM BMRC REACTOR

Isotope	Half-Life	Decay Constant λ (S^{-1})	Dose Conversion Factor[a] (rem/s per Ci/m^3)	Curies Released
^{85m}Kr	67.6 min	4.30×10^{-5}	2.9×10^{-2}	0.638
^{85}Kr	64.8 days	2.05×10^{-9}	3.9×10^{-4}	0.035
^{87}Kr	76.3 min	1.51×10^{-4}	1.5×10^{-1}	1.231
^{88}Kr	02.84 h	6.78×10^{-5}	3.7×10^{-1}	1.735
^{89}Kr	03.16 min	3.64×10^{-3}	3.5×10^{-1}	2.247
^{131}I	08.04 days	9.98×10^{-7}	6.7×10^{-2}	0.987
^{132}I	02.3 h	8.37×10^{-5}	4.0×10^{-1}	8.650
^{133}I	20.8 h	9.26×10^{-6}	1.1×10^{-1}	13.367
^{134}I	52.6 min	2.20×10^{-4}	4.6×10^{-1}	15.496
^{135}I	06.61 h	2.91×10^{-5}	2.9×10^{-1}	12.590
^{133m}Xe	02.19 days	3.67×10^{-6}	5.4×10^{-3}	0.065
^{133}Xe	05.25 days	1.53×10^{-6}	6.0×10^{-3}	2.229
^{135m}Xe	15.36 min	7.38×10^{-4}	7.5×10^{-2}	0.405
^{135}Xe	09.083 hr	2.12×10^{-5}	4.5×10^{-2}	2.722
^{137}Xe	03.83 min	3.02×10^{-3}	3.2×10^{-2}	2.195
^{138}Xe	14.13 min	8.15×10^{-4}	2.1×10^{-1}	2.121

[a] D. C. Kocher, "Dose Rate Conversion Factors for External Exposures to Photon and Electron Radiation from Radionuclides Occurring in Routine Releases from Nuclear Fuel Cycle Facilities," Health Physics 38, 543 (1980).

TABLE XIII

(U) MOST SIGNIFICANT RADIONUCLIDES FOR THYROID DOSE FROM BMRC REACTOR

Isotope	Half-Life	Decay Constant λ (S^{-1})	Dose Conversion Factor[a] (rem/Ci Inhaled)	Curies Released
^{131}I	08.04 days	9.98×10^{-7}	1.1×10^{6}	0.987
^{132}I	02.03 h	8.37×10^{-5}	6.3×10^{5}	8.650
^{133}I	20.80 h	9.26×10^{-6}	1.8×10^{3}	13.367
^{134}I	52.60 min	2.20×10^{-4}	1.1×10^{3}	15.496
^{135}I	06.61 h	2.91×10^{-5}	4.4×10^{4}	12.590

[a]Letter to Jack Corley from J. W. Healey, HSE-DO, Los Alamos National Laboratory, May 25, 1984, that used ICRP-30 methodology to develop simple conversion factors.

(c)

(c)

(c)

(c)

(c)

(c)

(U) 6.2 Release Paths

(C)

(C)

(C)

(C)

(c)

(c)

(c)

(c)

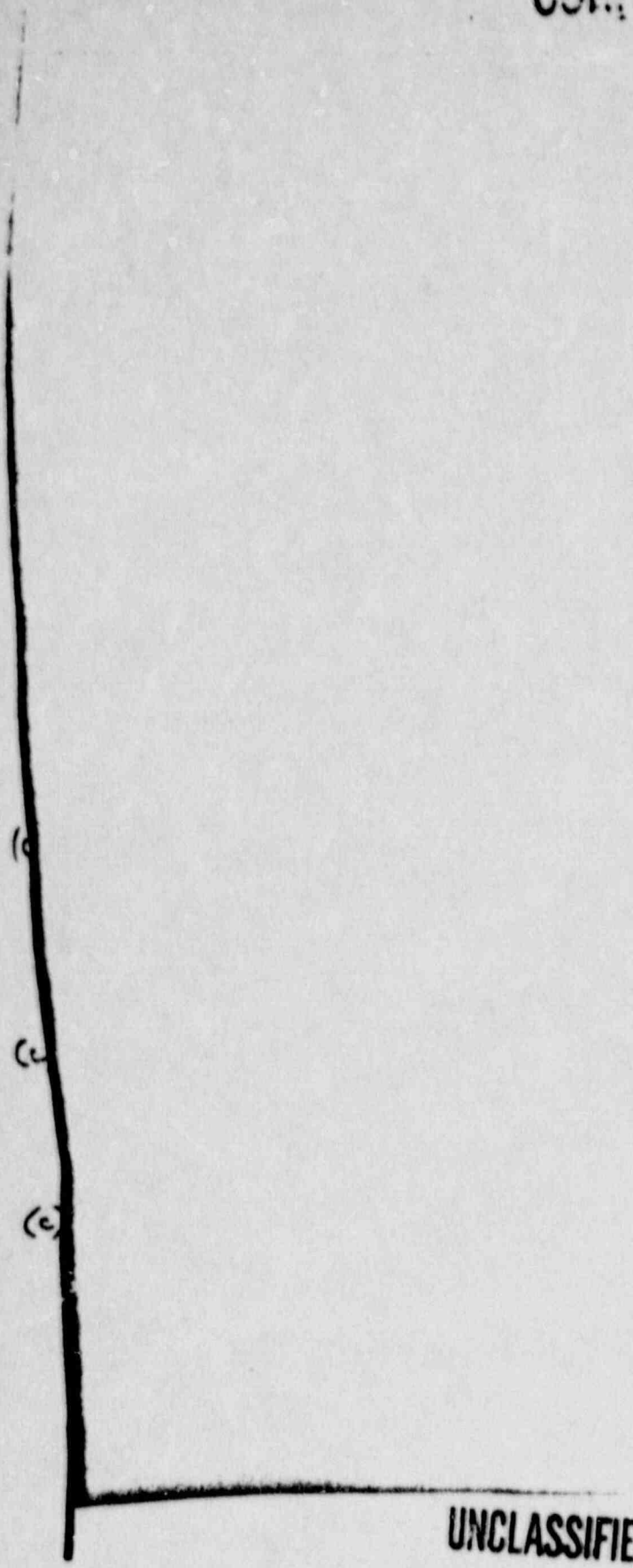
UNCLASSIFIED

(c)

(c)

(c)

(c)



(c)

(c)

(c)

(c)

(c)

(c)

(U) 6.4 Methodology Used in Calculating Exposures

(U) The consequences of the sabotage-induced releases were assessed by calculating the whole-body and thyroid exposures. The whole-body dose resulting from direct exposure to the radionuclide cloud is assumed to stop when the cloud passes. The thyroid dose is received from inhaling the radionuclide cloud and the subsequent incorporation of the radionuclides until they are removed from the thyroid by biological elimination and radioactive decay.

(U) We used whole-body dose conversion factors from D. C. Kocher in our calculations.[1] The dose conversion factors used to calculate the thyroid dose from inhalation were obtained from a letter to J. Corley from J. W. Healy, HSE-DO, Los Alamos National Laboratory, dated May 25, 1984, that used ICRP-30 methodology to develop single conversion factors.

(U) Three dispersion models were used to calculate the downwind radionuclide concentrations. A cavity model[21] was used to estimate the concentration inside the wake cavity. The second model was the model used in NRC Regulatory Guide 1.145,[22] which is valid for releases less than 2.5 times the height of the nearest building. The last model used to estimate the downwind concentrations was the Gaussian plume model.[23,24] Dry deposition and plume rise corrections were included in the calculations.

(U) The significant radionuclides used for the whole-body and thyroid dose calculations were determined by multiplying the curie yield, release fraction, and dose conversion factor for each radionuclide and dividing this by the sum of these factors for all of the radionuclides.

(U) After determining the significant radionuclides for the whole-body and thyroid exposures, potential doses were calculated for each dispersion model.

(U) 6.4.1. Cavity Model. The concentration inside the wake cavity was estimated using

$$X_i = \frac{Q_i}{C \cdot U \cdot AP} \quad , \tag{8}$$

where

X_i = the concentration of the i^{th} radionuclide at the receptor (Ci/m^3),
Q_i = the building release rate of the i^{th} radionuclide (Ci/s),
C = a constant between 0.5 and 5.0,
U = the average wind speed (m/s), and
AP = the frontal area of the building facing the wind (m^2).

(U) This equation is used for estimating close-in concentrations and assumes that the effluent is captured entirely in the building wake cavity. The mixing process in the cavity is not actually rapid enough to overcome some nonuniformities in the radionuclide concentrations, but the equation does estimate the average concentration inside the wake cavity.

(U) 6.4.2. NRC Regulatory Guide 1.145 Model. The NRC uses the following three equations to calculate the radionuclide concentrations downwind for release heights that are less than 2.5 times the height of the nearest building.

$$X_i = \frac{Q_i}{\left[U \left(\pi \cdot S_y \cdot S_z\right) + \left(\frac{A}{2}\right)\right]} \quad , \tag{9}$$

where

X_i = the concentration of the i^{th} radionuclide at the receptor (Ci/m^3),
U = the average wind speed (m/s),
π = 3.14159,

S_y = the standard deviation of the distribution of material in the plume in the lateral direction (m),

S_z = the standard deviation of the distribution of material in the plume in the vertical direction (m),

A = the smallest vertical plane cross-sectional area of the building (m^2), and

Q_i = the release rate of the i^{th} radionuclide (Ci/s).

(U) This first equation [Eq. (9)] uses the volumetric wake correction with a shape factor of 1/2 and the minimum cross-sectional area of the building.

In the second equation,

$$X_i = \frac{Q_i}{3 \cdot U \cdot \pi \cdot S_y \cdot S_z} \quad , \tag{10}$$

where

X_i = the concentration of the i^{th} radionuclide at the receptor (Ci/m^3),

U = the average wind speed (m/s),

π = 3.14159,

S_y = the standard deviation of the distribution of material in the plume in the lateral direction (m),

S_z = the standard deviation of the distribution of material in the plume in the vertical direction (m), and

Q_i = the release rate of the i^{th} radionuclide (Ci/s).

(U) This is the basic equation for a ground-level release with a reduction factor of 3 used for additional dispersion produced by the turbulent wake.

In the third equation,

$$X_i = \frac{Q_i}{U \cdot \pi \cdot \Sigma_y \cdot S_z} \quad , \tag{11}$$

where

$_i$ = the concentration of the i^{th} radionuclide at the receptor (Ci/m^3);

U = the average wind speed (m/s);

π = 3.14159;

Σ_y = the lateral plume spread with meander and building wake effects, with $\Sigma_y = (M \cdot S_y)$, where M is the meander factor, S_y is the standard deviation of the distribution of material in the plume in the lateral direction (m);

S_z = the standard deviation of the distribution of material in the plume in the vertical direction (m); and

Q_i = the release rate of the i^{th} radionuclide (Ci/s).

(U) This third equation [Eq. (11)] is an empirical formulation based on NRC staff analysis of the atmospheric dispersion experiment results at Rancho Seco.[25] Correction factors for building wake effects and meander according to atmospheric stabilities were taken from the figure in NRC Regulatory Guide 1.145.

(U) A conditional use of Eqs. (9) and (10) provides for an assessment of the atmospheric diffusion including only the effects of building wake mixing with wind speeds greater than 3 m/s (7 mph). These two equations over-estimate the ground-level concentration under stable conditions; therefore, Eq. (11) is used to correct this situation as it takes into account both building wake effects and meander of the plume during stable conditions. The higher value of Eq. (9) and Eq. (10) is compared with Eq. (11), and the lower value of this last comparison is used in the calculations.

(U) 6.4.3. Gaussian Plume Model. The Gaussian plume model is used to estimate the downwind radionuclide concentrations from an isolated source. It is estimated to be within a factor of 2 for distances of 0.06 to 12.4 miles (0.1 km to 20 km).[26] Beyond 12.4 miles (20 km), Gaussian dispersion calculations can be considered only to be order of magnitude estimates. The distribution of material in the plume is assumed to have a Gaussian distribution in both the vertical and horizontal directions. The following equation was used in this study.

$$X_i = \frac{Q_i}{2\pi \cdot U \cdot S_y \cdot S_z} \cdot e^{-0.5y^2/S_y^2} \cdot \left(e^{-0.5(z-h)^2/S_z^2} + e^{-0.5(z+h)^2/S_z^2}\right), \quad (12)$$

where

X_i = the concentration of the i^{th} radionuclide at the receptor (Ci/m^3),

Q_i = the release rate of the i^{th} radionuclide (Ci/s),

π = 3.14159,

U = the average wind speed (m/s),

S_y = the standard deviation of the distribution of material in the plume in the lateral direction (m),

S_z = the standard deviation of the distribution of material in the plume in the vertical direction (m),

y = the distance the receptor is off the centerline (m),

z = the height of the receptor (m), and

h = the height of the release (m).

(U) When plume rise was included in the calculation, the plume rise height was added to the release height to obtain an effective stack height. This was used in place of h in the above equation. Dry deposition also was included in the plume calculations.

(U) 6.4.4. Whole-Body Dose Calculations. The following equation was used to calculate the whole-body dose.

$$D_i = X_i \cdot DILUTE \cdot DCF_i \cdot DEPFACT_i \cdot DECAY_i \cdot RF_i \cdot T \ , \quad (13)$$

where

D_i = the whole-body dose at the receptor from the i^{th} radionuclide (rem),

X_i = the (uncorrected)concentration of the i^{th} radionuclide at the receptor location (Ci/m^3), as calculated in Eqs. (8--12),

DILUTE = the dilution factor for the building that addresses the transport path from the core to the point of release,

DCF_i = the dose conversion factor of the i^{th} radionuclide (rem/s per Ci/m^3),

$DEPFACT_i$ = the dry deposition correction factor of the i^{th} radionuclide,

$DECAY_i$ = the decay correction factor of the i^{th} radionuclide as a result of holdup,

RF_i = the release fraction of the i^{th} radionuclide, and

T = the time the receptor is in the plume, (s)

(U) 6.4.5. Thyroid Dose Calculations. The following equation was used to calculate the thyroid dose.

$$D_i = X_i \cdot DILUTE \cdot DCF_i \cdot DEPFACT_i \cdot BR \cdot DECAY_i \cdot RF_i \cdot T \quad , \qquad (14)$$

where

D_i = the thyroid dose from the i^{th} radionuclide (rem);

X_i = the (uncorrected) concentration of the i^{th} radionuclide at the receptor (Ci/m^3), as calculated in Eqs. (8)--(12);

DILUTE = the dilution factor for the building that addresses the transport path from the core to the point of release;

DCF_i = the thyroid dose conversion factor of the i^{th} radionuclide (rem/Ci inhaled);

$DEPFACT_i$ = the dry deposition correction factor of the i^{th} radionuclide;

BR = the breathing rate (3.54×10^{-04} m^3/s);

$DECAY_i$ = the decay correction factor of the i^{th} radionuclide as a result of holdup;

RF_i = the release fraction of the i^{th} radionuclide; and

T = the time the receptor is exposed to the plume (s).

(U) 6.5 Potential Exposures

(C)

(c)

(U) 7.0 CONCLUSIONS AND RECOMMENDATIONS

(C)

(U) 7.1 Limiting Factors in Release Scenarios

(C)

(C)

(C)

(U) 7.2 Assessment of the Risk to the Public

(C)

(C)

(U) 7.3 Conservatisms

(U) As noted in greater detail in Sec. 2.3, this study used conservatisms in potential core inventories, saboteur knowledge and ability, the various release conditions, and the potential receptor availability. It should be emphasized that this study makes no attempt to define or even estimate the size of the force required to attempt one of the described scenarios. Because of the complexity and time constraints imposed by many of these scenarios, the definition of the manpower available to the attack force may preclude their completion.

(U) 7.4 Core Melt Analysis

(c)

(U) 7.5 Recommendations

(U) The following measures could be considered to further mimimize the risk to the public if the probabiltiy of successful sabotage is considered high enough to warrant action.

(C)

(c)

(c)

(c)

(C)

(C)

(C)

(C)

(U) REFERENCES

1. D. C. Kocher, "Dose Rate Conversion Factors for External Exposures to Photon and Electron Radiation from Radionuclide Occuring in Routine Releases from Nuclear Fuel Cycle Facilities," Health Physics 38, 543 (1980).

2. "Limits for Intake of Radionuclides by Workers," International Committee on Radiological Protection Publication 30 (ICRP-30) (Pergamon Press, New York).

3. T. R. England, W. B. Wilson, and M. G. Stamatelos, "Fission Product Data Set for Thermal Reactors, Part 1: A Data Set for EPRI-CINDER using ENDF/D -IV," Los Alamos Scientific Laboratory report LA-6745-MS (December 1976).

4. R. J. Jiacoletti, G. E. Cort, A. M. Gage, J. M. Graf, and L. J. Walker, "Consequences of Sabotage of Nonpower Reactors ," Los Alamos Scientific Laboratory report LA-7845-MS (Confidential) (June 1979).

5. R. R. Burns, Ed., Research, Training, Test and Production Reactor Directory, Vols. I and II (American Nuclear Society, La Grange Park, Illinois, 1980).

6. "Final Safety Analysis Report on the National Bureau of Standards Reactor, NBSR 9," National Bureau of Standards report 8998, Supplement A (October 1966), Supplement B (December 1966), and Addendum 1 (November 1980).

7. W. W. Graham III and D. M. Walker, "Final Safety Analysis Report for the 5 MW Georgia Technology Research Reactor," Georgia Institute of Technology report GT-NE-7 (December 1967).

8. A. H. Emmons, D. G. Fitzgerald, and E. L. Cox, "University of Missouri Research Reactor Facility Hazards Summary Report, University of Missouri" (March 1982).

9. "Safety Analysis Report for the MIT Research Reactor (MITR-II), Massachusetts Institute of Technology report MITNE-115, (October 1970).

10. "Safety Evaluation by the Division of Reactor Licensing in the Matter of Docket No. 50-193, Rhode Island and Providence Plantations," US Atomic Energy Commission (June 1968).

11. "Safety Analysis Report, Revision II," Western New York Nuclear Research Center, Inc. (September 1963).

12. "Safety Analysis Report for the Union Carbide Research Reactor," Union Carbide Corporation (May 1980).

13. "Safety Analysis Report, Ford Nuclear Reactor, Michigan Memorial Phoenix PRoject," Univerty of Michigan (November 1984).

14. J. L. Meem, "Revised Safety Analysis Report in Support of Amendment to License R-66 for Two Megawatt Operation University of Virginia Reactor," (Revision 1, January 1970 and Revision 2, April 1981).

15. J. F. Wett, Jr., "Surface Temperatures of Irradiated ORR Fuel Elements Cooled in Stagnant Air," Oak Ridge National Laboratory report ORNL-2892 (April 6, 1960).

16. "Explosives and Demolitions," United States Army Field Manual FM 5-25.

17. E. E. Lewis, Nuclear Power Reactor Safety (John Wiley and Sons, New York, 1977).

18. J. O. Blomeke and M. F. Todd, "Uranium-235 Fission Product Production as a Function of Thermal Neutron Flux, Irradiation Time, and Decay Time," Oak Ridge National Laboratory report ORNL-2127, Part 1, Vols. 1 and 2 (November 1958).

19. E. Walker, "A Summary of Parameters Affecting the Release and Transport of Radioactive Material for an Unplanned Incident," Battelle Northwest Laboratories report BNLD-81-2 (September 1978) (Reissued August 1981).

20. "Preliminary Hazards Summary Report, Appendix II, Meteorological Summary for the Department of Commerce, National Bureau of Standards Reactor" (January 1959).

21. R. P. Hosker, Jr., "Methods for Estimating Wake Flow and Effluent Dispersion Near Simple Block-Like Buildings," NOAA technical memorandum ERL/ARL-108 (May 1981).

22. W. G. Snell and R. W. Jubach, "Technical Basis for Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Nuclear Regulatory Commission report NUREG/CR-2260 (October 1981).

23. David H. Slade, "Meteorology and Atomic Energy 1968," Atomic Energy Commission report TID 24190 (July 1968).

24. D. Bruce Turner, "Workbook of Atmoshperic Dispersion Estimates," Department of Health, Education, and Welfare publication 999-AP-26 (May 1970).

25. "Rancho Seco Building Wake Effects on Atmospheric Diffusion," NOAA technical memorandum ERL ARL-69 (November 1977).

26. "Accuracy of Dispersion Models - A Position Paper of the AMS 1977 Committee on Atmospheric Turbulence and Diffusion," Bulletin of the American Meteorological Society 59(8) (August 1978).

APPENDIX [redacted]

ASSUMPTIONS USED IN THE NONPOWER REACTOR SABOTAGE STUDY

(U) I. ASSUMPTIONS USED IN GENERATING SCENARIOS

(C) 1.

(U) 2. The release is assumed to occur immediately before a shutdown for refueling for an equilibrium cycle. Again, this leads to the largest possible source term and therefore is appropriately conservative for any other point in the cycle.

(U) 3. The accident analyses presented in the facilities' Safety Analysis Reports (Hazard Summary Reports) are assumed to be acceptable and therefore need not be reanalyzed. This information already has been reviewed and approved by the NRC as appropriately conservative.

(U) 4. It is assumed that the adversary has available to him all information available in the open literature. Because all information about the facilities used in the analyses was from unclassified sources (including site tours and discussions with facility personnel), the adversary can duplicate any scenario that will be presented in the report.

(U) 5. No credit is given to the scenario for random failures. The adversary is credited with a certain level of intelligence and will plan for all events necessary to complete the scenario. It is not logical to assume that a random failure will occur at the precise moment necessary to insure a successful scenario nor is it logical to assume a failure occurrence that prevents completion of the scenario.

II. ASSUMPTIONS USED IN SOURCE-TERM CALCULATIONS [redacted]

(C)

(U) For the building wake cavity releases, the following are applicable.

(U) 1. The model conservatively assumed that homogeneous mixing occurred instantaneously.

(U) 2. The release was transported through a single room whose volume was equivalent to the sum of all the individual volumes in the transport path (from the reactor to the building wake cavity). No leakage that might occur through each transport path was allowed.

(U) 3. The leak rate out of the building was assumed to be based on the building leak rate given in the Safety Analysis Report for each facility.

(U) For the stack level releases, the following was assumed.

(U) 1. The radionuclide cloud was assumed to be released into the nearest volumes for transport through the stack (that is, the shortest path).

(U) 2. The stack flow rates were based on the values found in the Safety Analysis Report for each facility.

(U) 3. No reduction in activity based on any filtration was allowed.

(U) III. ASSUMPTIONS USED IN THE TRANSPORT AND DISPERSION CALCULATIONS

(U) Three models were used to estimate the atmospheric dispersion from a sabotage-induced release for these nonpower research reactors. The models included a cavity model[2] used for determining the dispersion characteristics for exposures within the building wake cavity, the NRC 1.145 atmospheric dispersion model[3] for determining exposures for release heights less than 2.5 times the building height, and a Gaussian plume model.[4,5]

(C)

(U) The following assumptions are applicable to all calculations.

(U) 1. Stable weather conditions were assumed. For the Gaussian plume and NRC models, Pasquill category type E and F (stable) could be used for most facilities. This condition occurs only at night and usually only in rural settings. Because type F conditions were so infrequent at NBS, only type E conditions were considered. The criterion used in this determination was that the condition should exist at least 5% of the time. These stability classes, in conjunction with the appropriate windspeeds, were assumed because they resulted in the maximum potential exposures.

(U) 2. Appropriate wind speeds for the above types of weather conditions were taken from Slade.[4] For type E stability conditions, the wind speed used was 3 m/s, and for type F stability conditions, the wind speed used was 1 m/s.

(U) 3. The above weather conditions were assumed to persist for the entire release period so that the concentrations at the location of the receptor would be maximized. The short durations of the release scenarios (a few hours) increases the likelihood that the stable meteorological conditions can persist throughout the exposure time time, thus maximizing the dose commitment to the receptor.

(C) 4.

(C) 5.

(U) 6. For the Gaussian plume and NRC models, the Pasquill-Gifford curves were used for estimating the standard deviation of the distribution

of material in the cloud. These curves are used widely and were taken from actual diffusion experiment results for distances of less than 1 km in an open field.

(U) 7. The receptor was assumed to be located in the center line of the downwind direction. This assumption is the most conservative because any slight shift in wind direction would result in reduced concentrations and subsequently lower exposures to the receptor. The maximum exposures are received by a receptor located directly downwind and in the center-line path of the radionuclide cloud.

(U) 8. The particulates were assumed to be less than 10 μm in diameter. This size was assumed to ensure that they are in the respirable range.

(U) 9. The release fractions of the noble gases, halogens, and particulates from the core melt were assumed to be 100%, 50%, and 1%, respectively, and an additional 50% reduction of the iodines was assumed to account for removal of the airborne iodine through various physical phenomena (that is, adsorption, adherence, and so on).[6] This constitutes a release of ~15% of the gross fission product activity.[6] These are very conservative assumptions that may lead to doses that are much too high in some cases; however, they currently are accepted fractions.

(U) 10. The thyroid dose calculations conservatively assumed that all of the material inhaled at the receptor location was respirable. The external dose calculation assumed that the cloud exposing the receptor is semi-infinite. This assumption produces the maximum exposures.

(U) 11. No credit was given for any filtration.

(U) 12. The stack plume releases were assumed to possess no driving forces other than momentum, and therefore, the plumes were assumed not to have risen significantly.

(U) IV. ASSUMPTIONS USED IN THE CAVITY MODEL

(U) The following assumptions are applicable only to the cavity model used in the calculations.

(U) 1. The release was entrained entirely into the building wake cavity. This was the most conservative case because none of the radionuclide material initially was released from the cavity. This assumption produced the maximum dose.

(U) 2. As a result of the turbulence inside the cavity, the material mixes fairly rapidly. Realistically, there will be some nonuniformities in the concentration of radionuclides within the building wake cavity; however, the model estimates an average concentration of radionuclides within the cavity. The receptor was assumed to be located inside the building wake cavity; however, because it would be impossible to predict exactly where, he was not assumed to be situated directly at the release point. This assumption was assumed to be realistic and not extremely conservative.

(U) 3. The constant value C in the cavity model equation can have a value between 0.5 and 5.0. It was assumed conservatively that the value was between 0.5 and 1.0, which were the values used in this calculation because higher numbers would indicate more rapid dispersion and therefore lower doses.

(U) V. ASSUMPTIONS USED IN THE NRC 1.145 MODEL

(U) The following assumptions are specific to the NRC model used in the calculations.

(U) 1. The release height was assumed to be less than 2.5 times the height of the nearest building. This assumption was required for the equations used in the NRC model calculation to be directly applicable.

(U) 2. The three equations used in the calculation incorporate the dilution caused by the building wake effect and also the meander effect that results during stable weather conditions and low wind speeds.

(U) VI. ASSUMPTIONS USED IN THE GAUSSIAN PLUME MODEL

(U) The following assumptions were made specifically with regard to the Gaussian plume model.

(4) 1. When making a dry deposition correction, the average deposition velocity of the particulates was assumed conservatively to be 0.003 m/s. This produced the highest exposure.

(4) 2. The buoyancy correction fractor was obtained by assuming that the release was not heated. This was a conservative assumption as buoyancy would decrease the ground level concentrations near the release.

REFERENCES

1. E. E. Lewis, Nuclear Power Reactor Safety (John Wiley & Sons, New York, 1977).

2. R. P. Hosker, Jr., "Methods for Estimating Wake Flow and Effluent Dispersion Near Simple Block-Like Buildings," NOAA technical memorandum ERL/ARL-108 (May 1981).

3. W. G. Snell and P. W. Jubach, "Technical Basis for Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Nuclear Regulatory Commission report NUREG/CR-2260 (October 1981).

4. David H. Slade, "Meteorology and Atomic Energy 1968," Atomic Energy Commission report TID 24190 (July 1968).

5. D. Bruce Turner, "Workbook of Atmoshperic Dispersion Estimates," Department of Health, Education, and Welfare publication 999-AP-26 (May 1970).

6. J. O. Blomeke and M. F. Todd, "Uranium-235 Fission Product Production as a Function of Thermal Neutron Flux, Irradiation Time, and Decay Time," Oak Ridge National Laboaratory report ORNL-2127, Part 1, Vols. 1 and 2 (November 1958).

Nuclear Proliferation: Studies and Strategies for Stopping the Spread of the Bomb

NUCLEAR CONTROL INSTITUTE

1000 Connecticut Avenue, N.W., Suite 704, Washington, D.C. 20036 (202) 822-8444

August 31, 1988

FREEDOM OF INFORMATION ACT REQUEST

FOIA-88-451
DocID 9-2-88

Linda Robinson
Chief, FOIA/LPDR Branch
U.S. Nuclear Regulatory Commission
Washington, DC 20555

FREEDOM OF INFORMATION REQUEST

Dear Ms. Robinson:

The following is a request being made pursuant to the Freedom of Information Act (5 USC 552) on behalf of the Nuclear Control Institute. NCI is a non-profit, public-interest, research institute that provides public education on the subject of nuclear terrorism and nuclear proliferation.

We request you furnish the report, "Nonpower Reactor Sabotage Study," LA-CP-87-02, dated January 1987. This report is referred to in a Commission correspondence to the Buffalo office of the Federal Bureau of Investigation, dated August 2, 1988. In addition, please furnish all letters, memoranda, Commission papers and other documents related to the origin, preparation, and evaluation of this report.

In accordance with the provisions of the Freedom of Information Act, we expect a response to this request within 10 working days. As a non-profit, public-interest organization, we believe that furnishing the records requested can be considered as primarily educational and benefitting the public as opposed to the requestor; consequently, we feel that any applicable fees for the search or reproduction should be waived as permitted under the Act. Should you decline to waive all fees, we ask that you obtain our authorization before incurring search costs in excess of $100.

We believe that disclosure of this information is in the public interest because it bears directly on the Commission's mandate to promote, defend, and secure "the common defense and security" and to protect "the public health and safety" of the United States as required by the Atomic Energy Act of 1954, as amended.

If you determine that some or all of the records requested are exempt from release, we would appreciate your stating which exemptions you believe cover the material you are not releasing.

8811030434 (2pp)

Further, if you determine that some portions of the requested records are exempt, we ask that you make available the remainder of the records, to the extent that the records determined to be exempt are "reasonably segregable" as provided in the Act.

In order to expedite the release of the documents in question, we ask that you release them as they become available to you, without waiting until all the documents have been assembled.

Thank you.

Sincerely,

Alan J. Kuperman

Alan J. Kuperman
Issues Director

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of THE REGENTS OF THE UNIVERSITY OF CALIFORNIA (UCLA Research Reactor)	Docket No. 50-142 (Proposed Renewal of Facility License)

DECLARATION OF DR. JAMES C. WARF

I, James C. Warf, declare as follows:

1. I am Professor of Chemistry at the University of Southern California (USC), where I have been a member of the faculty for the last thirty-four years. Prior to that time, I spent five years with the Manhattan Project, mostly at Ames Iowa, and to a lesser degree at the University of Chicago and at Oak Ridge, Tennessee. I specialized in the chemistry of nuclear materials and was Group Leader of the Analytical Section and, part of the time, the Inorganic Section, at times with seventy people working under me. Directly after World War II, I played a role in the formation of the Federation of Atomic Scientists (later Federation of American Scientists). Nearly thirty years ago I helped found the Los Angeles Chapter of the Federation of American Scientists, which later became the Los Angeles Federation of Scientists and, most recently, the Southern California Federation of Scientists. I remain active with the organization to this day. A more detailed statement of professional qualifications is attached hereto.

2. I have reviewed certain documents related to the UCLA Argonaut reactor. These documents have included: (1) "Analysis of Credible Accidents for Argonaut Reactors" by S. C. Hawley, et al, particularly those sections dealing with explosive chemical reactions and graphite fire, (2) a draft analysis by David DuPont of the Wigner energy section of the Hawley report, supra, (3) "Fuel Temperatures in an Argonaut Reactor Core Following a Hypothetical Design Basis Accident (DBA)" by G.E. Cort, and (4) the fire response section of the March 1982 Emergency Response Plan for the UCLA Reactor, specifically the Los Angeles Fire Department fire response plan attached thereto as "Attachment A." Certain other relevant documents, identified below, have also been reviewed.

3. It is my understanding that, having operated for roughly twenty years, the UCLA Argonaut reactor is currently the subject of a safety review by the U.S. Nuclear Regulatory Commission as part of a license renewal proceeding. Such a review seems to me to be a sensible precaution, as occasionally some significant fact or facts, overlooked in an original analysis decades before, may be uncovered. And if nothing significant is found, a greater level of assurance of safety has been established. Thus, in my opinion, it would be prudent for such a safety review to take into account the following facts:

4. The original Hazards Analysis for the UCLA reactor dismissed the probability of damage from fire resulting in the release of fission products as "very small" in part because "none of the materials of construction of the reactor are inflammable." (1960 UCLA Reactor Hazards Analysis, p. 62, "Fire"). While other factors may affect the probability of fission product release from fire, the statement that none of the materials of construction of the reactor are inflammable is simply incorrect. A number of those materials--particularly the graphite, uranium, magnesium, and even the aluminum, among others--are, under the right conditions, most definitely combustible.

5. The first and most obvious of the combustible materials used in the Argonaut reactor is the graphite--used as moderator, reflector, and thermal column. Graphite will, under the right circumstances, most definitely burn, as the Hawley report correctly indicates. (Charcoal is, after all, a graphitic substance, and it will, of course, readily burn.)

6. On page 82 of the Proceedings of the 1958 Atomic Energy Commission and Contractor Safety and Fire Protection Conference, held at AEC Headquarters in Germantown, Maryland, June 24-25, 1958, held in part to analyze the implications for reactor safety of the Windscale accident in which the graphite moderator and the uranium fuel both caught fire, Dr. C. Rogers McCullough of the USAEC is quoted as saying:

> By the way, this is an amusing point. The belief had grown up on the part of many people in this country that graphite will not burn. This is nonsense. Graphite is carbon, and anyone knows that carbon will burn if you get it hot enough. But this glib remark, that graphite will not catch on fire, had become prevalent.

While not having personal knowledge of any widespread belief in this country that graphite could not burn, I concur with Dr. McCullough's statement that it, of course, can burn in air, as the Windscale fire unfortunately so clearly demonstrated. A belief to the contrary would be neither correct nor prudent.

7. As to the matter of the ignition temperature of graphite, it is dependent upon a number of factors such as the purity and density of the graphite, the amount of air present and the velocity of the air, the particle size and surface-to-volume ratio of the graphite, and structural configuration influencing heat loss. Furthermore, there appear to be other uncertainties, as evidenced by Dr. McCullough's comments at the same page of the above-cited

proceeding:

> Research work is going on; we are not satisfied that we know the ignition point of graphite. . . . At any rate, research is going on to learn more about the ignition temperature. It is a tough problem to solve, and we are exploring possibilities.

Thus, there are some uncertainties as to ignition temperature of graphite, and it might be wise from the point of view of a conservative safety analysis to place or establish the magnitude of error on whatever estimate of ignition temperature is used. However, I am not prepared at this time to suggest what error limits might be appropriate for any specific estimate of ignition temperature.

8. The Hawley report uses a figure of 650° C as the point at which graphite will burn readily if sufficient oxygen is supplied. That figure seems to me to be reasonable for reactor-grade graphite, although as I indicated in 7 above there are some uncertainties and some error limits might be appropriate. Any temperature estimate is valid only for a fixed set of parameters (density, purity, particle size, air supply, etc.)

9. Once ignited, self-sustained combustion of the graphite must be assumed if the air supply is adequate. Although this depends upon configuration, airflow, and the like, it appears to me that somewhere around 650°C is the critical temperature for induction of a self-sustained fire in the Argonaut reactor's graphite. This temperature is above a glowing red heat but below a white heat. The reaction is exothermic, so if some of the graphite were ignited, it could release enough heat to bring other graphite to the ignition temperature.

10. In addition to graphite, I understand the Argonaut reactor at UCLA employs metallic uranium in a uranium-aluminum eutectic, clad with aluminum. Metallic uranium readily burns in air if ignited, and under somewhat more restrictive conditions, so can aluminum. Aluminum gives off more heat, pound for pound, than uranium metal when burned, but it is somewhat more resistant to burning. The fact that the uranium and the aluminum are in a eutectic will not affect the ability of either to burn, although burning of the eutectic will give off slightly less heat than if the materials were not in a eutectic. However, the difference is insignificant. In addition, the fact that the eutectic melts at a relatively low temperature (640°C--Hawley, p. 18), will not affect the ability of the materials to burn. The metals can burn as well in a liquid form as a solid. In fact, molten metal can cause fresh aluminum, without the normal protective oxide layer, to be exposed to air, making burning far more likely.

11. As to ignition temperature for uranium metal, again there are some uncertainties. Charles Russell (Reactor Safeguards, Pergamon Press, Oxford, 1962, p. 115-116, citing W.C. Reynolds, Report NACA TN D-182, "Investigation of Ignition Temperatures of Metals") gives the ignition temperature of solid uranium metal in oxygen at 1 atmosphere as 608° F (320° C). Yemel'yanov and Yevstyukhin (The Metallurgy of Nuclear Fuel,

Pergamon Press, Oxford, 1969) state, "At a temperature above 700°C solid compact uranium burns in air and in oxygen emitting a blinding white light. Here uranium mixed oxide is formed according to the reaction

$$3U + 4\ O_2 \xrightarrow{>700^\circ C} U_3O_8 + Q$$

where Q = 845.2 kcal/mol." Turnings of reactor-grade uranium have ignited when being cut using a lathe, evidently from friction. Finely divided uranium ignites in air at room temperature. Thus the ignition temperature is a variable, depending on circumstances, but in general uranium metal must be considered more combustible than graphite. I have had no experience with uranium-aluminum eutectic, but the combustibility of the alloy certainly merits investigation, in both solid and liquid states.

12. It is my understanding that the control blades at the UCLA reactor are cadmium-tipped and protected by magnesium shrouds. Magnesium can also burn, and when it does so it gives off considerable energy. The ignition temperature of Mg metal is variable, depending on its particle size, etc. If you specify an ignition temperature you want, from 25° up, I can prepare a specimen which will ignite at that temperature. One should be aware that slow oxidation occurs below ignition temperature.

Cadmium metal is a low-melting metal with a relatively high vapor pressure. The Handbook of Chemistry and Physics reports its melting temperature as 320°C. If the control blades are made of the metal and not the oxide, it would thus seem prudent to analyze the reactivity and other possible consequences of an incident which resulted in the melting of the control blades. Furthermore, the volatility of cadmium could potentially result in cadmium vapor being released in a fire or other incident involving elevated temperatures. If so, the cadmium vapor or its oxide would likely rapidly condense in air as minute particles and could cause a potential hazard for fire-fighters or others due to the toxic nature of cadmium. This, too, should probably be considered, it would seem to me, in designing fire-fighting plans and analyzing potential accident sequences and consequences.

13. I also understand that UCLA is requesting a license for 2 curies of plutonium-239 in a plutonium-beryllium neutron source for the reactor facility. Were this Pu-Be source to become involved in fire, the consequences could verge on the catastrophic. Plutonium metal, of course, can burn, releasing minute particles into the air, dispersed by the energy of the fire. Fire-fighting would be extremely hazardous due to the presence of the plutonium oxide in the air, and the public health implications would be awful. (2 curies of Pu-239 is by no means an insignificant amount; placed near the skin, it will cause radiation burns in a few minutes; inhalation of even microgram amounts is exceedingly dangerous).

When Pu metal burns, it goes to PuO_2 in limited air, to Pu_3O_8 in excess air, just like uranium. Be is comparable to Al in its combustion, but is higher melting. Again, the chemical form of the material is important, i.e. whether in metal or oxide. BeO is volatile in steam at high temperatures.

14. The issue of how to fight a graphite-uranium fire, leaving aside the possibility of cadmium and plutonium particles being released, has no easy answers and would require considerable prior analysis of the problems inherent and preparation in advance in the form of emergency planning. There could be great danger, in particular, in employing either water or, to a lesser degree, carbon dioxide to put out the fire. In either case, an explosion might occur, owing to the formation of combustible gases.

15. Dr. McCullough's report on the Windscale incident, in the AEC document referred to above, describes how those fighting the fire tried various methods over a couple of days to put the fire out, which involved both uranium and graphite, all to no avail, and how they had to try, as a last resort, water:

> Now they were faced with the decision either to use water or to let the fire burn up. They decided there was nothing left for them to do but put water in. There was some trepidation about this, as you can imagine, because they well knew that water on glowing uranium makes hydrogen. Water on glowing carbon makes hydrogen and CO; you have then a nice mixture of hydrogen, CO, and air, and you might have an explosion.
>
> But they had no other choice.

They, in the end, followed techniques learned during World War II in extinguishing incendiary bombs, and fortunately the gamble paid off. But they had no other choice, and righty were extremely worried about the potential for an explosion. The fact that one did not occur at Windscale, in my opinion, does not get one around the fact that such an explosion is clearly possible, could be quite dangerous, and that water should, if at all possible, not be used, or if used, used with the potential danger clearly thought out. As McCullough concluded:

> I think it took a great deal of courage on the part of these people to put water on this reactor. They did it with fear and trepidation, and in talking with them they will not guarantee that they could do it a second time without an explosion.

I note also that the steam that ensued carried with it very significant quantities of fission products into the environment.

16. The potential for metal-water or metal-steam reactions should be examined in putting together fire-fighting plans. Aluminum, uranium, magnesium, and graphite all can react in a steam environment, producing large amounts of energy, liberating hydrogen which can cause explosion dangers. Russell indicates the Al-H_2O reaction liberates more than twice the energy of nitroglycerin, in calories per gram, and five times the energy of black powder; the magnesium-water reaction just slightly less than aluminum; and the U-H_2O reaction just somewhat less than black powder. (Al + NH_4NO_3 was used as a cheap explosive in Vietnam, "Daisy Cutter.")

17. I do not believe it likely that a group of firefighters arriving on the scene would have the competence to judge whether to use water, and if so, how, etc. Furthermore, it would seem most prudent for an emergency plan to have been considered in advance of the appropriate fire-fighting response, and for the requisite materials to be readily available for such fire-fighting. There are non-moderating materials that could be used to smother the fire that would not react explosively with burning core components; careful consideration should be given to the choice of these. My reading of the one-page fire-fighting plan included in the March 1982 emergency plan seems to me inadequate in these regards.

18. The use of CO_2 on such a fire could also be dangerous. Graphite is oxidized by CO_2, yielding carbon monoxide, which is also explosive in the presence of air.

19. Simple carbon tetrachloride extinguishers that formerly were used for lab fires have a host of problems asssociated with their use, notably the toxic phosgene they give off when used on fires. And even some chemical foams might have a favorable moderating effect that needs to be taken into account (this can be gotten around, perhaps, by the addition of boron-containing compounds to such foams).

20. Firefighters would also have to be prepared to deal with potentially toxic substances such as cadmium fumes in the air, and work in an environment possibly contaminated with fission products and perhaps plutonium. They would need good information as to what materials had been released in to the air and roughly in what concentrations, good detectors for those materials, and ability to read and interpret that information. They would need appropriate equipment to protect themselves from inhalation of the materials and from direct exposure.

21. As stated above, the one page plan by the LA Fire Department, in my opinion, does not adequately address the above potential problems. While one hopes that such an emergency never occurs, and trusts that adequate precautions will be taken to minimize any potential for such an emergency, an emergency plan must realistically deal with the conditions that could occur if such an emergency were to happen. The existing plans to control a reactor fire are, it seems to me, inadequate. A revised emergency response could profitably include the following: rapid determination of any radiation hazard, rapid evacuation of personnel, stockpiling of fire-fighting substances safe for reactor materials, and knowledge of access ports to the reactor. The fire-fighters should not have to locate and confer with any particular reactor personnel, who might not be available at once.

22. I understand that there is some question about positive temperature coefficients of reactivity for graphite. Such a positive effect has been known for a long time--certainly we in the Manhattan Project knew about it forty years ago.

23. As to the Wigner effect, the small size of the UCLA reactor does not necessarily mean that the amount of Wigner energy absorbed per gram of graphite is likewise small. In fact, were a large-sized reactor and UCLA's far smaller reactor to both produce 1 MW-day of energy, all other things being equal, the amount of Wigner energy absorbed in each gram of adjacent

graphite would be considerably greater in the UCLA reactor than in the larger reactor, for the simple reason that the larger reactor has far more graphite to absorb the same amount of energy, thus the energy absorption per gram of graphite is "diluted." All other things being equal, a large reactor with the same neutron flux as the UCLA reactor, run for the same length of time, would produce the same amount of energy absorbed per gram of graphite as the UCLA reactor. And it is the energy absorbed per gram of graphite that is the key to whether enough energy has been stored to bring any part of the graphite to ignition if enough air is present; and, given the proper configuration, one unit of graphite ignited could release enough heat to bring many additional units of graphite to the ignition point.

24. I have read the Hawley, et al, analysis of the Wigner energy matter, as well as Mr. DuPont's critique thereof. It appears to me that there is considerable disagreement as to how much Wigner energy can actually be absorbed, given operating limits, in the UCLA reactor. As I understand it, Mr. DuPont uses the same analytical method as Mr. Hawley, yet takes issue with some of the numerical values Mr. Hawley used in his calculations, particularly the neutron flux and number of MWD* of operation at UCLA and the appropriate cal/g absorption figure that should be used for exposures at low doses. It appears that, if Mr. Hawley's calculational method is correct and if Mr. DuPont's numerical values are the appropriate ones, the amount of Wigner energy that could be absorbed in the UCLA reactor's graphite would be roughly twenty times the amount Mr. Hawley indicates.

25. Mr. Hawley uses a neutron flux of 10^{12} n/cm^2-sec. Mr. DuPont uses 1.5×10^{12}, taken from the UCLA Application for Relicensing at page III/6-5. Mr. Hawley's report takes the value 0.5 cal/g per MWD/AT as the best value for the rate of energy storage in graphite irradiated at 30°C. Yet Nightingale (p. 345) states, "More-accurate values derived from measurements at very low exposures range from 0.6 to 1.0 cal/MWD/AT." Mr. DuPont further takes issue with the Hawley study conversion to energy storage rate at 50°C; graphing the Nightingale data for the change in the rate of energy storage with temperature, Mr. DuPont finds 5/6ths the energy stored at 50°C than at 30°C, whereas the Hawley report uses a smaller fraction. Finally, the Hawley study indicates 12 MWD to-date at UCLA; Mr. DuPont says the correct figure is 17 MWD, and if the reactor were to operate its licensed limit of 5% per year through the proposed license period (until the year 2000), an additional 37 MWD could be produced. These modifications of the Hawley study calculations by Mr. DuPont seem reasonable, and raise a substantial question as to how much Wigner energy might be absorbed in the UCLA graphite.

26. In addition, there are some uncertainties in making such calculations, as they rely on employing empirically derived data from various irradiation locations in a few reactors and then extrapolating to another reactor of a different kind and configuration. Plus, I understand there is some uncertainty as to the past irradiation history of the UCLA reactor's graphite--whether, for example, it might have been previously used in another reactor prior to the construction of the UCLA reactor. In light of the foregoing, I suggest removing some of the graphite from different

* MWD means megawatt-days.

locations in the UCLA reactor core and experimentally determining how much Wigner energy has indeed been absorbed to date in the graphite. This could be done by any of a variety of methods--calorimetric annealing, X-ray diffraction patterns, heat of combustion measurements. (I understand the UCLA reactor is occasionally used to color diamonds. If this effect is due to changes in the diamond's crystalline structure and not to impurities in the diamond, this would be further evidence of this reactor's capability of causing radiation damage in graphite, as graphite and diamond are the two crystalline forms of carbon and would react similarly to neutron bombardment. I also understand there is some question as to whether the UCLA graphite has exhibited some swelling or dimensional change; if this is confirmed, it would also be evidence of Wigner energy storage and would lend further reason to the possible usefulness of making actual measurements.)

27. Both the Hawley and the Cort studies examine certain accident scenarios that could, by themselves, cause substantial temperature rises in the UCLA reactor. In both cases--the Hawley analysis of power excursions and the Cort analysis of coolant restriction following earthquake--the temperature did not reach that of the melting of the fuel eutectic or cladding. However, if substantial Wigner energy were stored in the graphite, such an incident could, conceivably, release that energy and substantially raise the temperature that could be reached. In addition, some experimental materials in the reactor core may have ignition temperatures below the melting point of the fuel, in which case fire could be initiated even though the initiating temperature did not approach the fuel's critical temperature. Thus, the significance of possible flammable characteristics of the reactor core contents and the true amount of Wigner energy that could be absorbed during the license period may well have significance in a safety review.

28. Much work has been done on the attack of uranium ingots, clad in aluminum, through a pin hole. At elevated temperatures, air or water enters the pinhole, reacts, and the resulting oxide swells. This breaks more Al skin, and the process continues faster; but so far as I know oxidation is retarded so much the ignition temperature is not reached. Powdered uranium (from decomposition of UH_3) can react with liquid water and glow red, forming UO_2 and H_2. Massive U metal must be heated to react.

29. Uranium and aluminum can be separated chemically from their eutectic by any number of techniques. One method is to dissolve the eutectic in hydrochloric acid, and oxidize the uranium to uranyl ion using nitric acid. Addition of excess sodium hydroxide precipitates the uranium as sodium diuranate, but converts the aluminum to the soluble aluminate ion. Separation is effected by centrifuging. Alternatively, the uranyl nitrate can be extracted by ether or butyl phosphate, leaving the aluminum in the aqueous phase.

30. I might also add that as I read the Hawley, et al, analysis of "credible accidents" for Argonaut reactors, I had the impression that certain extremely unlikely scenarios were examined and then dismissed, with the conclusion then asserted that there are no serious credible accident scenarios

for Argonaut reactors, when scenarios more likely appeared not to have been analyzed whatsoever. Perhaps the above-described facts can be of use in a fuller review of potential accidents and consequences.

31. The above-cited facts might also be of use in mitigating consequences of or preventing accidents. For example, it might be prudent to consider use of a uranium oxide fuel, which would be far less susceptible to burning. A boron-based control blade might get around the low-melting temperature problem for the cadmium (if it is in the metallic form). Boron-based fire-fighting foams or other materials might ameliorate problems of using water alone. Sand or a silicate, as a clay, perhaps could be used to smother the fire. It might be best merely to close off the air supply mechanically, but the possibility that this might allow the reactor to overheat should be examined.

I cannot overstress consideration of the danger of using water on such a fire, should it ever occur. The use of water on such a fire could be disastrous. Careful emergency planning before such an event occurs should hopefully result in fire-fighters not having to face the terrible choices faced by those responding to Windscale.

I declare under penalty of perjury under the laws of the United States of America that the foregoing is true and correct to the best of my knowledge and belief.

James C. Warf

James C. Warf

Executed at Kota Kinabalu, Malaysia

This 17 th day of November, 1982.

DR. JAMES C. WARF
Professor of Chemistry
University of Southern California

Professional Qualifications

1939	B.S. degree in chemistry, University of Tulsa (Oklahoma)
1940-41	Research chemist, Phillips Petroleum Co.
1941-42	Instructor in chemistry, University of Tulsa
1942-47	Manhattan Project. Group leader of analytical section and, during the last years, of the inorganic chemistry section, mostly at Iowa State University, Ames, but frequently at the University of Chicago and other sites. Research on chemistry and analysis of nuclear materials, uranium, thorium, plutonium, fission products, etc.
1946	Ph.D. degree in inorganic chemistry, Iowa State University
1947-48	Guggenheim Fellow, inorganic chemistry, University of Berne, Switzerland
1948 to present	Assistant, Associate, Full Professor of Chemistry at the University of Southern California, Los Angeles, California. Research: rare earth metals, uranium, solid state, crystallography, thermodynamics, liquid ammonia solutions.
1957-59, 1962-64, 1974-76, 1978-79, 1982-	Visiting Professor of Chemistry at various universities in Indonesia or Malaysia
1969-70	Visiting Professor, Technical University of Vienna, Austria
1964-70	Consultant, Jet Propulsion Laboratory, Pasadena, California

Author of approximately 65 scientific papers or encyclopedia articles, 8 books on chemistry.