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SUMMARY REPORT ON THE HAZARDS
OF THE UTR TEST REACTOR

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I. SUMMARY

Advanced Technology Laboratories, a Division of American-Standard, proposes to install and operate a low-power UTR Test Reactor at its divisional headquarters in Mountain View, California. The UTR Test Reactor is a "Zero Power" prototype for American-Standard's line of University Teaching and Research Reactors which are offered for sale to domestic and foreign universities in ten watt (UTR-B), ten kilowatt (UTR-10) and 100-kilowatt (UTR-100) sizes.

The UTR Test Reactor will be used to obtain design and physics data, to test new operations and components and to train reactor operators for the various UTR models. The UTR Test Reactor will be installed in an existing reactor facility which has been licensed in the past for testing of UTR's.

This report presents a detailed description of the site, facilities, characteristics of the reactor and operations to be performed. Also presented is an evaluation of safety considerations and an analysis of potential hazards which might result from installation or operation of this reactor.

Based on satisfactory operation of similar prototype reactors in the past and on a detailed hazards analysis, it is concluded that the operation of the reactor at this site in the manner described herein will not constitute a potential hazard to property or persons in the surrounding area.

II. INTRODUCTION

The UTR Test Reactor described herein is a "zero power" prototype for the University Teaching and Research line of reactors offered for sale by Advanced Technology Laboratories to universities for teaching and research purposes.

The UTR is a heterogeneous, light-water-moderated, graphite-reflected, thermal reactor offered in three sizes: the 10-watt UTR-B, the 10-kilowatt UTR-10 and 100-kilowatt UTR-100. The basic UTR design has been developed from Argonne National Laboratory's "Argonaut" concept described in ANL-5647.* While retaining the basic features of the "Argonaut", design modifications have been introduced to enhance the operability and safety of the UTR as a teaching and research tool.

Low-power models of the UTR have been satisfactorily operated at Atomic Energy Commission sponsored exhibits at the Tokyo Trade Fair in Tokyo, Japan and "Atom's for Peace" expositions in Cairo, Egypt and Lahore, Pakistan. UTR-10 reactors are in operation at Virginia Polytechnic Institute, Iowa State University and the Australian Atomic Energy installation at Lucas Heights. These operations have proven the operability and safety of the reactor concept.

In addition, two UTR experiments have been performed in the ATL reactor test facility. The first was performed in a 1-watt prototype of the UTR-10, constructed and operated under license No. R-41 issued May 21, 1958. The second was installed and operated under license No. R-64 issued in March 1960 to establish physics and design data and train reactor operators for the 0.1-watt exhibition UTR for the AEC exhibits at Cairo and Lahore. The description and hazards analysis of these experiments were presented in hazards summary reports ASAE-16** and ASAE-16 (Rev. 1)†, which were submitted with the licence applications. The previous licenses R-41 and R-64 were for the specific facility and experiments described and each was cancelled when no further tests were anticipated.

The proposed UTR Test Reactor will be installed in the reactor building, using the concrete biological shield constructed for the two earlier tests. The reactor core, shield, instrumentation and process system will be essentially the same as for the first tests except that the control system and hazards analysis have been modified to cover a maximum excess reactivity of 1.5% (versus 0.5%). This will permit low-power tests to experimentally verify critical mass, control rod worths and rod imbalance effects for the largest of the UTR line, the 100-kilowatt UTR-100, which will be designed for a maximum excess reactivity of 1.5% to provide a higher flux research tool.

The UTR Test Reactor is, therefore, designed as a prototype for the entire line of UTR's and will be used for experimentation, component testing and operator training for future design and modification of the UTR-B, UTR-10 and UTR-100.

* D. H. Lennox and C. N. Keller, "Summary Report on the Hazards of the Argonaut Reactor," Argonne National Laboratory, ANL-5647, December 1956.

** "Summary Report on the Hazards of the UTR-1 Reactor," ATL, ASAE-16.

† "Summary Report on the Hazards of the Exhibition UTR," ATL, ASAE-16(Rev. 1).

The operating power will be limited to 15 watts, with the additional provision that the integrated power in any one day will not exceed 10 watt-hours.

Table I summarizes data on the UTR Test Reactor with 0.5% excess reactivity, which is representative of experiments conducted previously in the facility, and with 1.5% excess reactivity representative of planned experiments on the UTR-100.

TABLE I

SUMMARY OF UTR TEST REACTOR DATA

	Typical 4-Control-Rod Operation	Typical 5-Control-Rod Operation
Maximum Excess k_{eff}	0.5%	1.5%
Maximum Power level	15 watt	15 watt
Critical mass	3.1 kg	3.1 kg
Maximum loading all plates in core	3.3 kg U-235	3.3 kg U-235
Geometry	Two slabs separated by 18" of graphite	Two slabs separated by 18" of graphite
Moderator	Light water	Light water
Fuel		
No. of fuel element assemblies	12	12
No. of fuel plates per assembly	12	12
Maximum number of fuel plates	144	144
Fuel plate dimensions	3" x 25.94" x 0.110	3" x 25.94" x 0.110
U-235 loading per plate	23 gm	23 gm
Water gap	3/8"	3/8"
Enrichment	93.5% U-235	93.5% U-235
Cladding	Aluminum	Aluminum
Fuel-Containing Matrix	UA1 ₄ + A1	UA1 ₄ + A1
Control rods and drives		
Drive type	Window Shade	Window Shade
Number of regulating rods	1	1
Shim-safeties	1	4
Safety rods	2	0
Total number of rods	4	5
Rod Worth		
Regulating	~0.2% $\Delta k/k$	~0.2% $\Delta k/k$
Shim-safety	0.62% $\Delta k/k$	~1.4% $\Delta k/k$
Safety rod	0.62% $\Delta k/k$	

TABLE I (Continued)

	Typical 4-Control-Rod Operation	Typical 5-Control-Rod Operation
Rod Speed Characteristics		
Regulating Rod	27 in/min	27 in/min
Shim-safety and Safety Rods		
Withdrawal rate	6 in/min	3 in/min
Scram (Maximum time for complete insertion)	~ 0.5 sec	~ 0.5 sec
Control Elements		
Material	Boral	Boral
Size		
Shim-safety and Safeties	7" x 7" x 1/8"	9 1/8" x 23 5/8" x 1/8"
Regulating Rod	2" x 2" x 1/8"	4 1/2" x 4 1/2" x 1/8"
Reflector		
Material	Graphite	Graphite
Dimensions	44" x 56" x 48"	44" x 56" x 48"
Reactivity Effects		
1% in Fuel Content	0.29% $\Delta k/k$	0.29% $\Delta k/k$
Temperature coefficient of reactivity	- 0.0068% $\Delta k/k$ per °C	- 0.0068% $\Delta k/k$ per °C
Moderator void coefficient of reactivity	- 0.18% $\Delta k/k$ per %void	- 0.18% $\Delta k/k$ per % void
Reactivity Input Rates		
Regulating Rod	0.010% $\Delta k/k/sec.$ (max)	0.010% $\Delta k/k/sec.$ (max)
Shim-safety Rod	0.045% $\Delta k/k/sec.$ (max)	0.045% $\Delta k/k/sec.$ (max)
Safety Rods	0.045% $\Delta k/k/sec.$ (max)	0.045% $\Delta k/k/sec.$ (max)
Neutron Lifetime	1.35×10^{-4} sec	1.35×10^{-4} sec

III. FACILITY DESCRIPTION

A. Site

The UTR Test Reactor will be installed adjacent to ATL's plant on a 10-acre site in Mountain View, California. The Division's plant covers over 45,000 square feet of floor space occupied by administrative and engineering offices, laboratory and manufacturing facilities. Approximately 180 people will occupy the plant during normal working hours.

A plot plan for the site is shown on Drawing D-2867.* A detailed description of the relationship of the site to its environs is presented in Appendix B. The physical characteristics of the site including its hydrology, seismology and meteorology are also presented in Appendix B.

B. Reactor Building

The reactor will be housed in an existing 20' x 32' x 14' eave height steel frame reactor building. The building is supported on a reinforced concrete foundation and has corrugated steel walls and roof and a concrete floor sealed with Amercoat. A concrete curb around the base of the building will prevent run off of radioactive water. Gas-tight construction has not been provided. Entry to the building is provided by two sliding doors, which may be locked with a padlock. A stairway provides access to the top of the reactor. A one-ton electric hoist is available for removing the concrete closures from the reactor.

The building plan and the arrangement of the reactor, process system and control console within the building are shown on Drawing D-2866.

C. Reactor

The UTR Test Reactor will be used to (1) provide experimental verification of critical mass, temperature coefficient, void coefficient, control rod worth and imbalance effects and flux distribution; (2) test experimental facilities for the UTR line of reactors; and (3) provide reactor operator training. The facility consists of a heterogeneous, light-water moderated, graphite-reflected, thermal reactor core housed in an existing concrete shield complete with attendant control rods and drives, process system and instrumentation. The over-all assembly for this test reactor is shown on Drawing E-2868. A description of the various components and systems of the reactor is presented below.

1. Reactor Core

The reactor core consists of a 3'8" x 4'8" stack of graphite 4'0" high, in which are embedded two parallel core tanks spaced 24 inches on center. Six fuel elements each containing 12 fuel plates are positioned within each of the core tanks, which are filled with ordinary water of high purity. The water, serving as moderator and coolant, circulates through the fuel elements. No self-sustaining chain reaction is possible in this system unless the water is present. This arrangement provides an "external" graphite reflector and a high-flux experimental region between core tanks. An assembly of the core is shown on Drawing E-2868.

* Drawings referred to in this section are in Appendix A.

The reactor is controlled by varying the vertical position of neutron-absorbing control rods located in the reflector adjacent to the core tanks. Four or five control rods and drives will be employed, depending upon the excess reactivity installed for a given experiment. For an excess reactivity up to about 0.5%, four control rods positioned immediately adjacent to the outside edge of the core tanks will be provided: two safety rods, one shim-safety rod and one regulating rod. For an excess reactivity of 0.5-1.5%, five control rods will be employed including four shim-safety rods adjacent to the outside of the core tanks and one regulating rod between the core tanks. Reactor power is controlled by positioning the shim-safety and regulating rods in the reflector from the control console. When a "scram" condition occurs, the reactor is shut down by rapid injection of the safety and/or shim-safety rods into the reflector by spring and gravitational forces. Components of the core are described below.

a. Core Tanks

The core tanks are rectangular chambers 5-27/32" x 19-15/16" high, formed from 1/4-inch aluminum plate. Each tank is subdivided by 1/8-inch aluminum dividers into six fuel chambers. The tanks are open top and are subject to no pressure except the hydrostatic head of the moderator-coolant which is continuously recirculated through the core tanks. The coolant enters the top of each core tank through a 1/2-inch supply line. After traversing the fuel elements, the water flows out of the tank through a 1/2-inch line.

b. Fuel Elements

The UTR fuel elements, as shown in Drawing R3-D-139, are assemblies of 12 uranium-bearing flat plates spaced 3/8-inches apart. A spacer assembly at either end of each element assures rigid maintenance of plate spacing to close tolerances. A lifting eye provided at the upper end of the element is used for withdrawal and replacement of the element.

Details of a typical fuel plate are shown on Drawing R3-C-145. The full-load fuel plates contain 23 grams of U-235. The half load fuel plates contain 11.5 grams of U-235. In either case, the plate includes a 23-inch length of aluminum-clad fuel matrix.

The uranium, enriched to 90-percent U-235, is contained as UA_{14} solid solution in Al. This matrix is clad with aluminum, metallurgically bonded to the fuel region.

c. Reflector

The reflector is composed of 4" x 4" x 48" graphite blocks which are stacked vertically upon a base plate around the core tanks. This arrangement provides an "external" graphite reflector at least one foot thick in each lateral direction. The water in the core tanks provides a reflector directly above and below the core.

d. Control Rods

The reactor is controlled by adjusting the position of 1/8" thick boral plates within the graphite reflector adjacent to the core tanks. Shim control is

provided by manual positioning of the shim-safety rods either individually or in unison. Fine control is provided by manual or automatic positioning of the regulating rod. The size, number and sequencing of the various rods are dependent upon the amount of excess reactivity installed. Control rod sizes, speeds, worths and reactivity insertion rates are summarized in Table I for the 4- and 5-rod operations. Sequencing of the rods is discussed in paragraph 3, Instrumentation & Control System, below.

The operation of the shim-safety and safety control rod drives is analogous to that of a window shade. A stainless steel spring connects the boron plate with a motor-driven drum. A reversible, constant speed motor rotates the drum, thus winding the spring around the drum and withdrawing the control element. This action stores energy in the spring which, supplemented with gravity forces, drives the safety and/or shim-safety control rods into the reflector when a scram occurs. The rods are scrambled by decoupling the drum from the motor drive by means of a magnetic clutch. The regulating rod operates on the same principle but is not scrambled. This type of drive has been operating satisfactorily at several university installations for some time.

e. Neutron Source

A 1-curie plutonium-beryllium source with a strength of approximately 1.4×10^6 neutrons per second is installed in the reflector to provide neutrons for starting the chain reaction and to provide indication on the startup instruments.

f. Experimental Facilities

Four vertical graphite stringers are provided in the reflector region between the core tanks. One of the stringers is $3\frac{3}{4}'' \times 3\frac{3}{4}'' \times 48''$ long and is located at the center of the core. The other three are $1\frac{3}{16}'' \times 3\frac{3}{16}'' \times 48''$ long and are grouped around the central stringer. The stringers will normally be in place during operation.

2. Shield

The reactor shield, as shown on Drawing E-2868, provides a minimum of 2 feet 6 inches of ordinary concrete at the sides and 1 foot of ordinary concrete across the top. The radiation intensities which have been measured during past operations in this facility at a power level of one watt are summarized in Table II. Although at a power of 15 watts radiation intensities in many places will be in excess of 10 mr/hr, limitations upon operation will be such that, (1) a 15 watt peak power will not be exceeded, (2) in any one day the integrated power will not exceed 10 watt hours, and (3) personnel will not receive exposures in excess of those specified in 10 CFR 20.

3. Instrumentation and Control System

All indicators, controls, recorders and alarms required for control of the UTR Test Reactor are assembled in a control console designed for one-man operation.

The control instrumentation consists of six systems: 1) reactor-control instrumentation, 2) process instrumentation, 3) area radiation monitoring, 4) startup, 5) scram and 6) annunciator. These systems are shown schematically on Drawings D-2864, D-2865, D-370, D-2869 and D-2870 and are described below.

TABLE II
RADIATION SURVEY - UTR TEST REACTOR
(POWER LEVEL = 1 WATT)

Sampling Point Location		Film Badge Analysis (millirem/hr)			Survey Instrument Analysis (millirem/hr)		
Description of Point	Distance from UTR-1 Shield (ft)	Gamma	Fast Neutron	Total	Gamma	Fast Neutron	Total
Center of north side shield, 26" from floor	Contact w/shield	<5	0	<5	4.5	1.25	5.75
North - top center closure - chain hoist eye hook well	Contact w/shield	70	32	102	48	93.8	141.8
Center of top center closure	Contact w/shield	40	7	47	41	81	122
Conduit opening, north side of west shield	Contact w/conduit opening	5 (est.)	0	5 (est.)	0.8	0	0.8
South rod-control mechanism on west side	Contact w/cover 0.5	0	0	0	0.25	0	0.25
Front of reactor console	Contact w/console 6.5	0	3.5*	3.5	0.13	0	0.13
Stairway gate	Contact w/gate 3	0	0	0	0.20	0	0.20
Between center and west segments of top closure	Contact w/top shield	40	3.5	43.5	38	81	119
Reactor bldg. Film badge board	17.5	0	0	0	0.11	0	0.11
Center of west side of shield	Contact w/shield	0	3.5*	3.5	1.4	0	1.4
3" above top of north side shield	--	0	0	0	2	2	4
Cable outlet from conduit by console remote monitoring station	7	0	0	0	0.15	<1	<1.15

* These two readings should be viewed with some degree of skepticism, since they are the only two fast neutron film measurements which exceed the corresponding instrument measurements and since they were determined from a low fast neutron exposure and are therefore subject to large statistical variation.

a. Reactor-Control Instrumentation

This system provides a continuous indication of reactor flux from startup to operating power and includes the startup channel, intermediate channel and servo channel and rod-position indicators and controls. These channels are shown schematically on Drawing D-2864 for the four-control-rod operation and on Drawing D-2865 for the five-control-rod operation. The operation of these instrument channels is described below.

(1) Startup Channel

The startup channel utilizes a BF_3 proportional counter. The amplified output from the counter is displayed on a log-count-rate and period meter with a range of 10 to 10,000 cps and a period indication of -30 to ∞ to $+3$ seconds.

(2) Intermediate Channel

The intermediate channel employs a compensated ion chamber to drive a log-N and period amplifier. The flux is displayed on a meter graduated from 0.0001 to 300%. Period indication is also provided.

(2) Servo Channel

The servo channel uses a compensated ion chamber to measure flux. The output of the chamber is displayed on a micromicroammeter with a range of 10^{-11} to 10^{-3} ampere. At the option of the operator, a flux controller provided in this channel can be used to operate the regulating rod automatically to control reactor power. A permanent record of reactor power is provided by a strip-chart recorder with a range of 0 to 100%.

(4) Control Rod Position Indication and Control

The shim-safety and regulating rods are provided with manual control switches and dial position indicators. A light indicates full-down position of the shim-safety rods. A digital position indicator and switch are provided to accurately indicate position of any of these rods. For the four-control-rod operation, a manual control switch is provided for withdrawing the two safety rods, and lights indicate full-in or full-out position of these rods. For the five-control-rod operation, the four shim-safety rods may be operated either individually or in unison.

b. Process Instrumentation

Process instrumentation is shown schematically on Drawing D-370. A temperature controller and heater are provided to maintain moderator temperature above 75°F ; an interlock prevents startup until the moderator exceeds this temperature. A low temperature alarm is also provided. Core inlet and outlet temperatures and system flow rate are indicated locally in the process system cabinet.

c. Area Radiation Monitoring

Two area monitors are provided in the reactor room to measure gamma radiation levels. The monitors utilize the output of ion chambers and indicate gamma levels from 0.1 to 100 mrem/hr. High radiation levels are annunciated.

d. Startup

A startup panel that contains operating switches, indicating lights and interlocks is provided on the console. The startup sequence, shown schematically on Drawing D-2870 for the four-control-rod operation and on Drawing D-2869 for the five rod operation, insures that the major operating equipment is functioning before the reactor is brought to criticality, and that the reactor is made critical in a safe and prescribed sequence.

To place the reactor in operation the startup system requires that the operator initially turn on the console power. If the scram bus is continuous, the count rate greater than 10 cps and the moderator temperature 75°F or higher, the safety rods, if any, may then be withdrawn in sequence. The reactor may then be made critical by adjusting the position of the shim-safety and regulating rods.

Auxiliary interlocks are provided which prevent startup of the reactor until the top closures are in place and the gate at the bottom of the stairway is closed and locked.

e. Scram

Conditions that will cause reactor scram or annunciation are listed in Table III and are shown schematically on Drawings D-2870 and D-2869.

TABLE III

Conditions Causing Scram and Annunciation

<u>Condition</u>	<u>Scram</u>	<u>Annunciation</u>
Period less than 10 sec		X
Period less than 5 sec	X	
Safety amplifier (150% of rated power or equipment failure)	X	
Loss of power - intermediate chamber		X
Loss of power - servo chamber		X
Power recorder off scale		X
Servo deviation greater than 15%		X
Over tolerance - area monitor		X
Scram bypass - log N and period amplifier		X
Low moderator temperature		X
Manual (at console)	X	
Earthquake	X	
Closure or Stairway Gate open	X	

The safety channel consists of two uncompensated ion chambers supplying current to a safety amplifier. The current from each chamber is amplified and fed to a flux-level indication. Relay contacts in the Safety Amplifier open at a pre-set flux level and initiate scram.

The scram conditions shown on Drawings D-2870 and D-2869 and the high-flux scram signals from the safety amplifier operate series contacts in the power supply to the rectifier supplying the holding current for the electromagnetic clutches on the safety and/or shim-safety rods. When a scram signal is generated, these contacts are opened, cutting off the holding current to the electromagnetic clutches, and the reactor is scrammed. The safety amplifier is designed to fail safe and provides maximum operating reliability. Indicator lights provided on the console light when scram occurs to indicate the cause of the scram.

Fast scram and backup are provided for the period and high-flux scrams by contacts in the grid circuit of the magnet control tubes. If one of these contacts is opened by a scram signal, a strong negative bias is produced on the tube, and the reactor is scrammed by cutting off the holding circuit to the electromagnetic clutches.

Provision is also made at the console for manually scrambling one or all of the shim-safety rods.

f. Annunciator

An alarm system is provided to warn the operator of conditions that require attention but do not immediately imperil the reactor or operating personnel. These conditions are listed in Table II and are shown schematically on Drawings D-2870 and D-2869. The annunciator alarm signals are indicated by red lights on a panel in the console and by an audible horn.

4. Process System

The Piping and Instrument Diagram - Process System is shown on Drawing D-370. The process system is designed to maintain the purity and base temperature of the recirculating moderator-coolant. The moderator water is pumped from the bottom of the core tanks through an electric heater and/or deionizer and back to the core tanks.

D. Fuel Handling

At low power levels the activation of the fuel elements is negligible, and the fuel elements can be handled safely with tongs. A lifting eye is provided on each fuel element to facilitate handling (Drawing R3-D-139). The activated fuel elements will be stored in wells recessed in the reactor shield when they are removed from the reactor. The wells are equipped with locks.

E. Waste Disposal

The only continuous wastes produced in operation of the UTR Test Reactor are found in the coolant-moderator. These products are continuously removed from the moderator water by the deionizer. The deionizer has been designed to facilitate removal and replacement of the resin.

Prior to disposal of the moderator water, a sample will be withdrawn and its activity established. If the activity is above tolerance, it will be stored until its activity decays below tolerance and it can safely be drained to the sanitary sewer

in accordance with AEC Regulations. To prevent inadvertent drainage of water leakage, a 6-inch high containment curb is provided around the periphery of the building. The only building drain is located in the process pit and this drain is plugged when not in use. A sump pump is provided in the process pit and a pump-out connection in the primary system to transfer contaminated water to storage or shipping tanks, if required.

Negligible radioactive gaseous wastes will be generated during these experiments. Therefore, no stack or methods for sampling atmospheric discharges are provided in the design.

F. Security

The following measures are taken to protect the reactor against sabotage or tampering by the general public or unauthorized personnel. When the reactor is not in use, the required safeguard provisions will be in effect:

1. The ATL plant is surrounded by a 8-foot security fence and entry to the facility is restricted by security regulations. From 12 midnight to 8 a. m. and on weekends no one is present at the plant but access to the facility or reactor building is prevented by locked gates and doors.
2. The reactor building is provided with locked doors. A reactor in operation light outside the building warns unauthorized personnel when the reactor is in operation.
3. The console is provided with a locked power switch and only the reactor operator will have a key to this lock.
4. All excess fuel is stored in a combination locked fuel vault located in the reactor building.
5. Fuel elements removed from the reactor will be stored in the locked fuel storage container.
6. Access to the top of the reactor during operation is prevented by a locked gate at the bottom of the access stairway. An interlock prevents startup of the reactor with this gate open.
7. The electric hoist will be locked to prevent unauthorized removal of the concrete closures over the reactor core.

IV. OPERATIONAL PROCEDURES

A. General

The UTR Test Reactor will be used for determination of various reactor parameters, testing of components, demonstrations and training of personnel in reactor operation. Fuel loading procedures and typical experiments are described in Appendix C.

B. Organization and Responsibility

Responsibility for the operation of the reactor will be held by the Manager, Nuclear Development Laboratory. He will be advised on the radiological safety aspects by a health physicist. In the event of a disagreement or dispute between the two, the decision will be referred to the Division Technical Director or President. A fuel custodian will be appointed by the Manager, Nuclear Development Laboratory. It will be his responsibility to 1) insure that fuel additions to the reactor either during or after the initial loading do not exceed specified (see Appendix C) or safe amounts and 2) maintain fuel inventory records.

Actual reactor operation will be by or under the direct supervision of (to allow for operator training) an operator licensed in accordance with C. F. R. Title 10, Part 55, Chapter I.

Procedures and regulations will be reviewed periodically by management representatives responsible for the safety, health physics, and technical aspects of the facility.

C. Fuel Control

The fuel plates, until installed in the reactor, will be stored in boron-lined containers in a vault with a triple combination lock. Accurate inventory of that fuel in the reactor and in the vault will be maintained. Assembled fuel elements, which are temporarily removed from the reactor, will be stored in a locked fuel storage pit. During periods of reactor shutdown, the electric hoist, which is required to remove the reactor closures, will be locked. The reactor building will be locked when not in use and during non-working hours.

D. Records

A log of the operation will be maintained. Items to be noted in the log include time of start up, power changes, shutdown, power levels, personnel present during operation, experiments conducted and any unusual or non-routine event that occurs during operation. In addition, detailed check lists of procedures to be performed prior to and during start up, check lists for procedures after shutdown, and operating log sheets showing hourly readings of control instruments will be prepared and retained for record. Records of tests of scram and annunciator operation will be kept.

E. Radiation Safety

Operation will be so conducted that no personnel receive radiation exposures in excess of the limits specified in the Code of Federal Regulations, Title 10, Part 20. Operating personnel will be issued gamma and neutron sensitive film badges. Visitors, if any, will be required to wear gamma and neutron sensitive dosimeters. Frequent inspection

and survey of the reactor and procedures used will be made by the health physicist to determine the extent of hazards and to insure that safe operational procedures are in use. Film badges and analytical services will be provided by Radiation Detection Company of Palo Alto, California.

F. Limitations

1. Operators

Two people will be required to be present during operation; one of whom must be licensed to operate UTR reactors.

2. Safety Trips

One period scram trip and two power level scram trip circuits shall be active during operation. If the excess reactivity has been reduced to 0.25% or less (the minimum possible period is 10 seconds), the period trip is not required.

3. Power Level

A maximum power level of 15 watts [to be determined by irradiation of gold foils in the central stringer to determine neutron flux (for 1 watt $\phi = 1.2 \times 10^7$ neutrons/sec-cm²)] will not be exceeded. In any one day the integrated power shall not exceed 10 watt-hours.

4. Excess Reactivity

For the five-control-rod operation the excess reactivity will not exceed the sum of 0.2% (the nominal worth of the regulating rod) and the amount that can be held down by one shim-safety rod or 1.5%, whichever is least. For four control-rod operation the excess reactivity will be limited to 0.5%.

5. Reactor Period

All operations will be so conducted that the reactor period is not shorter than 20 seconds.

G. Experiments

Written procedures for experiments shall be prepared and submitted to the Manager, Nuclear Development Laboratory for assessment of potential hazards, if any, in compliance with the above operating procedures and limitations. This approval is required before the experiments are performed.

H. Prevention of Accidental Criticality

A fuel custodian will be appointed by the Manager, Nuclear Development Laboratory. It will be his responsibility to see that fuel and moderator are added to the reactor in such a manner or amount as to prevent accidental criticality or violation of operating procedures or limitations. In addition the three following rules will be adhered to:

1. Before fuel additions are made to the core, the core tanks will be filled to operating capacity with moderator.

2. During fuel additions the neutron detecting instrumentation in the console will be closely observed, with the requirement that the neutron level within the reactor achieve a steady state before the fuel element is disengaged from the handling tool.

3. Moderator additions, if required, will be made only after at least two, and for large additions all, fuel elements have been removed from the reactor core.

V. SAFETY EVALUATION

A. General

The UTR Test Reactor has been designed with safety as a principal design criterion. In evaluating the safety of the reactor, the following major items were considered:

1. The inherent safety of reactors of this type has been demonstrated by the excursion tests conducted on the BORAX and SPERT reactors. Operability and safety have been verified by successful operation of UTR's at several AEC exhibitions and at permanent university installations.
2. The operator is provided with adequate instrumentation and alarms at the control console to permit an analysis of the operation of the reactor and to warn him of malfunctioning of equipment. Automatic shutdown devices are provided to scram the reactor in the event unsafe conditions occur.
3. All controls and safety features are designed to fail safe in the event of a power failure or interruption of control signal.
4. Operating procedures have been developed to provide for safety of personnel during startup, operation, and maintenance of the reactor.
5. The consequences of an accident which might result from failure of the safety system have been analyzed and shown to cause only minor injury to operating personnel and no danger to the well-being of the general public.
6. A security system has been established to protect the reactor from a direct act of sabotage and from tampering by the general public.
7. The site, facilities, and operating procedures have been evaluated in terms of mechanical accidents, earthquakes, fires, floods, storms, strikes, and riots, and found to be adequate.

B. Inherent Safety of the UTR-1 Reactor

The UTR Test Reactor is an inherently safe reactor because it has:

1. A long neutron lifetime.
2. Low built-in excess reactivity.
3. A negative temperature coefficient of reactivity.
4. A negative void coefficient of reactivity.

Therefore, one can expect self-limitation of the reactor in the event of a power excursion similar to the self-limitations demonstrated experimentally on the SPERT and BORAX reactors.*

* "Experimental Investigations of Reactor Transients," IDO-16285, 1956.

"Experimental Investigation of the Self-Limitation of Power During Reactivity Transients in a Subcooled Water Moderated Reactor, BORAX I Experiments, AECD-3668, 1954.

C. Mechanical Accidents

1. Power Failure

In the event of loss of power to the console, a scram will be initiated by all safety and interlocked circuits and moderator flow will stop. The power failure will be indicated to the operator by the loss of all lights on the console.

2. Rupture of Core Tank or Piping

If rupture or failure of the core tank or piping occurs, the moderator-coolant will drain from the core tank, thus reducing k_{eff} by 30% which will shut the reactor down.

If a moderator leak occurs such that the graphite reflector is saturated, the reflector savings will decrease thus decreasing reactivity.

The normal activity of the moderator is low and will not result in overtolerance exposure to operating personnel if spilled in the area. Any spills will drain to the process pit and will be retained there until disposed of after decay or dilution to a safe level as established by Title 10, Code of Federal Regulations, Part 20.

3. Fuel-Element Rupture

If the integrity of a fuel element is lost through corrosion of the cladding, there is a possibility that fission products will be leached into the moderator. The absolute amounts of fission products likely to be released under these conditions are difficult to predict; however, the quantity will be small because of low power operation and because of the use of an Al-U dispersion-type matrix in the fuel element. The residual quantity of fission products in the moderator will be kept small by the deionizer.

Since the moderator is contained in a closed circulating system, no uncontrolled release of fission products will result.

The fuel-element rupture will be detected through periodic sampling of the moderator. The contaminated moderator and deionizer resins will be disposed of as radioactive waste in accordance with Title 10, Code of Federal Regulations, Part 20.

4. Instrumentation System Failures

The instrumentation system will be designed to fail safe in the following manner:

- a. The inputs to the scram system appear as discontinuities in a series relay bus. Any discontinuity initiates a scram.
- b. Tubes will be installed in a normally energized position.
- c. Relays will be installed in a normally energized position.

D. Earthquake

The Santa Clara Valley region in which the UTR Test Reactor will be installed is subjected to earthquakes. The frequency of high intensity shocks since 1836 is tabulated in Appendix B, along with all quakes occurring since June 1955.

The area has been classified as Zone 3 (major damage) as compiled by the U. S. Coast and Geodetic Survey and presented on the "Seismic Probability Map of the United States," SMC-76, revised 1958.

The UTR Test Reactor and related facilities have been designed to withstand lateral loads resulting from earthquake in accordance with the requirements of the Uniform Building Code, except that a coefficient C of 0.20 has been selected to provide for the greater hazards connected with structural failure. (Uniform Building Code calls for C = 0.10 for structures other than buildings.) The formulas used are as follows:

1. Total lateral force or base shear V_e

$$V_e = CW$$

where C = 0.2

W = total operating weights

2. Force at level X

$$F_x = CW_x$$

3. Allowable stress

Normal + 1/3

In addition, a shock operated switch is provided to automatically scram the reactor in the event of an earthquake.

E. Fire

The reactor site is served by the City of Mountain View's Fire Department. In the event of a fire, the reactor will be secured and the Fire Department summoned. The reactor building is of fire-proof construction and a fire of sufficient severity to cause melt down of the fuel elements is not considered likely.

F. Floods

There is no record of the reactor site having ever been flooded. If a flood does occur, the reactor will be secured and the fuel elements will be removed out of the flood area if necessary.

G. Strike

If a strike occurs, the safeguards set forth in Section III, Facility Description, will be placed in effect. The reactor building will be secured and locked.

H. Riot

The reactor site is served by the City of Mountain View's Police Department. If a riot occurs, the safeguards set forth in Section III, Facility Description, will be placed in effect and the police summoned to cope with the rioters. The reactor building will be secured and locked.

I. Storms

The reactor site is in a "mild weather" area. The reactor and related facilities are housed in a building to protect them from the weather. If a violent storm occurs that might imperil the reactor, the reactor will be secured and the fuel unloaded and stored in the fuel storage pit and the pit locked.

VI. MAXIMUM CREDIBLE ACCIDENT

The maximum credible accident is postulated to result from a combination of:

1. The sudden insertion of the total available excess reactivity (1.5%).
2. Failure of the automatic safety system.

The resulting energy release would be initiated by a self-limited 24 megawatt-second burst with an initial 15 millisecond period (see Appendix D). Subsequent behavior could vary from immediate termination due to the expulsion of water moderator from the core tanks to repeated power bursts of no greater severity or quasi-steady state operation. A more likely accident would be that due to a ramp insertion of the excess reactivity resulting in a less severe initial burst. In any event the total energy release of the excursion would not exceed 200 Mw-sec, which is the energy required to compensate the excess reactivity by heating all of the moderator to 100°C and evaporating the upper reflector (15 gallons).

The initial power burst would cause a maximum pressure in the core of 10 psi and a maximum fuel plate temperature of 275°C which is well below the melting point of aluminum. Melting of the fuel is therefore extremely improbable. Nevertheless the consequences of partial meltdown have been investigated and are discussed in Appendix E.

The foregoing analysis has been based upon experiments performed in the SPERT-I* reactor, particularly the B-16-40 core to which the UTR core bears the closest resemblance.

A 200 Mw-sec energy release would result in radiation doses at the control console of 7 Rem, at contact with the west face of the reactor shield of 77 Rem and at contact with the north shield face of 250 Rem. Radiation doses in the ATL buildings proper (~ 80 feet from the reactor at the closest point) would be 1 Rem or less. Radiation received by inhabitants of the surrounding area would be considerably less than 100 millirem.

* "Experimental Investigation of Reactor Transients," IDO-16285, 20 April 1956.

"Analysis of Self-Shutdown Behavior in the SPERT-I Reactor," IDO-16528, 23 July 1959.

APPENDIX A
DRAWINGS

<u>Drawing No.</u>	<u>Title</u>
D-2867	Plot Plan - UTR Test Reactor
D-2866	Floor Plan - UTR Test Reactor Building
E-2868	Assembly - UTR Test Reactor
R3-D-139	Assembly & Details - Fuel Element
R3-C-145	Assembly - Full Load Fuel Plate
D-2869	Diagram - Startup Scheme, Scram Bus & Annunicator - UTR Test Reactor - 5 Rod Operation
D-2870	Diagram - Startup Scheme, Scram Bus & Annunicator - UTR Test Reactor - 4 Rod Operation
D-2865	Diagram - Nuclear Instrumentation - UTR Test Reactor - 5 Rod Operation
D-2864	Diagram - Nuclear Instrumentation - UTR Test Reactor - 4 Rod Operation
D-370	Piping & Instrument Diagram - Process System

PLAN
(TOP CLOSURES REMOVED)

SECTION A-A

SECTION B-B

[illegible]

LIST OF MATERIALS PER 12 ASSEMBLIES			
ITEM NO.	DESCRIPTION	NO. REQD.	QTY. REQD.
1	SPACER	24	D16-B-16
2	PIN	24	D16-B-16
3	RETAINING RING	48	D16-B-16
4	FUEL PLATE (DNG R3-C-145)	144	D16-B-3
4	" (DNG R3-C-146)	6	D16-B-3
4	" (DNG R3-C-147)	72	D16-B-16
5	LIFTING EYE	12	D16-B-16

① SPACER
MATERIAL ALUMINUM
5052-T6, 6061-T6 OR
1100-H18

② PIN
MATERIAL ALUMINUM
5052-T6, 6061-T6
OR 1100-H18

④ FUEL PLATE
R3-C-145
OR
R3-C-146
OR
R3-C-147

ASSEMBLY

③ RETAINING RING
WALDES TRIARC #5100-25
STAINLESS STEEL

⑤ LIFTING EYE
MATERIAL ALUMINUM
6061-T6

1. TOLERANCES EXCEPT WHERE NOTED OTHERWISE:
FRACTIONAL - $\pm \frac{1}{32}$
DECIMAL - $\pm .002$

NOTES:

REFERENCE DRAWINGS	
FULL LOAD FUEL PLATE	R3-C-145
HALF LOAD FUEL PLATE	R3-C-146
DUMMY FUEL PLATE	R3-C-147
FUEL HANDLING TOOLS	R3-C-137

REVISIONS

REV.	DATE	BY	CHKD.
1	2-20-59	V.J.	
2	2-21-59	D.S.	
3	2-24-59	J.H.	

ASSEMBLY & DETAILS
FUEL ELEMENT
EXHIBITION UTR

ADVANCED Technology LABORATORIES
AMERICAN Standard MOUNTAIN VIEW, CALIFORNIA
R3-D-139-0

MATERIAL (PER SPEC. RED-101-1)
 ① - 1100 GRADE ALUMINUM
 ② - U+AL

- NOTES:
1. FUEL LOADING 23.0g U²³⁵ 90% OR GREATER ENRICHMENT IN METAL FORM.
 2. PLATE TO BE FLAT WITHIN 0.032"
 3. EDGE FINISH - AS MILLED
 4. PLATE TO BE FABRICATED IN ACCORDANCE WITH SPEC. RED-101-1

REFERENCE DRAWINGS		ASSEMBLY	
FUEL ELEMENT ASSY.	R3-D-139	DATE	2/20/59
		BY	2/24/59
		CHECKED	3-14-59
ADVANCED Technology LABORATORIES		FULL LOAD FUEL PLATE	
AMERICAN Standard MOUNTAIN VIEW, CALIFORNIA		EXHIBITION UTR-	
		R3-C-145-0

LEGEND

LINE NUMBERING IS AS FOLLOWS:

LINE SERVICE
LINE NUMBER

PW-1-1/2-A

MATERIAL CLASS
(SEE PIPING STANDARD)

LINE SIZE

PW-PRIMARY CIRCULATING WATER

SYMBOLS

VALVES

GLOBE VALVE

GATE VALVE

Y-STRAINER

INSTRUMENTS

LOCALLY
MOUNTED

BOARD
MOUNTED

INSTRUMENT LEGEND

FIRST LETTER PROCESS VARIABLE	SECOND LETTER INSTRUMENT FUNCTION	THIRD LETTER INSTRUMENT FUNCTION
T-TEMPERATURE	I-INDICATOR	
F-FLOW	C-CONTROL	
M-MANUAL	A-ALARM	

DESIGN NOTES

A. SYSTEM VOLUME

COMPONENT	VOLUME (GAL)
CORE TANK	40.0
DEIONIZER	3.7
PIPING	2.3
TOTAL	46.0

B. DESIGN DATA-

	CLEANUP	HEATING
FLOW	2 GPM	4 GPM
TEMP.	100°F (MAX)	180°F (MAX)
PUMP		
SUCTION	15 PSIG	0.5 PSIG
DISCHARGE	16.8 PSIG	127 PSIG
NPSH	20.6 FT	18.2 FT

C. FRICTIONAL LOSSES

COMPONENT	LOSS (PSI)
PIPING (INC. HEATER)	1.5
D-1	1.2
FILTER	1.2
THROTTLE VALVE	11.4
TOTAL	15.3

D. PURITY MODERATOR 0.5 PPM TOTAL SOLIDS
1 MMHO CONDUCTIVITY

E. DO NOT OPERATE DEIONIZER D-1 ABOVE 100°F

REFERENCE DRAWINGS

PLAN & ELEV. REACTOR ASSY	F-347
LAYOUT DETAILS-PROCESS SYSTEM PIPING	F-382

REVISIONS

1			
2			
3			
4			
5			

ADVANCED Technology LABORATORIES

AMERICAN Standard MOUNTAIN VIEW, CALIFORNIA

SCALE: NONE

DATE: 1-16-60

DRAWN: WELDON

CHECKED: RLC 2/16/60

APPROVED: [Signature]

PIPING & INSTRUMENT DIAGRAM

PROCESS SYSTEM

EXHIBITION QTR - MOD 2

13023 | D-370-0

APPENDIX B

SITE DATA

1. Make Up of Surrounding Area

Figure 1 shows the locations of the site in relation to the surrounding facilities which include several cities, Moffett Field Naval Air Station, isolated industrial establishments and large areas of agricultural land.

The population distribution, during normal working hours, within a one-mile radius from the site is shown in Figure 2. At the present time this population is concentrated to the north and west of the site, due to the residential area to the west and to the presence of Moffett Field, Fairchild and the Ames Aeronautical Laboratory to the north.

Figure 3 shows the location of various industries within a one-mile radius of the site. In general, these are small plants employing less than 20 employees, with the following exceptions:

a. Sylvania Microwave and Electronics Laboratories	1500 persons
b. Ferry-Morse Seed Company	175 "
c. Minton Lumber Company	100 "
d. Ames Aeronautical Laboratory	1440 "
e. Carl E. Anderson Building Materials	100 "
f. Rheem Semiconductor Corporation	900 "
g. Fairchild Semiconductor Corporation	1100 "
h. Lockheed Aircraft Corporation	1000 "
i. American-Standard	180 "

There are approximately 7000 people at Moffett Field. Approximately two thirds of these, or a total of 4600 people may be assumed to be on the base at a given time.

Figure 4 shows the present zoning within a one-mile radius of the site.

The locations of various utilities within a $\frac{1}{2}$ -mile radius are shown in Figure 5.

2. Meteorology

Figure 6 shows wind roses by months, prepared from the Summary of Monthly Aerological Records at Moffett Field, California.

Figure 7, wind roses by quarters at approximately 5,000 feet elevation were prepared from "Frequency Distribution of Winds" records at Oakland Airport, Oakland, California. Both sets of records were obtained from the National Weather Records Center, Ashville, North Carolina.

Table B-I is a summary of temperature inversions occurring, during each month for the past year, below an elevation of approximately 381 meters. This table was compiled from data obtained at the Oakland, California Airport, the nearest location at which observations for this type of information are taken. Conditions at the Oakland Airport are considered to be representative of the entire San Francisco Bay area.

3. Hydrology

The surface of the site is covered by about one foot of firm black clayey silt. This material is underlain by deposits of firm to stiff brown and grey-brown sandy and silty clays containing some lenses of medium dense grey-brown sand and clayey sands. Lenses of gravel are occasionally found in the above deposits.

The surface material is moderately permeable and it can be expected that a considerable amount of rain water will be absorbed before runoff begins. However, during intensive or prolonged rainfall, runoff from the site can be expected.

The water shed area surrounding the site for one-mile radius is very flat land having a slope of less than one percent to the north. All surface water from the site will be held on the easterly side of Whisman Road.

The average elevation of the site is 62 feet above mean lower water, U. S. G. S. datum.

The ground water table for Santa Clara Valley varies according to the seasons. In 1950 the average ground water elevation near the site was minus 16 feet below sea level or approximately 80 feet below the ground. In 1953 the ground water elevation for the same area was minus 40 feet below sea level or approximately 100 feet below the ground.

Studies by the State Water Resources Board as presented in their Bulletin No. 7 (1953) covering the Santa Clara Valley indicate that ground water at the site flows generally in a northerly direction.

There are approximately 40 wells located within a $\frac{1}{2}$ -mile radius of the site. Their locations are shown in Figure 5. The nearest well is located approximately 500-feet from the reactor. In general these wells are in use by small landholders for irrigation and some domestic supply. Unless the depth is shown in Figure 5, the wells are less than 150 feet in depth.

4. Seismology

The two major faults in the bay region, the San Andreas Fault and the Hayward Fault, have been the source of several severe earthquakes and of many smaller ones. Some of larger known quakes of record are as follows:

<u>Date</u>	<u>Epicenter</u>	<u>Magnitude</u>
1836, June 10	Hayward Fault	
1838, June	San Andreas Fault	
1865, October 8	San Francisco Area	
1868, October 21	Hayward Fault	
1906, April 18	San Andreas Fault	8.25

TABLE B-I

AVERAGE MONTHLY TEMPERATURE INVERSION DATA
Surface to 381 meters Elevation

Month	Time					Time				
	**	3:15 A. M.		3:15 P. M.		**	3:15 P. M.		3:15 P. M.	
		Inv. Base*		Inv. Top*			Inv. Base*		Inv. Top*	
		Height	Temp.	Height	Temp.		Height	Temp.	Height	Temp.
1960										
May	9	115	10.8	604	17.2	11	228	15.9	608	18.3
June	13	7	14.2	683	25.5	21	205	18.2	628	25.1
July	11	99	13.4	809	29.2	27	177	17.5	750	25.6
August	16	119	12.1	784	24.6	24	189	16.9	693	25.2
September	15	53.2	12.5	766	24.2	22	145	18.6	659	24
October	17	21.7	11	399	16	13	182	16.7	588	18.8
November	18	SFC†	7.87	355	11.3	6	174	14.5	388	21.6
December	26	21.9	5.8	509	11.4	10	145	8.7	574	12.4
1961										
January	28	20.0	4.5	700	12.6	16	151	9.9	554	13.6
February	19	38.9	7.2	375	10.5	1	191	14.5	810	17.1
March	14	SFC	7.4	190	9.6	1	270	14.7	380	17.1
April	14	SFC	9.7	328	14.9	10	171.6	15.7	476	17.6

* Height is reported in meters and temperature in degrees centigrade.

** Number of temperature inversion bases below 381 meters.

† SFC - Surface.

NOTE: This information was supplied to the State Department of Public Health Bureau of Air Sanitation, by the US Weather Bureau, Oakland Airport Station.

Since 1906 there have been some moderate earthquakes in the bay area. Some of the quakes that were felt in the Santa Clara Valley are as follows:

<u>Date</u>	<u>Epicenter</u>	<u>Magnitude</u>
1911, July 1	Centered near Coyote, Santa Clara County	
1914, October 17	Bay Area	
1933, May 16	Centered near Niles, Alameda County	5.0
1937, March 8	Bay Area	

There have been a number of small shocks felt in the Santa Clara Valley in recent years. The following is a list of quakes felt since June 1955:

<u>Date</u>	<u>Epicenter</u>	<u>Magnitude</u>
1955, June 16	Southwest of Palo Alto	2.4
1955, September 5	6 miles west of Mt. Hamilton, affected 12,000 sq. mile area	5.0
1955, September 5	After shock	2.8
1955, October 24	Walnut Creek, Concord Area affected all Bay Area	5.4
1955, October 26	After shock	3.1
1955, October 28	Northeast of San Jose	2.3
1955, November 9	Northeast of San Jose	2.5
1955, November 22	South of San Francisco Bay Area	2.2
1955, December 1	South of San Francisco felt in San Mateo	2.4
1955, December 3	15 miles south of Palo Alto	2.5
1956, April 8	14 miles south of Palo Alto	2.0
1956, August 1	South of Palo Alto	2.3
1957, March 22	South of San Francisco	5.5
1957, April 7	South of San Francisco	3.3
1957, August 16	South of San Francisco	2.9
1957, September 2	South of San Jose, affected 1500 sq. mile area	3.9
1957, October 31	Southwest of Palo Alto	4.1
1958, July 8	West of Morgan Hill, affected coastal area of about 3000 sq. mile	4.1
1958, October 30	Northeast of San Jose, affected 1500 sq. mile area	4.2
1958, December 11	Southwest of San Francisco, affected 6500 sq. mile area	4.7

<u>Date</u>	<u>Epicenter</u>	<u>Magnitude</u>
1959, March 2	Near Gilroy, affected land area of 8000 sq. miles	5.3
1959, March 23	Southwest of Mt. Hamilton	4.1
1959, December 28	Northwest of Hollister, affected area of 5000 sq. miles	4.7
1960, January 19	South of Hollister, affected area of 7500 sq. miles	5.0
1961, April 8	South of Hollister	5.8 & 5.5

Seismological data were obtained from the State of California, Department of Natural Resources, Bulletin 154, Geologic Guidebook of the San Francisco Bay counties, the University of California Seismological Department and the United States Department of Commerce, Coast and Geodetic Survey.

Seismological history for this area indicates the probability of major shocks at unpredictable intervals. However, when structures have been properly designed in accordance with code requirements and sound engineering design no material damage has resulted.

FIG. 1

ADJACENT INDUSTRIES

1. SYLVANIA MICRO-WAVE LABORATORY
2. FERRY - MORSE SEED CO.
3. MINTON LUMBER CO.
4. AMES AERONAUTICAL LABORATORY
5. SYLVANIA ELECTRONIC LABORATORY
6. MOFFETT FIELD
7. CARBORUNDUM PLANT
8. SUTHERLAND LACQUER CO.
9. MT. EDEN FLOWER GROWERS ASSN.
10. HEMMETTER CORP.
11. SHELL OIL CO.
12. CARL E. ANDERSON BLDG.
13. STANDARD OIL CO.
14. HUBBARD JOHNSON LUMBER CO.
15. LITE MASTER CORP.
16. FLOWER NURSERY
17. ELEMENTARY SCHOOL
18. RHEEM SEMICONDUCTOR CORP.
19. LOCKHEED AIRCRAFT CORP.
20. JACK DYMOND CO.
21. EXACTEL INSTRUMENT CO.
22. PENINSULA TESTING & CONTROLS
23. FAIRCHILD SEMICONDUCTOR CORP.

FIG. 3

UTILITIES ADJACENT TO SITE

1/2 MILE RADIUS

LEGEND

- AGRICULTURAL WELL
- POTABLE AND IRRIGATION WELL

FIG. 5

SURFACE WIND ROSES

MOFFETT FIELD, CALIF.

FIG. 6

UPPER AIR WIND ROSES

OAKLAND, CALIF.
5000 FT. ELEVATION

WINTER

SPRING

SUMMER

AUTUMN

LEGEND

②	% OF CALMS (0-1 M.P.H.)
—	1-10 M.P.H.
□	11-22 M.P.H.
▨	23-33 M.P.H.
■	34 & OVER M.P.H.

% SCALE

FIG. 7

APPENDIX C

PROCEDURES FOR LOADING OF FUEL AND TYPICAL EXPERIMENTS

1. Procedures for Loading of Fuel

A reciprocal multiplication experiment will be conducted to determine successive amounts of fuel to be added to the reactor so as to achieve a critical loading in a safe and reliable manner. The neutron density of the reactor will be monitored by at least four instrument channels which will consist of the start-up channel (BF_3 counter and count rate meter) and servo channel (ion chamber and micro-micro ammeter) of the reactor instrumentation plus at least two additional BF_3 counters or ion chamber channels.

The first three fuel additions are specified as (1) four complete elements, (2) two complete elements and (3) two complete elements. The reciprocal multiplication will be measured several times during the withdrawal of the control rods. For example, if the rods are withdrawn sequentially, measurements will be made after each rod is withdrawn. A plot of reciprocal multiplication vs. mass of fuel loaded will be used to obtain values for the critical mass. Subsequent fuel loadings will not exceed half the amount required to achieve the most recent extrapolated critical mass value or one fully loaded fuel plate, whichever is greater.

2. Procedures for Typical Experiments

Before an experiment is conducted, a written procedure including an assessment of potential hazards, if any, will be prepared and submitted to the Manager, Nuclear Development Laboratory. He will review the experiment, potential hazards and verify that it can be performed without violating operating procedures prior to approving the experiment. Experiments to be performed may include, but are not restricted, to those described below.

- a. Measurement of Critical Mass (procedure in Part 1)
- b. Measurement of Control-Rod Worth

Two types of experiments may be performed: rod drops and positive period measurements.

In the rod-drop experiment, the reactor is operated at a steady power long enough to obtain equilibrium of delayed neutron precursors, at which time the control rod is instantaneously inserted into the reactor. The decay of power is analyzed to obtain the worth of the rod.

In the positive period method, a measurement is made of the reactor asymptotic positive period resulting from a step withdrawal of the control rod. The period is related

to the excess reactivity by the inhour equation. By performing period measurements with the control rod at different initial positions, a calibration of the rod and a measurement of its total worth or the total excess reactivity of the reactor (whichever is smaller) can be accomplished.

The initial rod position will be adjusted by one of three methods: (1) removal or addition of fuel, (2) positioning of another control rod or (3) uniform addition of neutron absorbers to the fuel region of the reactor.

c. Power Calibration

The power level will be determined by irradiation of foils and absolute counting of the foils to obtain the thermal neutron flux in the reactor. The reactor power will then be determined by calculation of the volume-integrated fission rate over the reactor core.

d. Rod Imbalance Effects

The object of these experiments will be to determine the amount of power reduction (decrease in neutron flux) in one half of the split-core resulting from maximum insertion of the control rods adjacent to that half with the reactor still critical. A simultaneous effect to be measured is the increase in power production of the other half. The effects will be determined by the irradiation of foils.

e. Temperature Coefficient of Reactivity

The moderator will be heated externally during operation of the reactor. The temperature defect will be obtained from the amount of control required to maintain criticality. The temperature coefficient is thus $C_t = \Delta k / \Delta t$ (% $\Delta k/k$ per degree C).

f. Void Coefficient of Reactivity

Voids will be simulated by the addition of styrofoam strips between fuel plates. From the difference in the control-rod positions required for criticality obtained in two successive runs, one with voids - one without, the reactivity worth of the void can be determined. The void coefficient is thus $C_v = \Delta k / \text{Vol.}$ (% $\Delta k/k$ per cm^3 of void.)

APPENDIX D

UTR TRANSIENT CHARACTERISTICS

1. Small Reactivity Bursts

The prediction of UTR transient characteristics is based on the close analogy between the UTR and the SPERT-I, a reactor on which an extensive series of reactor transient experiments has been performed. J. O. Bradfute* has studied this analogy and has shown that for small reactivity insertions, the core properties listed in Table D-I control the transient behavior. His analysis postulates that the temperature and void coefficients of reactivity combine to compensate for the reactivity step injected into the core. These measured parameters are introduced into the reactor kinetic equations and are then solved with the aid of an analog computer. With this approach, it is shown first that the SPERT-I burst characteristics can be predicted closely (Figure 1). The method then is extended to the UTR, with the results shown in Figure 2, which shows the bursts that can be sustained by the UTR without causing film boiling or net steam generation.

Table D-I

UTR and SPERT-I Comparison

Characteristic	Value or Description	
	UTR	SPERT-I
Neutron Lifetime, sec	1.35×10^{-4}	0.5×10^{-4}
Temperature Coefficient of Reactivity, $\rho/^\circ\text{C}$	-6.8×10^{-5}	-9.0×10^{-5}
Maximum-to-Average Flux Ratio	1.53	1.9
Coolant	Water	Water
Heat Capacity of Core, mw-sec/ $^\circ\text{C}$	0.309	0.311
Heat-Transfer Area, ft ²	126.6	263
Pressure	Atmospheric	Atmospheric
Temperature, $^\circ\text{C}$	20-100	~20-97
Initial Stable Period, sec	0.140 to 0.270	0.51 to 0.343
Heat-Transfer Mechanism	Convection and subcooled boiling	Convection and subcooled boiling

2. Large Reactivity Bursts

The study of UTR transients has been extended by A. M. Leppert** to cases with reactivity inputs up to 2% $\Delta k/k$. This prediction is based on a semi-empirical approach developed from the so-called Fuchs model, in which an unspecified shutdown mechanism, represented by an adjustable coefficient, is postulated to shape the burst and limit it.

* "Power Burst Characteristics of the UTR," Advanced Technology Laboratories, a Division of American-Standard, ATL-322, June, 1960.

** Intra-Department (Nuclear Design & Development) Memoranda from A. M. Leppert to G. Hovorka: "Hazards Analysis of the Maximum Credible Accident of the UTR Reactor Family," 1 December 1960; "Hazards Analysis of the Maximum Credible Accident of the UTR Reactor Family, Addendum I," 12 January 1961; "Boiling Stability of the UTR Reactors," 24 January 1961.

Application of this model to the SPERT-I experiments has shown that the characteristics of bursts from reactivity insertions ranging up to 4% $\Delta k/k$ for a variety of cases can be correlated as functions of the reciprocal period in terms of a shutdown parameter, C_v/ℓ . * The peak power, P, and energy release to peak power, E, are found to be inversely proportional to the square root of this parameter; that is, they obey equations D-1 and D-2 respectively:

$$P_1 (C_v/\ell)_1^{\frac{1}{2}} = P_2 (C_v/\ell)_2^{\frac{1}{2}} \text{ at constant } \alpha, \quad (D-1)$$

$$E_1 (C_v/\ell)_1^{\frac{1}{2}} = E_2 (C_v/\ell)_2^{\frac{1}{2}} \text{ at constant } \alpha, \quad (D-2)$$

where P = burst peak power (mw),

E = energy release to burst peak power (mw-sec),

C_v = void coefficient of reactivity $\left(\frac{\% \Delta k/k}{\% \text{ void}} \right)$,

ℓ = reciprocal period corresponding to reactivity (sec^{-1}).

The UTR shutdown parameter $\left(C_v/\ell = 1.81 \frac{\% \Delta k/k}{\text{cm}^3 \text{-sec}} \right)$ falls into the mid-range

covered by the SPERT-I experiments. This fact and the other evidence of UTR and SPERT-I analogy give confidence to the prediction of UTR behavior on the basis of SPERT-I experiments. The peak power and energy release to peak resulting from step injections of reactivity are given as functions of excess reactivity in Figures 3 and 4, respectively. These values are estimated to be accurate within 25%. The Fuchs model on which the correlation is based predicts that the shape of the burst is given by the expression

$$\frac{1}{P} \frac{dP}{dt} = \alpha - b [E(t - \tau)]^n, \quad (D-3)$$

where P = reactor power (mw),

t = time after reactivity injected (sec),

α = reciprocal period (sec^{-1}),

b = shutdown coefficient, a function of C_v/ℓ ,

τ = delay between energy release and resulting shutdown effect (sec),

n = a constant = 1.5 for UTR.

The burst shape is shown in Figure 5, and the predicted values for various bursts are summarized for UTR-B, UTR-10, and UTR-100 in Table D-II. These data show that under the most severe conditions studied (2% $\Delta k/k$ excess reactivity injection), the maximum fuel-plate surface temperature is about 300°C less than the melting point of aluminum. The pressure surge could damage the core under these extreme conditions, but no fission-product release or other adverse effect on the surroundings would ensue.

* S. G. Forbes, et al, "Analysis of Self-Shutdown Behavior in the SPERT I Reactor," Phillips Petroleum Co., IDO-16528, 23 July 1959.

Table D-II

Characteristics of UTR Power Bursts

Reactor	Reactivity	Stable Period, $1/\alpha$	Peak Power, $P_{\max}$	Energy to Peak Power, E	Total Energy of Excursion	Fuel-Plate Surface Temp.	Peak Pressure
	<u>(% $\Delta k/k$)</u>	<u>(sec)</u>	<u>(mw)</u>	<u>(mw-sec)</u>	<u>(mw-sec)</u>	<u>(°C)</u>	<u>(psig)</u>
UTR-B	0.25	9.0	0.2	5.8	8.7	70	0
UTR-10	0.50	1.0	1.43	3.9	5.9	100	0
UTR-100	1.00	0.032	1.7	9	14	200	2
UTR-100	1.50	0.0154	520	16	24	275	10
UTR-100	2.00	0.010	1300	23	35	350	20

SPERT-I POWER BURSTS
 COMPARISON OF ANALOG COMPUTER SIMULATIONS WITH
 EXPERIMENTAL RESULTS
 (SMALL REACTIVITY INSERTIONS)

FIGURE 1

UTR POWER-BURST CHARACTERISTICS
(SMALL REACTIVITY INSERTIONS)

FIGURE 2

UTR PEAK POWER AS A FUNCTION OF EXCESS REACTIVITY

FIGURE 3

UTR ENERGY RELEASE TO PEAK POWER
AS A FUNCTION OF EXCESS REACTIVITY

FIGURE 4

366_054

DIMENSIONLESS PROFILE OF UTR POWER BURST
(LARGE REACTIVITY INSERTIONS)

FIGURE 5

APPENDIX E

FUEL MELTDOWN ANALYSIS

Melting of the fuel during the excursion is not deemed credible since the moderator-coolant water must be present for the excursion to occur, the fuel plates do not reach melting temperatures during the excursion, and the excursion is essentially terminated by the thermal expansion of the fuel plates before significant boiling of the coolant occurs. However, the following sequence of events, by which localized melting might occur after termination of the excursion, has been postulated.

The excursion occurs releasing up to 200 Mw-sec of energy and causes rupture of the core tanks allowing the coolant to drain out. Subsequent decay heating of the fuel elements may be sufficient to cause partial meltdown. In a 200 Mw-sec energy release, 17 Mw-sec is released during the decay of the fission products. This amounts to 126 calories per gram of fuel element, an amount sufficient to cause an average fuel element temperature rise of 500°C which, in itself, is insufficient to cause melting. In addition, some of the heat will be lost due to conduction to the coolant prior to leakage, evaporation of a water film from the surfaces of the fuel plates and a small amount of transfer by convection and radiation.

Nevertheless, the non-uniform decay-heat generation resulting from neutron-flux peaking in the center region of the core may cause localized melting. If melting does occur, only a small fraction of the fission-product gases will be released. Parker and Creek* state that in experiments conducted on MTR type aluminum-alloy fuel that a total of from 1 to 3% of the total iodine and from 8.4% to 11.2% of the total rare gases are released during three successive meltings of the fuel during which the melt is held at 700°C for two minutes.

An analysis of the hazard resulting from a partial fuel element meltdown was performed in which the following assumptions were made:

1. An energy release of 200 Mw-sec occurred causing rupture of the core tanks and loss of coolant resulting in 10% of the fuel melting.
2. 10% of the rare gases and iodine present in the molten fuel are released to the atmosphere after being detained in the reactor shield and building for 100 seconds.
3. Long-lived (> 20 minute half life) fission products with gaseous daughters and short-lived (< 3 second half life) gaseous fission products and particulate activity are retained in the melt.

* G. W. Parker and G. E. Creek, "Fission Product Release from Molten Reactor Fuels," Peaceful Uses of Atomic Energy, 1958, Vol. 11, page 171.

4. Atmospheric inversion conditions which inhibit rapid dispersal of the fission-product cloud exist, and the point of maximum fission product concentration remains at ground level.

Fission-product accumulation prior to the postulated power burst is negligible. The fission products which could be released consist of Br, Kr, I, Xe and their daughters. One hundred seconds after the power burst there would be 2000 curies of activity released; at 300 seconds the activity would have decayed to 900 curies and at 1000 seconds to 240 curies.

The maximum concentration of fission-product gases as a function of distance down wind was computed using a modified form of Sutton's diffusion equation as stated by Leonard.* The equation is:

$$X(x,y,0) = \frac{2Q_0 \exp \left[\frac{-y^2}{C_y^2 x^{2m_y}} - \frac{h^2}{C_z^2 x^{2m_z}} \right]}{\pi \bar{U} C_y C_z x^{(m_y+m_z)}}$$

As used herein $y=h=0$. Thus, only the centerline concentration is computed.

For typical inversion conditions the following parameter values listed by Leonard were used. $C_y=0.3$, $m_y=0.84$, $C_z=18$, $m_z=0.18$ and $U=3$ meters/sec.

Table E-I shows values of the concentration of fission-product gases and integrated exposures for the various radioisotopes present. Included in the table are values for the yearly maximum exposure to these isotopes as determined from either the maximum permissible concentrations listed in 10 CFR 20, Appendix B, Table II, Column 1 or by the procedure described below.

If a value for the mpc for one isotope of an element is listed in 10 CFR 20, then a value for the mpc of another isotope of the same element can be determined from the relative β decay energies and half lives of the two isotopes. Let subscript 1 pertain to the isotope with a specified value for the mpc, and subscript 2 pertain to the isotope whose mpc is to be determined, then

$$mpc_2 = mpc_1 \frac{E_{\beta 1} T_1}{E_{\beta 2} T_2}$$

where E_{β} is the β decay energy
 T is the half life

If T_1 was long, an effective biological half life, obtained from NBS Handbook 52,** was used.

Exposures were computed by assuming that a person was exposed to the maximum concentration for a period of time sufficient for that portion of the cloud

* B. P. Leonard, Jr., "Hazards of Fission Product Release," Peaceful Uses of Atomic Energy, 1958, Vol. 11, page 157 (Published by the United Nations).

** "Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water," Handbook 52, National Bureau of Standards, March 20, 1953.

containing 95% of the fission-product gases to pass. In addition the following factors were neglected in calculating the exposures: (1) cloud rise above ground level; (2) additional dispersal of the cloud at ground level by obstructions such as trees, plants and buildings; and (3) cloud depletion by ground deposition.

The unit used to express exposure is the curie-second per meter*, the product of the cloud concentrations and the time of exposure to the cloud. Kuper and Cowan* state that for direct exposure no injury will result from exposures of 10 curie-sec/meter³ or less and the injury is unlikely for exposures up to 80 curie-sec/meter³.

Table E-I indicates that maximum permissible yearly exposure for any of the radioactive isotopes would not be exceeded at distances of 100 meters or more down wind. The sum of exposure to all isotopes is less than one maximum permissible exposure. Since decay is the main cause of reduction in exposure and the effects of decay are small in the first 100 meters down wind, exposures in the first 100 meters will not be much larger than the values tabulated for 100 meters.

An upper limit for the amount of surface deposition can be estimated from the product of the cloud concentration, deposition velocity and transit time. If a value of 0.025 for the deposition velocity, which is 10 times the velocity for iodine deposition recommended by Leonard** for typical inversion conditions, is used, 0.25 curie/meter² would be deposited 100 meters down wind. This would decay to 0.03 curie/meter² in 1000 seconds and would be negligible within a few hours.

* J.B.H. Kuper and F. P. Cowan, "Exposure Criteria for Estimating the Consequences of a Catastrophe in a Nuclear Plant," Peaceful Uses of Atomic Energy, 1958, Vol. 18, page 319.

** B. P. Leonard, Jr., "Hazards of Fission Product Release," Peaceful Uses of Atomic Energy, 1958, Vol. 11, page 157 (Published by the United Nations).

TABLE E-I

Radiation Exposure Resulting from a Gaseous Fission Product
Cloud at Various Distances Down Wind for Inversion Conditions

(10% of the Gaseous Fission Products from a 10% meltdown are
assumed to be released to the atmosphere 100 seconds after a
200 Mw-second power burst.)

Isotope and Half Life	Maximum Permissible Exposure ₃ in 1 year - curie-sec/meter	Exposure at specified distances down wind - curie-seconds/meter ³			
		100 meters	400 meters	1000 meters	3000 meters
Br ⁸³ (2.4h)	0.13	0.020	0.005	~0	0
Br ⁸⁴ (3.2m)	1.3 *		0.012	0.01	0.01
Br ⁸⁵ (3.0m)	25 *	0.32	0.25	0.15	~0
Br ⁸⁷ (55s)	3.1 *	0.68	~0	0	0
Kr ⁸⁷ (78m)	0.65	0.015	0.018	0.016	0.01
Kr ⁸⁸ (2.8h)	1.3 *	0.019	0.016	0.013	0.01
Kr ⁸⁹ (3.2m)	16 *	0.93	0.60	0.31	0.01
Rb ⁸⁸ (18m)	4.7 *				0.01
Rb ⁸⁹ (15m)	12 *	0.085	0.12	0.14	0.09
Rb ⁹⁰ (2.7m)	17 *	1.10	0.70	0.40	0.01
Sr ⁹⁰ (28y)	0.00032	~0	~0	~0	2 x 10 ⁻⁶
Sr ⁹¹ (9.7h)	0.28	0.004	0.004	0.004	0.004
I ¹³⁵ (6.7h)	0.13	0.014	0.013	0.012	0.007
I ¹³⁶ (86s)	6.3 *	1.8	0.80	0.25	~0
Xe ¹³⁷ (3.9m)	126 *	1.1	0.90	0.39	0.04
Xe ¹³⁸ (17m)	32 *	0.3	0.30	0.17	0.08
Xe ¹³⁹ (41s)	630 *	1.5	~0	0	0
Cs ¹³⁸ (32m)	1.8 *	0.011	0.015	0.019	0.04
Cs ¹³⁹ (9.5m)	4.7 *	0.55	0.33	0.29	0.097
Cs ¹⁴⁰ (66s)	18 *	1.8	0.80	0.14	~0
Ba ¹³⁹ (84m)	1.6 *	~0	0.010	0.015	0.02
TOTAL		10	4.9	2.4	0.44

* Maximum permissible concentration for these isotopes are not listed in 10 CFR 20; they were computed by the method stated in the text.

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Office

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FROM (Name and unit)

E. R. Fleury, DL&R
B-119
Washington, D. C.

REMARKS

Attached is a copy of a license application for construction

& operation of a Class 104c utilization facility the UTR Test

Reactor to be installed & operated at Mountain View Plant.

PHONE NO.

DATE

3715

6/20/61

Advanced Technology Laboratories, Docket No. 50-189

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ATOMICS / ELECTRONICS / MECHANICS / ASTRONAUTICS

369 WHISMAN ROAD
MOUNTAIN VIEW, CALIFORNIA
YORKSHIRE 8-4461

JUN 22 1961

June 14, 1961

MASTER

Mr. Lyall Johnson
Chief, Licensing Branch
Division of Licensing and Regulation
U. S. Atomic Energy Commission
Washington 25, D. C.

Dear Mr. Johnson:

We are submitting herewith twenty-two (22) copies of a license application for the construction and operation of a Class 104(c) utilization facility known as the UTR Test Reactor to be installed and operated at our Mountain View plant. Also submitted are thirty-six (36) copies of the hazards summary report, ATL-D-619, on this facility. Four (4) copies of this report were submitted to Mr. Charles Edwards of the Licensing Branch on June 9, 1961.

We request that a construction permit for construction of the facility be issued. After satisfactory construction and inspection of the facility in conformity with the application, we request that a facility license be granted.

Our Special Nuclear Materials License, SNM-115 amended, will be in effect through August 31, 1965. This license was amended August 23, 1960, to include a General By-Product License to own and possess, but not to separate, by-product materials produced during irradiation of the fuel plates used in previous UTR operations. This license will, therefore, cover the slightly spent Exhibition UTR fuel plates which will be used for the initial experiments to be performed.

The UTR Test Reactor is a prototype for ATL's line of University Teaching and Research Reactors. It is similar to the UTR-1 and Exhibition UTR test reactors which were successfully operated at Mountain View under Facility Licenses R-41 and R-64, respectively. The UTR Test Reactor will be installed in the concrete biological shield utilized in these earlier tests. It will be used to experimentally verify nuclear properties, to train operators and to test components for the entire line of UTR Reactors, including the 100 kilowatt UTR-100 which is now being designed. Accordingly, we have extended the design, operating procedures and hazards analysis, as presented in ATL-D-619, to cover a range of operations employing excess reactivities up to the 1.5 percent required for the UTR-100.

366 2059

Mr. Lyall Johnson

- 2 -

June 14, 1961

As with other UTR Reactors, safety has been a paramount design consideration. Based on satisfactory operation of similar prototype reactors in the past and on a detailed hazards analysis, it is concluded that the operation of the reactor at this site and in the manner prescribed in the hazards summary report will not constitute a potential hazard to property or persons in the surrounding area.

Immediately upon issuance of the facility license we plan to initiate a critical experiment to verify control rod worths and the effect of control rods on flux distributions for the UTR-100 Reactor. This experiment will be performed in accordance with the Five-Control-Rod Operation described in ATL-D-619. For this experiment we will utilize core components (e.g., graphite, core tanks, fuel elements, neutron source), process system and control console from the Exhibition UTR operated under Facility License R-64 in 1960. Control rods and drives, excess reactivity and control rod instrumentation will be as specified in ATL-D-619 for the Five-Control-Rod Operation.

This is the only series of experiments now scheduled. Before proceeding with further experiments or changing the design of the reactor, plans will be submitted to you for approval. We shall advise you immediately of any change in the status of the reactor.

These initial experiments will provide the data and operating experience required to complete final design of the UTR-100. We are currently planning to begin construction of a UTR-100 Reactor in late 1961. It is, therefore, necessary that we complete the planned experiments at the earliest possible date.

We are prepared to begin installation of the reactor components immediately and will be able to start operation in the first week of July. Your assistance in expediting the facility license will be very much appreciated.

Very truly yours,

W. C. Wolff
Contracts Manager

WCW-DWB:wg
Enclosures

American Radiator & Standard Sanitary Corporation
 Advanced Technology Laboratories
 (vice Atomic Energy Division)

APPLICATION FOR CONSTRUCTION PERMIT AND FACILITY LICENSE

June 14, 1961

A. General

1. Name of Applicant: Advanced Technology Laboratories
2. Address: 369 Whisman Road, Mountain View, California
3. Description of Applicant's Business: The Corporation manufactures a wide variety of products for home and industry including, for example, air conditioning and air handling equipment, instrumentation and control mechanisms, plumbing and heating fixtures, steam boilers and heat exchangers. Advanced Technology Laboratories' activities involve nuclear, instrumentation, and materials technologies primarily in the areas of reactor design, heat transfer and thermodynamics, and design of advanced control equipment for space vehicles. ATL is engaged in the design and development of power reactors, research and training reactors, and special purpose reactors. The Division also designs and manufactures components and instruments for nuclear systems.
4. State of Incorporation: Delaware
 - a. Home Office: 40 West 40th Street, New York 18, New York
5. Directors and Officers:
 - a. Board of Directors

Courtney C. Brown	Columbia University, New York 27, New York
Mark A. Brown	111 West Monroe Street, Chicago 3, Illinois
Adrain R. Fisher	22 East 40th Street, New York 16, New York
Joseph A. Grazier	40 West 40th Street, New York 18, New York
John C. Linsenmeyer	40 West 40th Street, New York 18, New York
David L. Luke	230 Park Avenue, New York 17, New York
George P. MacNichol, Jr.	608 Madison Avenue, Toledo 3, Ohio
Alexander C. Nagle	20 Exchange Place, New York 5, New York
Henry S. Wingate	67 Wall Street, New York 5, New York
Laurence C. Ward	40 West 40th Street, New York 18, New York
Herbert B. Woodman	67 West 44th Street, New York 36, New York

Officers of the Corporation

Joseph A. Grazier President	40 West 40th Street, New York 18, New York
John C. Linsenmeyer Executive Vice President, Research	40 West 40th Street, New York 18, New York
Paul M. Augenstein Executive Vice President	40 West 40th Street, New York 18, New York
John K. Miller Vice President, International Operations	40 West 40th Street, New York 18, New York
Gerald F. Gamber Vice President, Personnel	40 West 40th Street, New York 18, New York
Howard L. Spindler Vice President, Public Relations	40 West 40th Street, New York 18, New York
Laurence C. Ward Vice President, Finance	40 West 40th Street, New York 18, New York
James L. Briggs, Treasurer	40 West 40th Street, New York 18, New York
Benjamin F. Van Wormer Comptroller	40 West 40th Street, New York 18, New York
Reed D. Andrew Assistant Secretary	Detroit 32, Michigan
David A. De Wahl Secretary	40 West 40th Street, New York 18, New York
Frederick W. Jaqua Assistant Secretary	40 West 40th Street, New York 18, New York
Sydney A. Woodd-Cahusac Assistant Treasurer	40 West 40th Street, New York 18, New York
Harry W. Siefert Assistant Comptroller	40 West 40th Street, New York 18, New York

- b. All the above-named Directors and Officers are citizens of the United States of America.
- c. The Corporation is not owned, controlled, or dominated by any alien, foreign corporation or foreign government.
- d. Applicant is acting on its own behalf and not as agent or representative of any other person.

B. Licenses

We hereby request that Facility License No. R-64 be renewed or reissued or that a new Class 104(c) license be issued in lieu thereof to permit us to construct and operate the UTR Test Reactor described in the attached hazards summary report ATL-D-619*. The license shall be valid for a period of ten years after date of issuance. We further request that a Construction Permit be issued prior to issuance of the license.

The UTR Test Reactor will be installed within the existing concrete biological shield that housed the UTR-1 Reactor under Facility License R-41 and the Exhibition UTR, Model II reactor under Facility License No. R-64. The reactor will be assembled in the above mentioned shield in the reactor building at our Mountain View, California, facility. The reactor is a prototype for ATL's line of University Teaching and Research Reactors.

The UTR Test Reactor will be similar to the Exhibition UTR, Model II reactor except that the excess reactivity and the control rod number, size, sequencing and worth and attendant instrumentation will be as described in ATL-D-619. The test reactor will be used for training reactor operators in startup and operational techniques, to test UTR components and experimental facilities, and to verify experimentally the following nuclear properties of the UTR-B (10 watts), UTR-10 (10 kilowatts) and UTR-100 (100 kilowatts) reactors:

- a. Critical mass
- b. Flux distributions
- c. Void and temperature coefficients
- d. Control rod worth and effects of rod position on flux distribution
- e. Other similar properties

The earliest completion date for the reactor is July 1, 1961 and the latest completion date is September 30, 1961. Immediately upon completion of construction and issuance of the Facility License, operation of the reactor will be initiated to experimentally verify control rod worths and effects of rods on flux distribution for the 5-Control-Rod Operation described in ATL-D-619.

C. Financial Qualifications

1. Applicant herewith submits, as evidence of its financial qualification to proceed with the work described herein, a copy of its Annual Report for 1960. This report includes the financial statements of the Corporation as of December 31, 1960.

2. Applicant further states that adequate funds have been allocated by the Corporation for construction of the reactor and performance of the experiments.

* "Summary Report on the Hazards of the UTR Test Reactor", Advanced Technology Laboratories, ATL-D-619, June 7, 1961

D. Insurance for Third Party Public Liability

A NELIA policy covering the underlying financial protection required for reactor operation is in force.

E. Technical Qualifications

1. ATL has a staff of 185. Biographies on the Division President, Dr. E. Justin Wilson, Jr., and the Director of Nuclear Technology, Mr. David P. Herron, and the experience and technical qualifications of the Division have been submitted with previous applications.

2. Design, construction, and operation of the UTR Test Reactor will be the responsibility of Dr. Sidney A. Bernsen, Manager, Nuclear Development Laboratory. Mr. George Joki, Senior Nuclear Physicist, will be in charge of the initial experiments to be performed. Background resumes on Dr. Bernsen, Mr. Joki, and on qualified staff members who may participate in operation of the UTR Test Reactor are attached.

F. Hazards Summary Report

Forty (40) copies of the Hazards Summary Report ATL-D-619 are submitted herewith.

G. Restricted Data

Applicant agrees that it will not permit any individual to have access to Restricted Data until the Atomic Energy Commission shall have determined that permitting such person to have access to such data will not endanger the common defense and security.

Respectfully submitted,

ADVANCED TECHNOLOGY LABORATORIES
A Division of American-Standard

William C. Wolff, Contracts Manager

Enclosures:

American-Standard 1960 Annual Report
Background Resumes
Hazards Summary Report ATL-D-619 (40 copies)

366 -064

SIDNEY A. BERNSEN
MANAGER, NUCLEAR DEVELOPMENT LABORATORY

EXPERIENCE

American-Standard, 1 year, Manager, Nuclear Development Laboratory.
General Atomic Division, General Dynamics, 2 years, Head of Reactor Engineering
Section, Maritime Gas Cooled Reactor Project
Argonne National Laboratory, 5 years, Associate Mechanical Engineer

As Manager of the Nuclear Development Laboratory at Advanced Technology Laboratories, Dr. Bernsen is responsible for all research and engineering development of nuclear systems and components. He has had seven years experience in the mechanical, thermal and nuclear design of reactor systems.

At General Atomic, Dr. Bernsen was concerned with the development and design of high-temperature gas-cooled reactors. His duties at General Atomic included Project Supervisor of a heavy-water-moderated gas-cooled reactor development and, most recently, Head of the Reactor Engineering Section on the Maritime Gas-Cooled Reactor Project.

At Argonne, he was associated with both the Borax, EBWR and Arbor boiling-water development programs. Dr. Bernsen also participated in 1) the development of the SPERT-III as a member of an Atomic Energy Commission planning committee; 2) the safeguard review of numerous Argonne reactor experiments as a member of that laboratory's safety review committee; 3) advanced-concept development as a member of the advanced planning section and planning committees at Argonne.

Dr. Bernsen has also participated in the establishment and review of reactor component codes and standards as the General Atomic representative to an ANS codes committee. He has performed extensive review and evaluation of numerous reactor concepts including sodium graphite, heavy- and light-water-cooled and gas-cooled reactors and has been involved in the utilization of both pressurized-water and gas-cooled in-pile loops.

EDUCATION

Purdue University	B. S. , Mechanical Engineering	1950
	M. S. , Mechanical Engineering	1951
	Ph. D. , Mechanical Engineering	1953

PUBLICATIONS

Maritime Gas-Cooled Reactor - Propulsion Plant of the Future, Paper GA1040 presented at November 1959 Meeting of Atomic Industrial Forum.
The Maritime Gas-Cooled Reactor, Paper presented at Oak Ridge Gas-Cooled Reactor Symposium - Fall 1959.
The Design and Fabrication of the EBWR Core and Fuel Handling System, Paper presented to the American Nuclear Society, 1957.
Boiling Reactor Experimentation Program at Argonne National Laboratory, Paper presented to the American Nuclear Society, 1957.
Evaluation of Sodium Graphite Reactors for Plutonium and Power Production, Argonne classified report, 1954.

MEMBERSHIPS

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366 1065

GEORGE JOKI
SENIOR NUCLEAR PHYSICIST

EXPERIENCE

American-Standard, $4\frac{1}{2}$ years, Senior Nuclear Physicist
Convair Division of General Dynamics Corporation, 1 year, Physicist
Phillips Petroleum Company, Atomic Energy Division, 4 years, Physicist

Since joining American-Standard, Mr. Joki has been responsible for experimental reactor physics activities at ATL. He has developed procedures for the criticality, startup, and operation of UTR reactors and conducted the actual startup of several of them. He contributed extensively to the design of this reactor and developed training manuals and laboratory courses for its use.

Mr. Joki has been active in analytic physics investigations of maritime propulsion and power reactors, and has planned, and supervised the performance of a series of power reactor critical experiments. In his previous positions, he conducted fundamental research on neutron cross sections and resonance parameters.

EDUCATION

Montana State College	B.S., Physics	1952
University of Idaho	Graduate Studies in Physics	

PUBLICATIONS

Laboratory Manual for the UTR-10, ASAE-31 (1958).
UTR Shield Design (1957).
Summary Report on the Hazards on the UTR Test Reactor, ASAE-16 (1961).
Neutron Total Cross Section of Np^{237} from 0.02 to 2.8 ev., Phys. Rev. 167, 525 (1957).
Total Neutron Cross Sections of Na, K and Rb in 0.03 to 10 ev. Region, Phys. Rev. 99, 610 (1955).
Evidence for a Second Naturally Occurring Isotope in Tantalum, Phys. Rev. 97, 565 (1955).
Hafnium and Zirconium Total Cross Section Measurements, Phys. Rev. 96, 849A (1954).
Total Neutron Cross Sections of Chlorine and Carbon, Phys. Rev. 104, 1054, (1956).
Total Neutron Cross Section of Thulium in the Energy Region 0.038 to 1.56 ev., Phys. Rev. 103, 1326 (1956).
Total Neutron Cross Section Measurements of Titanium, Zirconium and Hafnium, Submitted to Nuclear Science and Engineering (1961).
Technical Feasibility and Economic Potential of the Variable Moderator Reactor, ATL-A-109 (1960).
Startup of the Iowa State University UTR-10 Reactor, ATL-287 (1960).
Startup of the UTR-1 Reactor, ATL-244 (1960).

MEMBERSHIPS

American Nuclear Society

BRYCE W. JOHNSON
PHYSICIST-ENGINEER

EXPERIENCE

American-Standard, 1959 to present, Physicist-Engineer
Philips Petroleum Company, 6 months, Physicist
U. S. Air Force, 2 years, Engineering Officer
Pratt & Whitney Aircraft Company, 1 year, Test Engineer

At American-Standard, he has been engaged in work on fast-breeder-reactor core design and in the development and use of advanced methods of analysis of the spatial and spectral distribution of neutrons in thermal heterogeneous reactors. He has made shielding and core-design studies on the UTR. Mr. Johnson is a licensed UTR operator and has participated in the start-up of the UTR reactors at Iowa State University and Virginia Polytechnic Institute in 1959, the Exhibition UTR model at Mountain View in 1960 and the Australian Atomic Energy Commission Reactor in 1961. He served as field engineer on the Australian Reactor Project from November, 1960, to May, 1961, where he supervised the construction and start-up of the Australian UTR-10.

At Philips Petroleum Company, Mr. Johnson worked on a study of the use of plutonium as a reactor fuel, using diffusion theory and perturbation theory, and on heat generation with reactor components and thermal-stress effects.

EDUCATION

University of Idaho	B. S. , Mechanical Engineering	1954
North Carolina State College	M. S. , Nuclear Engineering	1958
Stanford University	(One year as A. E. C Fellow)	1959

PUBLICATIONS

Reactivity Effect of Equilibrium Xenon in a Nuclear Reactor (Thesis, 1958).
Plutonium Fuel Requirements for Operation of MTR, IDO 16378 (1957).
Comparison of Flux and Power Distributions and Void and Temperature
Coefficients of the MTR with U 235 and Pu 239 Fuel, IDO 16384 (1957).
Relative Neutron Lifetimes in the MTR Fueled with U 235 and Pu 239,
IDO 16391 (1957).
Reactivity Effects of Xe 135 and Sm 149 for the MTR Fueled with
U 235 and Pu 239 Fuels, IDO 16398 (1957).
UTR Shield Design, University Teaching and Research Reactor, Special Report,
ATL-384 (1960).
Startup Report on the University Teaching and Research Reactor at
Virginia Polytechnic Institute (1961).
Construction and Startup Report on the University and Teaching Research Reactor,
"MOATA," of the Australian Atomic Energy Commission Research
Establishment (1961).

MEMBERSHIPS

Sigma Pi Sigma (Physics Honorary)
Sigma Tau (Engineering Honorary)

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E. T. GELLENBECK

HEAD, MECHANICAL AND SYSTEMS DESIGN SECTION

EXPERIENCE

Advanced Technology Laboratories, 1956 to present

Job Responsibilities: Supervisory responsibility for all system and component designs for reactors.

Specific Job Assignments:

Project Engineer, UTR-10 Reactor -- Responsible for all aspects of the development of the UTR-10 reactor, which included design and testing of a 1-watt prototype and design of the 10-kw production model.

Project Engineer, Preliminary Design of 16,500-kw Nuclear Power Plant -- Responsible for preparation of a preliminary design for a 16,500-kw natural-circulation boiling-water reactor.

Project Engineer, Exhibition UTR, Cairo, Egypt -- Responsible for design, construction, and operation of a low-power UTR reactor installed at an Atomic Energy Commission Atoms at Work Exhibit in Cairo, Egypt.

Engaged in a supervisory capacity with the VMR development program, boiling mercury feasibility evaluation, and cost estimates of the ELPWR and SSPWR reactors projects.

Standard Oil Company of California, 1947-1956

Project Engineer -- Responsible for design, construction, and initial startup for the following chemical and petroleum processing plants: Meta-Xylene Plant, Polymer Plants, Vis-Breather, and a Bottleneck Removal Program for a complete petroleum refinery.

Design Engineer -- Responsible for detailed equipment design specifications, piping and instrument diagrams, plant layouts, piping layouts, and general feasibility and economic studies for a wide variety of facilities, including Para Xylene Plant, Phthalic Anhydride Plant, Hypersorber, Polymer Plants, Acid Treating Plant, Phenol Plant, Linear Accelerator (MTA), Gas Compressor Stations, Process Steam Plant, and general refinery utility systems.

EDUCATION

University of Washington

B. S., Mechanical Engineering 1946

PUBLICATIONS

UTR-1 Hazards Report

UTR-10 Hazards Report

UTR-10 Operating Instructions

Preliminary Design - 16,500-kw Nuclear Power Plant

MEMBERSHIPS

Tau Beta Pi, Engineering

Sigma Xi, Research

Zeta Mu Tau, Mathematics

JOHN O. BRADFUTE
SUPERVISORY ENGINEER, CORE DESIGN GROUP

EXPERIENCE

Advanced Technology Laboratories, 1957 to present.

Project Manager, Mercury Cooled Breeder Reactor Study. Engaged in analytical investigations of kinetic behavior and various performance characteristics of boiling-water reactors; design of in-pile experimental facilities. Engaged in evaluation of ATL competence and potential contribution to space-age programs. Project supervisor on installation, operation, and disassembly of UTR reactor at U. S. AEC exhibition in Pakistan, 1960. Group Leader of Core Design Group.

General Dynamics Corporation, 1955-1957

Engaged in heat-transfer and fluid-flow analyses and experimentation in connection with the ground handling equipment and subsystems for the Aircraft Nuclear Propulsion program carried out by Convair for the Navy. The experience dealt with liquid metals and ordinary fluids and included detailed analyses and design of components to handle radioactive fluids.

Oak Ridge National Laboratory, 1950-1955

Engaged in experimental fluid-mechanism research and in developmental activities in connection with the core hydrodynamics of circulating fuel reactors. Work included contributions to the Homogeneous Reactor Experiment and the Aircraft Nuclear Propulsion program.

EDUCATION

The Ohio State University	B. Ch. E. and M. Sc.	1950
Oak Ridge School of Reactor Technology	Selected courses as an ORNL employee	1950-1953

PUBLICATIONS

Turbulent Fluid Flow, Convair ZR-100.
Evaluation of Mercury Cooled Breeder Reactors, American-Standard, ATL-A-102, November 1959.
Classified reports related to reactor core hydrodynamics and other subjects.
Evaluation of Mercury Cooled Breeder Reactors, paper presented to the American Nuclear Society Meeting, December 1960.
Power Burst Characteristics of the University Teaching and Research Reactor, American-Standard ATL-322.
Investigation of Fluid Flow in the ART and Other Reflector Moderator Reactor Cores, Oak Ridge National Laboratories, ORNL-2199.

MEMBERSHIPS

American Nuclear Society
American Institute of Chemical Engineering
American Chemical Society
Tau Beta Pi
Phi Eta Sigma

JACK J. GORDON
SENIOR ENGINEER

EXPERIENCE

Advanced Technology Laboratories, 1957 to present

Project Engineer on Maritime Studies. Responsible for coordination and evaluation of design and economic studies of nuclear propulsion for merchant ships.

Project Engineer, University Training and Research Reactors. Responsible for detailed design and construction of UTR's.

Stanford Research Institute, 1948-1957

Head of Chemical Engineering Section - Responsible for direction of process engineering and development projects, particularly in the fields of petrochemical and wood utilization development. Part-time lecturer on chemical engineering subjects.

Chemical Process Company, 1946-1948

Supervisor of Engineering Department - Responsible for design and construction of a chemical processing plant for production of ion-exchange and adhesive polymer resins.

Tennessee Eastman Company, 1932-1946

Senior Research Chemical Engineer in charge of Pilot Plant Development. Engaged in chemical and petrochemical process development, including high-temperature pyrolytic and catalytic operation. Holder of several U. S. and foreign patents on chemical processing equipment.

EDUCATION

University of California, Berkeley		1927 - 1930
Stanford University	A. B., Chemistry	1931
Stanford University	M. A., Chemistry	1932
College of the Pacific	Graduate courses in nuclear engineering	1957

PUBLICATIONS

Contributor to Handbook of Chemistry.

Numerous publications in technical journals.

Numerous U. S. and foreign patents in chemical engineering processes and materials.

MEMBERSHIPS

American Institute of Chemical Engineers

American Chemical Society

Sigma Xi

RESA

Alpha Chi Sigma

Phi Lambda Upsilon

Registered Professional Engineer (California)

Listed in: Who's Who in Engineering, Who's Who in Chemistry, Who's Who in the West,
American Men of Science, International Blue Book.

DONALD W. BATTLES

SENIOR ENGINEER

EXPERIENCE

Advanced Technology Laboratories, 1959 to present.

Project Engineer in charge of a program to evaluate the kinetics and economics of advanced mercury cooled fast breeder reactor (MCBR) power plants. Responsible engineer for the preliminary systems and mechanical design for the small-sized pressurized water reactor (SSPWR) and for the conceptual design of a 100 Mw(e) MCBR power plant. Project Engineer in charge of preliminary design of a 100-kw University Teaching and Research Reactor.

Carroll E. Bradberry & Associates, Consulting Engineers, 1957-1959.

Project Engineer engaged in the process design and over-all project coordination of petroleum refinery, pipeline, steam boiler, and sea water distillation facilities for installation in the Middle East.

Standard Oil Company (New Jersey), 1952-1957

Process Design Engineer engaged in design, operations control, and economic evaluation of petroleum refinery processes, equipment, utilities, instrumentation, and safety facilities, and with related operations such as petroleum handling, product blending, chemical treating, and chemicals recovery and production.

Pittsburgh Plate Glass Co., 1950-1952

Development Engineer engaged in the development of protective coatings and associated applications systems.

EDUCATION

University of Wisconsin

B. S., Chemical Engineering

1950

PUBLICATIONS

Evaluation of Mercury Cooled Breeder Reactors, paper presented to the American Nuclear Society Meeting, December 1959.

Evaluation of Mercury Cooled Breeder Reactors, ATL-A-102, November 1959.

A Continued Evaluation of Mercury Cooled Breeder Reactors, ATL-A-116,
January 1961.

MEMBERSHIPS

Registered Professional Engineer (California)

American Institute of Chemical Engineers

American Nuclear Society