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ULTRA HIGH TEMPERATURE REACTOR EXPERIMENT
(UHTREX) HAZARD REPORT

prepared by
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ABSTRACT

UHTREX utilizes a high-temperature, helium-cooled, graphite moderated reactor employing unclad, refractory fuel elements. The reactor is designed to produce a maximum thermal power of 3 Mw and a maximum exit helium temperature of 2400°F. The purpose of the experiment is to evaluate the advantages of the simple fuel against the disadvantages of the associated operation of a contaminated coolant loop.

The mechanical and nuclear design of the reactor and related apparatus are described, discussed, and evaluated from the standpoint of hazards associated with conduct of the experiment. The building design and characteristics of the site are also examined from the same standpoint.

The probable effects of operational errors and component failures are studied. The consequences of credible accidents are not considered to be catastrophic for either operating personnel or personnel in surrounding areas.

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1. INTRODUCTION

As a step toward the development of a more economic nuclear fuel, the Los Alamos Scientific Laboratory has undertaken an Ultra High Temperature Reactor Experiment that involves the use of simple, unclad, refractory fuel elements in a helium-cooled, graphite-moderated reactor to be operated at a thermal power of 3 Mw and a maximum coolant temperature of 2400°F. Experience gained in the operation of the UHTREX reactor will enable a valid judgment of the advantages of a fuel of utmost simplicity against the disadvantages of the concomitant operation of a contaminated coolant system.

UHTREX is the culmination of a series of proposals for experiments built around the basic reactor concept of unclad refractory fuel elements in a graphite, gas-cooled, movable core. In the first Turret proposal,¹ a closed-cycle turbine was to be driven with nitrogen heated to 1400°F in the reactor. Design parameters were changed in the second proposal^{2,3} to deliver helium coolant from the reactor at 2400°F and 500 psia. Considerable design effort went into the development of the second proposal for the Turret reactor experiment^{4,5,6,7,8} before its termination in January, 1961. UHTREX, which dates from May, 1961, is an extension of the Turret experiment.^{9,10}

Should fuel simplification outweigh the difficulties arising from the operation of a contaminated coolant circuit, UHTREX may provide the basis for significant economic gains to be made in future gas-cooled reactors for both power and process-heat applications. As its fundamental purpose, therefore, UHTREX will demonstrate the practical effects on both the reactor and the coolant system of using porous, unclad, refractory fuel elements.

1.1 Summary

A summary of the reactor characteristics is given in Table 1.1.1.

1.2 Conclusions

For the assessment of hazards, the UHTREX design was analyzed for accidents that could arise from operational errors and for accidents that would result from mechanical failures. The maximum credible accident was determined to be a ductile rupture of the steel reactor pressure vessel.

This rupture would release into the secondary containment enclosure all of the primary loop helium coolant and would open a hole through which the air enclosed in the containment structure could enter the hot graphite reactor core. Coolant, mobile fission products, graphite combustion products, and the residuals of the enclosed air would fill the secondary containment to create a maximum pressure of 4.3 psig at a gas temperature of about 130°F. Despite the temperature and pressure rises inside the structure, the secondary containment would remain intact. Leakage of gas through the steel-lined concrete walls of the structure would occur at a rate of no more than 1% per day of the enclosed volume.

Radiation levels due to fission product activity carried in the gas leakage should not be high enough to necessitate evacuation of personnel from uncontrolled areas. Laboratory employees in nearby technical areas might be forced to leave their sites temporarily. Operating personnel at the UHTREX site would be able to remain safely in the shielded and ventilated control room for an indefinite period after the maximum credible accident.

TABLE 1.1.1

SUMMARY OF UHTREX REACTOR CHARACTERISTICS

Reactor Type:	Experimental, high-temperature, gas-cooled, unclad fuel
Rated Power:	3 Mw (thermal)
Fuel Elements:	Graphite cylinders, 1 in. o.d., 0.5 in. i.d., 5.5 in. length; loaded by solution impregnation with enriched U, 93% U^{235}
Core Moderator:	Vertical graphite cylinder, 39 in. high, 70 in. o.d., 23 in. i.d.
Core Reflector:	4 in. of graphite plus 12 in. of dense carbon
Reactor Vessel:	Carbon steel sphere, 13 ft 2 in. i.d., 1-3/4 in. wall, internally insulated
Number of Fuel Channels:	312 (24 equally spaced in each of 13 horizontal planes)
Typical Fuel Channel:	Radial, horizontal hole; 1.1 in. diam, 23.5 in. long; 4 fuel elements
Critical Loading:	6.6 kg 93% enriched U at design power and temperature, 3.65 kg 93% enriched U at zero power and 70°F
Temperature Coefficient of Reactivity:	-2.1 x 10 ⁻⁶ Δk/k per °C (0.03 ¢/°C) in operating region (prompt) -0.8 x 10 ⁻⁴ Δk/k per °C (1.1 ¢/°C) in operating region -1.4 x 10 ⁻⁴ Δk/k per °C (2.0 ¢/°C) average from ambient to operating temperature
Power Coefficient of Reactivity:	-9.2 x 10 ⁻⁶ Δk/k per kw (-0.13 ¢/kw)
Thermal Flux:	2 to 4 x 10 ¹³ n/cm ² sec

TABLE 1. 1. 1 (continued)

Specific Power in Fuel:	450 w/g U^{235} (average)
Power Density in Core:	1.3 w/cm ³
Fuel Burnup:	10 to 50%
Refueling:	At power
Rate:	1 to 6 fuel elements daily; cycle time about 6 min/element
Excess Reactivity Required to Compensate for Temperature Coefficient:	$\Delta k = \sim 20\%$
Control Rods:	
Material:	ZrB ₂
Total Design Worth:	$\Delta k = \sim 30\%$
Withdrawal Rate:	$10^{-4} \Delta k/k$ per sec
Coolant (Maximum Design Conditions):	
Type:	Helium at 500 psia
Flow Rate:	10,250 lb/hr
Reactor Inlet Temperature:	1600° F (max)
Reactor Outlet Temperature:	2400° F (max)

2. SITE INFORMATION

2.1 Location

The UHTREX site, designated TA-52, is located within a Los Alamos area owned and controlled by the Atomic Energy Commission. The site, shown in Fig. 2.1.1, is approximately 2-1/2 miles southeast of the city of Los Alamos, 22 miles northwest of the city of Santa Fe, and 2 miles southeast of the Administration Area of the Los Alamos Scientific Laboratory, designated TA-3 in Fig. 2.1.2. Most of the LASL technical area sites and much of the residential area of the city of Los Alamos lie within a three-mile circle centered on TA-52. However, none of the town sites and only three of the LASL sites lie within the one-mile circle. Names and use of the LASL technical areas that appear on Fig. 2.1.2 are listed in Table 2.1.1.

2.2 Layout

The site is located on a narrow mesa, shown in Fig. 2.2.1. Access to the site is limited on the east and west by fences and on the other two sides by the precipitous edges of the mesa. There is little traffic on Puye Road, which extends one mile beyond the UHTREX site to Beta Site, a LASL facility used only on rare occasions. A fence near Beta Site separates the two sites.

Fig. 2.1.1 Map of area surrounding Los Alamos.

Fig. 2.1.2 Map of Los Alamos townsite and technical areas.
Boxes show local area population.

TABLE 2.1.1
LASL TECHNICAL AREAS

<u>Technical Area Number</u>	<u>Name</u>	<u>Facilities and Functions</u>
TA-1	Former Main Technical Area	Offices, laboratories, shops
TA-2	Omega Site	OWR, Water Boiler Reactor, offices, laboratories
TA-3	South Mesa Site (Main Technical Area)	Offices, laboratories, shops
TA-5*	Beta Site	Laboratories
TA-6	Two-Mile Mesa Laboratory	Offices, laboratories
TA-7*	Gomez Ranch Site	Abandoned
TA-8	Anchor Site West	Offices, laboratories, sources for radiography
TA-9	Anchor Site East	Offices, laboratories
TA-10	Bayo Canyon Site	Explosive firing, La ¹⁴⁰ dispersals
TA-11	K-Site	Explosive testing
TA-14	Q-Site	Explosive firing
TA-15	R-Site	Offices, laboratories, explosive firing
TA-16	S-Site	Offices, laboratories, explosive plant
TA-18	Pajarito Laboratory	Offices, laboratories, critical assemblies

* Sites used occasionally, by fewer than 10 persons.

TABLE 2. 1. 1 (continued)

TA-21	DP-Site	Offices, laboratories
TA-22	TD-Site	Offices, laboratories, explosive plant
TA-26*	D-Site	Storage
TA-28*	Magazine Area A	Explosive Storage
TA-35	Ten Site	Offices, laboratories, reactor development tests: LAMPRE I
TA-36	Kappa Site	Offices, laboratories, explosive firing
TA-37	Magazine Area C (PMA)	Explosive storage
TA-40	DF-Site	Offices, laboratories, explosive firing
TA-41	W-Site	Offices, laboratories
TA-42*	Incinerator Site	Incinerator
TA-43	Health Research Laboratory	Offices, laboratories
TA-45	WD-Site	Offices, laboratories
TA-46	WA-Site	Offices, laboratories
TA-48	Radiochemistry Site	Offices, laboratories
TA-49	Frijoles Mesa Site	Laboratories
TA-50**	Liquid Waste Plant	Offices, laboratories
TA-51**	Radiation Exposure Facility	Laboratories
TA-52***	Reactor Development Site	UHTREX

* Site used occasionally, by fewer than 10 persons.

** Under construction.

*** Proposed.

Fig. 2.2.1 Map of UHTREX site and locality.

2.3 Population

In the vicinity of the UHTREX site, the population is concentrated inside the LASL technical areas and within the Los Alamos townsite. The numbers of people to be found within each area are shown in boxes on Fig. 2.1.2 and summarized in Table 2.3.1. These numbers are not additive because 75% of the people listed for the technical areas are also listed for the residential areas. During working hours, there are a total of 14,500 people in the Los Alamos area.

Except for the city of Los Alamos, the area within a ten-mile circle around the UHTREX site is sparsely populated. There are few permanent residents at Frijoles, 5 miles SSE, but, during the summer, visitors to the Bandelier National Monument can raise the population to nearly 300. At White Rock, 5 miles ESE, homes are now being built in a development that may house 2000 people within the lifetime of UHTREX. At San Ildefonso Pueblo, 9 miles to the ENE, there are approximately 400 people.

In the ten-twenty mile zone, the total population is 7000, with 4500 concentrated around Espanola, 15 miles to the northeast.

Between 20 and 30 miles away from the site, the total population is almost 40,000, with 85% of the people concentrated in the vicinity of Santa Fe, 22 miles to the southeast.

There are no other population centers within 50 miles of the site. Albuquerque, population 201,000, lies 53 miles to the south.

2.4 Meteorology

2.4.1 Winds

Surface movements of air at the UHTREX site are influenced strongly by its location on a mesa that slopes away from the Jemez Mountains on the west. The site is surrounded on the other three sides by deep canyons that

TABLE 2.3.1

POPULATION WITHIN RADIAL ZONES

<u>Zone Limits (Miles)</u>	<u>Los Alamos Areas Included</u>	<u>Number of People</u>
0 - 0.5	Technical Areas	255
0.5- 1.0	Technical Areas	0
1.0- 2.0	Residential and Central Business District Technical Areas	4000 515
2.0- 3.0	Residential Technical Areas	6000 2555
3.0- 6.0	Residential Technical Areas	3500 455

drain eastward into the Rio Grande Valley, 1500 feet below. After sunset, cooler air from the mountains flows across the mesas, and channels through the canyons to the river valley. The flow is reversed during the day.

Winds are generally light and southerly. The average wind speed is less than 10 mph about 75% of the time and reaches 30 mph only 0.1% of the time, usually in gusts. Over the entire year, average wind direction lies in the quadrant from south to west about 50% of the time. Wind roses for 0600 and 1200, based on data from 11 years' observations, are shown in Fig. 2.4.1. Tabulations of average wind speed and direction for each month of the year are presented in Tables A.1 and A.2 of Appendix A.

2.4.2 Inversions

In the Los Alamos area, inversions are almost always caused by surface cooling or frontal activity. Surface cooling inversions, by far the most common, can be expected to form during any night when the skies are generally clear and wind speeds are low. On the mesas, such inversions usually dissipate by mid-morning. Inversions due to frontal activity are subject to considerable mixing caused by the moderate winds that usually accompany the fronts. Lateral mixing is reinforced by vertical mixing created by the mechanical turbulence of the winds flowing across the broken terrain.

2.4.3 Precipitation and Temperature

Average precipitation and temperature data for Los Alamos are presented in Tables A.3 and A.4 of Appendix A.

2.5 Hydrology and Geology

Theis¹¹ and Hale* describe the hydrology and geology of the Los Alamos area and of the UHTREX site in particular. UHTREX operations should have

* W. E. Hale, Appendix B of this report.

Fig. 2. 4. 1 Surface wind roses for Los Alamos based on data from eleven years' observations. Directions are those from which the wind blows.

no effect on ground water since only gaseous wastes will be discharged under controlled conditions. There seems no credible mechanism for an accidental introduction of radioactive materials into the ground water.

2.6 Seismicity

Los Alamos lies in Zone 2 on earthquake probability charts. In structural designs for the area, including the UHTREX building and reactor support structure, Zone 2 loadings are used with a seismic coefficient of 0.1. Structural parts in the UHTREX core are keyed together, and restraints are provided to prevent shifting under transverse loads.

The Los Alamos area is on the Rio Grande structural trough near faults of large magnitude, and hence, according to Hale, earthquakes might be expected to occur at any time. Further, the Los Alamos area might experience tremors from quakes originating in other parts of the trough. However, Hale continues, there is no geologic evidence to indicate that intensive earthquakes have occurred recently in the Los Alamos region.

According to S. A. Northrup,* in a region like New Mexico that is subject to a high frequency of shocks of moderate intensity, the frequent shocks act as a safety valve to relieve earth stress before it can accumulate in force to produce a violent shock. Northrup reports that of 575 earthquakes recorded in New Mexico during the century preceding 1949, 94% originated in a 75-mile section of the Rio Grande Valley extending south from Albuquerque (53 miles south of Los Alamos) to Socorro. Only 2% of these shocks were of intensity VIII to IX on the Rossi-Forel scale. (Intensity I is a microseismic shock; intensity X is a shock that causes general disaster.) One of the strongest shocks, reported to be of intensity IX, originated near Cerrillos, 30 miles south of Los Alamos, on May 28, 1918. The resultant shock at

* Collaborator in Seismology for New Mexico, U. S. Coast and Geodetic Survey; and Head, Department of Geology, University of New Mexico

Los Alamos presumably attained intensity VI or, with remote possibility, intensity VII. Another report¹² lists a brief shock of intensity V that occurred at Los Alamos on August 17, 1952.

3. REACTOR

3.1 General Description

The reactor, illustrated in Fig. 3.1.1, comprises a vertical, cylindrical core, surrounded by a reflector and insulation, encased in a spherical pressure vessel. Control rods enter the vessel from above. Fuel elements are charged through the side and discharge from the bottom. At full power, the reactor delivers 3 Mw of heat to helium coolant flowing at the rate of 10,250 lb/hr at 500 psia. The coolant enters the reactor at 1600°F and leaves at 2400°F under maximum design conditions.

The core, in which graphite serves as structural material and moderator, is a hollow vertical cylinder, 70 in. o.d. x 23 in. i.d. x 39 in. high. Fuel channels are 1.1 in. diam radial holes bored through the cylinder wall at 15° intervals in 13 horizontal planes. The vertical separation of adjacent fuel channels is 3 in. center-to-center. Surrounding the core is the 16 in. thick reflector, 4 in. of graphite and 12 in. of dense, ungraphitized carbon. Between the reflector and the 13.5 ft o.d. carbon steel pressure vessel is rigid, low-density carbon insulation. The hollow, axial region of the core is filled with a graphite plug, tapered from 22 in. diam at the top to 20 in. diam at the base of the core. The taper leaves a plenum between the plug and core, through which the helium coolant enters the fuel channels. All of the core, together with portions of the reflector and insulation, can be rotated about the cylindrical axis to accomplish fuel loading and unloading. This movable portion of the reactor is mounted on a column of dense carbon that rests on a steel plate and thrust bearing at the bottom of the pressure vessel.

Fig. 3.1.1 Reactor cross section.

Fuel elements are unclad, hollow graphite cylinders, 1 in. o.d. $\times$ 0.5 in. i.d. $\times$ 5.5 in. long, solution impregnated with highly enriched uranium. Four fuel elements rest end-to-end in each of the 312 fuel channels to make a total of 1248 elements in a complete core loading.

A fuel loader, mounted on the side of the pressure vessel, moves fuel elements into and out of the core. The loader, essentially a vertically-aligned series of horizontal motor-driven rams, operates in conjunction with an indexing mechanism that rotates the core. While the reactor is operating, a particular fuel channel may be loaded by one of the 13 loading rams after the core is indexed to one of the 24 radial channel positions. As a new fuel element is charged, a spent element is displaced into a slot in the center plug of the core, falls out of the reactor, and is mechanically conveyed to a fuel handling cell.

Two sets of rods control the nuclear operation of the reactor. The central set of eight control rods is arranged within the stationary central graphite plug in a circle whose radius is 8.125 in., centered on the vertical axis of the core. The outer set, in a concentric circle of 39.75 in. radius, is located in the graphite reflector outside the movable portion of the core. All the rods are canned cylinders of zirconium diboride.

Helium coolant enters at the bottom of the reactor, flows up through the plenum around the central plug, passes through the fuel channels, collects in an annular plenum around the outside of the movable core, and leaves the reactor through a graphite pipe at the side.

A 13.16 ft i.d. carbon steel pressure vessel contains the reactor. Its 1.75 in. wall thickness is designed for a maximum wall temperature of 650°F at a pressure of 550 psia.

3.2 Reactor Structure

The core, shown in detail in Fig. 3.2.1, is a vertical graphite cylinder, 70 in. o.d. $\times$ 23 in. i.d. $\times$ 39 in. high, fabricated from five nested, concentric, hollow cylinders. These cylinders, designed to limit thermal stresses across each monolithic piece of graphite, are keyed to a graphite ring at the top and

Fig. 3.2.1 Reactor core assembly.

rest on a column of joined pieces of graphite. The column base bears on a stainless steel plate mounted on a large-diameter ball bearing in the bottom of the reactor vessel. Fuel channels drilled through each nested core cylinder are flared at the outer edge to accommodate vertical misalignment between cylinders.

The tapered plug, in the center of the core, is a monolithic piece of graphite slotted on one side to receive spent fuel elements and bored at the bottom for the coolant inlet channel. The plug rests on other fitted pieces of graphite that are bored to direct the fuel elements downward and the coolant upward. These pieces are, in turn, supported inside a steel bell welded to the bottom of the reactor vessel. Carbon felt insulation separates the hot graphite parts from the steel walls of the bell.

Graphite and dense carbon parts of the reflector (4 in. of graphite and 12 in. of carbon) are segmented to minimize stresses and facilitate assembly. The segments bear directly on the pressure vessel at the bottom and are supported by porous carbon insulating blocks at the sides. The top of the reactor contains porous carbon blocks supported by the top reflector. Ample clearance exists between carbon blocks to prevent the transmission of expansive forces to the pressure vessel. Gaps between the blocks are filled with carbon felt to hold the blocks in place and to reduce coolant by-pass flow through the insulation.

All graphite and dense carbon components are machined from molded stock, the most nearly isotropic material that is available. To minimize the effects of the anisotropies that do exist in the materials, the transverse grain direction of each part is vertical in the final assembly. Maximum dimensional stability of the components over their lifetime is insured by the use of graphite stock prepared at graphitization temperatures in the 2800 to 3000°C range and carbon stock baked at a minimum temperature of 1000°C.¹³

Misalignment that may develop between the fuel loader and the core fuel channels is minimized by the use of matched materials in both the innermost stationary reflector and the rotating core as well as in their respective supports. In addition, the loader is designed to function even in case of a mis-

alignment of as much as 3/8 in., in any direction, between a core fuel channel and the corresponding loader channel in the reflector.

Under steady-state operating conditions, the vertical displacement of the core due to thermal expansion is closely matched by the expansion of the steel reactor vessel, with the net result that the core fuel channels, the channels through the reflector, and the channels through the reactor vessel are in the same horizontal plane. However, during a period of temperature change, differential thermal expansion can alter the relationship of the reactor vessel channels to the inner channels. To accommodate this vertical misalignment, the Graph-i-tite G tubes which form the channel between reactor vessel and graphite reflector can move 2 in. in either direction.

Allowance for dimensional changes normal to the core axis is provided in the design of interior components as a series of concentric cylinders and in the fit tolerances of other components.

3.3 Fuel Elements

No processing of fuel elements is performed at the UHTREX site. New fuel elements are prepared at other LASL facilities designed for the chemical processing of enriched uranium, and recycle operations on spent fuel elements are studied there.

3.3.1 Manufacture

Fuel elements are prepared from blanks, hollow cylinders 1 in. o.d. $\times$ 0.5 in. i.d. $\times$ 5.5 in. long, machined from extruded rods of Type AUC graphite. The porous blanks are loaded by vacuum impregnation with aqueous uranyl nitrate solution, air dried, and then denitrated by slow heating to 300°C in air. Finally, the loaded elements are baked at 800°C to convert the uranium compounds to UO_2 . With this loading procedure, pieces can be loaded to 6.5g of uranium per element with a standard deviation of loading of 2.5%. As the elements are heated in the reactor, extensive uranium carbide formation occurs,

according to the following data obtained in laboratory tests:

<u>Temperature (°C)</u>	<u>Time (hr)</u>	<u>X-ray Analysis</u>
1500	24	Equal parts UC ₂ , UC, UO ₂
	89	Equal parts UC ₂ , UC; 10% UO ₂
	170	UC ₂ :UC::3:1; 10% UO ₂
2200	1	Mainly UC ₂ ; 5 to 10% UC

3.3.2 Recycle

After the reactor has operated for some time, irradiated uranium from spent fuel elements will be recycled in a simple recovery process to be performed on a laboratory scale in off-site hot cells. The recycling will not be done on a routine basis, but will be an extension of an experimental program now in progress. Steps in the recovery process include solution extraction of uranium and fission products from the graphite element, partial decontamination of the uranium in solution by chemical precipitation and filtration, and re-use of the recovered uranium to prepare impregnation solutions for the preparation of fuel elements.

3.3.3 Nuclear Criticality Safety

In the preparation and handling of fuel elements, all operations with enriched uranium are carried out within the limitations specified in TID-7016, Nuclear Safety Guide, or within limitations specified by the LASL Nuclear Criticality Safety Committee.

Calculations were made to determine the limiting number of fuel elements which may be safely accumulated in a close-packed spherical array. Consideration was given to both bare, dry assemblies and to water flooded and reflected assemblies. Results for various enriched uranium fuel element loadings are summarized in Table 3.3.3.1.

TABLE 3.3.3.1

FUEL STORAGE SAFETY CALCULATIONS

<u>Gram U²³⁵ per Fuel Element</u>	<u>Condition</u>	<u>Critical Radius, cm</u>	<u>Critical Mass, Kg U²³⁵</u>	<u>Number of Fuel Elements</u>
9.8	Dry	86.3	313.0	32,000
9.8	Flooded	21.7	4.9	510
4.0	Flooded	24.3	2.9	715
1.5	Flooded	32.9	2.7	1,790
1.5	Dry	96.2	67.5	45,000

3.4 Control Rods

The control rod design goal is to make available 30% Δk for reactor control; current design calls for 22 rods in two sets. The rods are placed with their axes parallel to the core axis and lie in two circles centered on the core axis. The central set of eight rods, used for operational safety and control, is arranged within the stationary central graphite plug at a radius of 8.125 in. An outer set, used with the central rods during extended shutdown periods, is located at a radius of 39.75 in. in the graphite reflector outside the movable portion of the core. All the rods are withdrawn vertically upward from the core.

Each rod is a 1.25 in. diam x 40 in. long cylinder of zirconium diboride, made from natural boron and compressed to a minimum density of 4.65 g/cc, that is contained in a metallic can. The canned rods, with an over-all diameter of 1.375 in., move in 2.0 in. diam holes in the graphite and carbon parts of the reactor.

An individual electro-mechanical drive, separately operated from the control console, moves each control rod. The central rod drives have two speeds, the first used for controlled movement of the rods during reactor operations, and the second, several times faster, for use only during rod cali-

bration experiments and rundowns. None of the rods has a free-fall "scram" capability.

The control rod system described briefly above and in more detail below is an interim design; nuclear and mechanical design calculations are not complete. A possibility exists that current and future calculations, when completed, will predicate a design in which control rod numbers, materials, and locations may be changed in order to achieve the 30% Δk of control.

3.5 Nuclear Design

Nuclear characteristics of the core have been calculated, for the most part using one-dimensional, five-group diffusion theory. For comparison, some calculations were made using one- and two-dimensional, five-group transport theory. In estimating control rod worths, five-group transport calculations and two-group diffusion calculations have been used. The calculational methods as applied to UHTREX are discussed in detail in reports LAMS-2482⁵ and LAMS-2561⁸.

3.5.1 Normalization

A graphite moderated, graphite reflected critical assembly in the Los Alamos Honeycomb apparatus¹⁴ was used as a reference critical experiment. The core was a 4-ft cube surrounded by a 1-ft thick reflector. The critical core consisted of 7.98 kg of 93.5% enriched uranium in the form of 0.001-in. thick foils which were uniformly distributed through the moderator. In the fuel region, the carbon-uranium atomic ratio was 6650, which falls within the range of interest for UHTREX, i. e., about 10,000 at room temperature to about 5000 at design operating temperature for critical conditions.

In the calculation of k_{eff} for the above experiment, C. B. Mills' 18-group cross sections¹⁵ were used in conjunction with a one-dimensional diffusion code, FIRE¹⁶. This code was used in calculating the cubic geometry by a technique similar to that applied in the UHTREX calculations. After a

self-shielding correction for the uranium foils was experimentally determined, a net calculated k_{eff} of 1.044 was obtained for the Honeycomb critical experiment. With a calculated void correction, FIRE code calculations for UHTREX have been normalized to a critical k_{eff} of about 1.06. The normalized critical k_{eff} for two-dimensional transport calculations, TDC code, is 1.03.

3.5.2 Critical Mass

Calculations were made to determine the dependence of k_{eff} on uranium mass at constant temperature, and on temperature for a constant uranium mass. All uranium was assumed to be 93.5% enriched. The results of the calculations are plotted in Fig. 3.5.2.1 and Fig. 3.5.2.2. For UHTREX, the critical mass at room temperature is about 3.65 kg, and at 1800° K it is approximately 6.6 kg. Typical flux profiles are shown in Figs. 3.5.2.3, 3.5.2.4, and 3.5.2.5.

In these calculations the fuel concentrations in the elements in each channel were adjusted to simulate 50% burnup, i. e., the condition that will exist when the innermost elements in the UHTREX fuel channels have about one-half the uranium content of the outermost elements, and the remainder of the elements contain appropriate intermediate amounts. The calculations were made with the additional assumption that the core and reflector were isothermal at a volume-average temperature corresponding to a calculated core temperature profile under design operating conditions. Validity of the isothermal results was verified in calculations which included the theoretical full-power temperature distribution. The results for the nonisothermal core were within a few tenths of a per cent of the k_{eff} for a given loading calculated for the isothermal condition. Therefore, many succeeding calculations are based on an isothermal core.

Fig. 3.5.2.1 Variation of effective multiplication factor with fuel mass in the core.

Fig. 3.5.2.2 Variation of effective multiplication factor with core temperature.

Fig. 3.5.2.3 Radial flux profile at a core height of 1.9 cm; 6.6 kg of 93% enriched uranium, 1800°K, 3 Mw.

Fig. 3.5.2.4 Radial flux profile at a core height of 43.8 cm; 6.6 kg of 93% enriched uranium, 1800°K, 3 Mw.

Fig. 3.5.2.5 Vertical flux profile at a core radius of 54.8 cm; 6.6 kg of 93% enriched uranium, 1800°K, 3 Mw.

3.5.3 Reactivity Coefficients

The temperature coefficient of reactivity is estimated to be $-8 \times 10^{-5} \Delta k/k$ per $^{\circ}\text{C}$ ($1.1 \text{ } \beta/^{\circ}\text{C}$) near the design operating temperature. (See Fig. 3.5.2.2). The effect of temperature on system reactivity is primarily dependent on change of cross sections with bulk moderator temperature. Since about 2.7% of the moderator graphite is in the fuel elements, variations in moderator temperature are not closely coupled to fission rate, and the main portion of the negative temperature coefficient is relatively slow in its response to changes in power level. Fuel element temperature changes do yield a small prompt temperature coefficient of about $-2 \times 10^{-6} \Delta k/k$ per $^{\circ}\text{C}$ ($0.03 \text{ } \beta/^{\circ}\text{C}$) which is augmented by direct deposition of about 5% of the fission energy in the bulk moderator by prompt neutrons and gammas. The effective prompt temperature coefficient is about $-2.1 \times 10^{-6} \Delta k/k$ per $^{\circ}\text{C}$ in the fuel alone.

With helium as its coolant, the reactor has a pressure coefficient of reactivity that is essentially zero. Furthermore, k_{eff} is essentially unchanged when the core temperature profile for 3-Mw operation at nominal design conditions is allowed to return to a volume-average temperature. The reactor power coefficient, then, is due primarily to the temperature difference required to effect the desired heat transfer between the core and coolant. At full power, with a coolant outlet temperature of 1320°C , the necessary temperature difference between coolant and fuel elements averages 220°C . The nominal mean power coefficient for operation under these conditions, relative to an isothermal core at 1320°C , is about $-9.2 \times 10^{-6} \Delta k/k$ per kw.

3.5.4 Control

The reactivities which must be accommodated by the control rods are:

Thermal effects:

Cross sections	0.18 Δk
Expansion	0.01
Equilibrium Xe and Sm	<u>0.025</u>
Total	0.215 Δk

While it is inconceivable that the gas plena would become completely filled with graphite, such a postulated filling sets an upper limit on plugging and the associated reactivity effect. If it is assumed for purposes of calculation that the plena are filled with carbon of the same density as the reflector, the calculated Δk is + 0.017. In order to allow for uncertainties in the evaluation of reactivities to be controlled and in the assessment of control rod worths, the design objective has been to achieve a net worth in the control rods of about 0.30 Δk .

Control rod worths were initially assessed with calculations, discussed in LAMS-2482⁵ and LAMS-2561⁸, that used a two-group, one-region model. Further calculations are now being done with a two-group, multi-region, diffusion code. The combined worth of the eight inner rods is about 16% Δk , and that of the remaining rods incorporated into the design will up the total worth to about 30%. The total worths of successive numbers of rods as they are inserted as shown in Fig 3.5.4.1; it is seen that in the central plug, negative rod shadowing occurs because of the close spacing of the rods.

The central control rods will be used for operational safety and control, and the remainder will be inserted along with the central ones during extended shut-down periods. Rod withdrawal rates are predetermined to restrict reactivity addition by any rod to no more than about 1¢/sec, and sequencing interlocks prevent the withdrawal of more than one rod at a time. The position of

Fig. 3.5.4.1 Control rod worths. ZrB_2 rods, 1.25 in. o.d., natural boron.
 ZrB_2 density 4.65 g/cm³.

each rod is continuously indicated in the control room with an accuracy that amounts to $10^{-5} \Delta k/k$ or better.

Provision of the usual "scram" capability in the control rod drives is made unnecessary by the unique design of the reactor. The rapid addition of reactivity during fuel loading operations is a physical impossibility, even when a mechanical failure in the loading system is considered. Moderation of the core is unaffected by helium coolant pressure changes. Thermal excursions in the core arise from difficulties in the coolant loops and affect metallic components in the loops much more rapidly and radically than the refractory core is affected. Action must be taken to protect the coolant loop from thermal effects well before the reactor suffers damage. Conversely, a scram in the reactor, initiated by high temperatures in the coolant, is not sufficient to protect the coolant loops unless independent action is taken there.

The central control rods are expected to have the largest individual worths. As shown in Fig. 3.5.4.1, the sticking of one of these rods would not impair the ability to shut down the reactor completely, since the design objective is to attain a total rod worth of about 30% Δk and the reactivity that must be accommodated by the rods is calculated to be more nearly 21% Δk . With the same considerations, the withdrawal of a single rod will not bring the reactor critical even though the core is at room temperature and loaded for full power, high temperature operation.

The reactivity addition on the insertion of a fresh fuel element can be assessed from Fig. 3.5.2.1. In the vicinity of critical at operating temperature, $\Delta k/k = 0.39 \Delta m/m$ where m is the mass of uranium in the reactor; and at room temperature $\Delta k/k = 0.35 \Delta m/m$. The most heavily loaded fuel elements will contain about 6.6 g of uranium which will give $\Delta k/k = 3.9 \times 10^{-4}$ (5.6¢) at design temperature and $\Delta k/k = 6.3 \times 10^{-4}$ (9¢) at room temperature. These reactivity increments assume that the element ejected from the innermost position in the fuel channel was a blank, as would be the case during initial loading. Assuming no compensation with control rods, the addition of a 6.6-gram fuel element would place the reactor on a positive period of about 3 min, which

would increase with temperature until the moderator had restabilized about 3°C higher. The maximum loading rate, an element every six minutes, is limited by the capabilities of the handling and loading apparatus.

3.5.5 Nuclear Instrumentation

Neutron sensitive detectors which are used during startup and operation are summarized and their ranges indicated in Fig. 3.5.5.1. During normal operation the overlap of channels provides at least two independent indications of the reactor level except at the lowest levels. During initial startup the lowest levels will also be duplicated. The three "Safety Linear Level" channels are not intended to provide precision read out; they are discussed below in section 3.5.6, Protective Circuitry.

Reliable power for these instruments is derived from a dc driven motor-generator set normally powered by rectified ac from substation lines. In the event of failure of the ac supply, the set runs on a floating battery bank.

3.5.6 Protective Circuitry

UHTREX nuclear safety circuits include the interlocking of certain vital operations, automatic reduction of nuclear power under particular conditions, and audio-visual annunciation in response to the instrumental detection of abnormal conditions. Electrical interlocks are used to enforce restrictions on certain operational procedures. Rods can be moved only one at a time except during an automatic rundown. In the fuel loading and removal mechanisms, only one end of either of the gas locks can be opened at any one time, only one loading ram can be in the withdrawn position ready to receive a fuel element at any one time, an element can be inserted in a fuel channel only if the lower gas lock is ready to receive an ejected fuel element, and a fuel element can be removed from the lower gas lock only if the transport conveyor is in position to receive the element. The core can be rotated only if all fuel loading rams are withdrawn from the core until their inner ends are flush with the inner surface of the stationary reflector.

Fig. 3.5.5.1 Ranges of neutron detectors. The shaded portions of the chamber ranges will be usable for startups from cold condition and, to some extent, after shutdown periods.

Automatic rundown is initiated by a reactor period shorter than a preset value, by neutron level above a preselected value, by power failure, and by operator action. In response to a rundown signal, two of the control rods are driven into the core at the fast insertion rate until suitable corrective action has been taken to permit termination of the rundown. Reactivity begins to diminish within several seconds after the signal. Although one rod is more than adequate for shutting down the reactor, two rods are inserted to avoid the placing of complete dependence on the successful insertion of a single rod.

After the two initial rods are inserted, the remaining central rods are automatically driven into the core, one at a time in sequence, until either all the central rods have been inserted, or the trouble has been corrected and the rundown circuit reset. The fast insertion rate is used throughout the rundown.

The period signal is obtained from a conventional log amplifier circuit containing appropriate differentiation. During steady power operations, the period rundown will be bypassed. Two linear level channels will use multi-sensitivity micro-micro ammeters with meter relays to give rundown signals. During a startup, the meter sensitivity will be adjusted to keep the trip levels within about a decade of the signal level. During stable operation, the trip levels will be set to about 20 to 30% above the operating level. The three safety channels are permanently set to give a rundown order on two out of three coincidence at about 30% above the full power level.

The annunciator channels are designed to warn operating personnel, by means of visible and audible signals, of off-normal conditions. Included in the conditions which trip annunciators are: deviations from preselected norms of flow, temperature, or pressure in various parts of the system; high radiation levels in selected locations; deviations of selected power supply voltages from normal; opening of doors into the secondary containment; and malfunctions in the ventilation system. In addition to the annunciators, there are numerous lights and meters for monitoring the positions of valves, switches, dampers, and other mechanical devices and for indicating line and power supply voltages, pressures, temperatures, flow, and similar information.

3.5.7 Power Distribution

Power distribution in the core is summarized in Table 3.5.7.1. Since a horizontal plane of symmetry passes through the central plane of the fuel channels and the core has radial symmetry, the power per fuel element is listed for the elements in the lower half of a typical vertical plane of fuel channels. The maximum-to-average power ratio in the core is about 1.2. The calculations assumed the 50% graduated fuel loading in each channel.

TABLE 3.5.7.1

POWER PRODUCED PER FUEL ELEMENT (kw)

Isothermal Temperature Distribution

Reactor Power = 3 Mw

<u>Channel*</u>	<u>Reactor Region</u>				<u>Total</u>
	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	
0	2.273	2.783	2.654	2.654	10.364
1	2.258	2.763	2.635	2.634	10.290
2	2.212	2.704	2.577	2.577	10.069
3	2.161	2.638	2.511	2.511	9.821
4	2.079	2.534	2.404	2.403	9.420
5	1.995	2.432	2.297	2.289	9.013
6	1.934	2.364	2.220	2.188	8.706

* Channel "0" is the horizontal central plane of the reactor.

3.5.8 Kinetic Behavior

The self-regulating response of the core during operation at 3 Mw has been calculated for the case of a sudden loss of coolant flow, assuming no rod motion. Two heat balance equations are written for lumped thermal parameters:

$$\frac{dT_f}{dt} = \frac{P}{C_f} - \frac{h}{C_f} (T_f - T_m)$$
$$\frac{dT_m}{dt} = \frac{h}{C_m} (T_f - T_m)$$

where T_f = fuel temperature, T_m = moderator temperature, P = core power level, C_f and C_m = heat capacity of fuel and moderator, respectively, t = time, and h = fuel-to-moderator heat transfer coefficient.

These two equations are solved simultaneously with the neutron balance equation and the six delayed neutron precursor equations. The coupling assumed between the delayed neutron equations and the thermal equations is that given by:

$$\Delta k(t) = \alpha [T_m(t_o) - T_m(t)]$$

where α = the bulk moderator temperature coefficient of reactivity.

Apart from the assumption of lumped parameters and the neglect of possible losses of precursors, the principal uncertainty is presumed to be in the lumped fuel-to-moderator heat transfer coefficient, h . Independent calculations of heat transfer have indicated that it should have a value of about 0.02 Mw/°C. Figure 3.5.8.1 shows the calculated time variation of power, fuel temperature, and moderator temperature following an assumed loss of coolant flow. About 35 sec after loss of flow, the fuel element temperature reaches a maximum about 125°C higher than the initial value. Considering the nature of the fuel elements, they should be able to withstand much more severe temperature overshoots than this.

Fig. 3.5.8.1 Power and temperature variations following a loss of coolant flow.

With unclad fuel elements the possibility exists that a temperature excursion could substantially change the rate of diffusion of fission products out of the elements, and thereby produce a positive change in reactivity. In UHTREX elements at operating temperatures, the poisons most likely to be affected are samarium and xenon. A credible excursion would produce changes in diffusion rates on the order of 10^{-4} sec^{-1} for samarium and 10^{-5} sec^{-1} for xenon. These changes are so small that a temperature overshoot would produce no significant increase in reactivity. In addition, the excursion would increase the diffusion rate of delayed neutron precursors by about 10^{-3} sec^{-1} . This second effect would tend to decrease the reactivity and compensate for the loss of poisons.

If it is assumed that all of the Xe^{135} is in the circulating gas, then about 1% of it will be in the core region at any given time. The equilibrium concentration and its reactivity worth will depend on the reactor temperature and power, but under the least favorable assumptions the removal of this Xe from the core will place the reactor on an asymptotic positive period of about 35 sec. If the reactor is permitted to re-equilibrate, and it is assumed that the coolant is lost, the average temperature will be about 18°C higher than would have been the case with the Xe retained and the coolant flow stopped.

3.5.9 Startup Source

For initial critical experiments, a PuBe_{13} neutron source will be placed in the central graphite island. To assure detection of the source neutrons in the absence of multiplication, the source strength will be 10^7 n/sec , which will require about 100 g of plutonium. For the core loaded to critical mass at room temperature, the reactivity worth of the Pu in such a source is calculated to be $\Delta k/k = 6 \times 10^{-4}$, or about 9¢. Therefore, the reactivity change due to insertion or removal of the source will pose no problem.

In order to avoid the materials problems associated with the use of an auxiliary neutron source for high temperature and high power operations, the feasibility of startup with only natural source neutrons was investigated.* For

* Appendix C, this report.

fresh fuel elements, the natural neutron source is attributed mainly to (α, n) reactions, in which the α particles come from U^{234} decay and produce neutrons when incident on C^{13} . Calculations, confirmed by an experiment, indicate that a 7 kg uranium loading should give about 10^3 neutrons/sec.

In the presence of a weak neutron source, the statistical variations of reactor neutron level from the average value predicted by the usual kinetics equations can be significant in determining the course of the rise in level during startup. As a consequence it is necessary to predict the range within which the level will lie as a function of time rather than only the average expected value.

A theory has been developed for fluctuations during startup. The prompt neutron lifetime is assumed to be very much less than the delayed neutron precursor lifetime, and only one group of delayed neutron precursors is considered. In spite of the simplicity of the model, it gives good qualitative agreement with Godiva fluctuations. For UHTREX, the expected fluctuations in neutron precursor population are found to be so small for the planned rod removal rate ($\leq 1\text{¢/sec}$) that this approximate theory should be satisfactory. In fact, the source could be nearly an order of magnitude smaller than 10^3 n/sec before the fluctuations would become serious.

3.6 Mechanical Design

3.6.1 Reactor Pressure Vessel

The reactor pressure vessel illustrated in Fig. 3.6.1.1 is a steel sphere, 13 ft 5 in. o.d., which provides structural support for the core, moderator, reflector, and insulation located within the vessel, and for external components including the fuel loading and unloading mechanisms and the control rod drive assemblies. The vessel serves as the primary containment for the helium coolant which enters the vessel through a nozzle on a bottom appendage and leaves through a side nozzle located on the vessel centerline. Carbon steel, ASTM A-212 Grade B is the principal material of the vessel, built in accordance with the 1959 edition (including revisions, addenda, and applicable

Fig. 3.6.1.1 Reactor vertical section.

cases, i.e. 1270N) of the ASME Boiler and Pressure Vessel Code, Section VIII, Unfired Pressure Vessels, as it applies to a vessel that contains lethal substances.

Penetrations of the 1.75 in. thick vessel wall are made at opposite sides, on centerlines, by the fuel loader and the helium outlet nozzle. The fuel elevator housing and a massive section penetrated by the fuel loading holes, both pieces integral parts of the vessel wall, set up in the shell severe thermal stresses that are accommodated in a 3 in. thick segment of the vessel wall. On the opposite side of the vessel, a short 38 in. i.d. nozzle terminates in a flange that mates with the inlet flange of the recuperator. A circular flanged cap, 11 ft 8 in. o.d., at the top of the vessel is penetrated by the control rod nozzles. This bolted cap with seal-welded flanges closes the opening through which the internal components of the reactor are passed during the initial assembly. At the bottom of the vessel, the massive section in which the core support bearing is mounted has two appendages. On the side is a nozzle terminated by the Conoseal flange which joins the indexing mechanism housing. At the bottom is a cylindrical extension, with a side nozzle joined to the helium inlet line and a dished bottom head penetrated by the fuel discharge channel. A final penetration of the vessel wall is made at a point in the lower half of the sphere where the thermocouple leads leave the vessel. The leads are encased in a 6 in. pipe that ends in the room below the reactor. Seals around the leads inside the pipe maintain the reactor vessel integrity.

The weight of the reactor vessel and the components it supports is borne on two horizontal support girders and four sets of vertical support girders, all welded to the vessel outer wall. Sets of vertical members terminate in brackets which rest on pads supported by four reactor support columns that are bolted to the reactor cell floor. The pads, brass under the support bracket and steel between the brass and the column top, provide surfaces over which the support brackets can move as the reactor vessel expands and contracts. Movement is restrained by bolts in oversize holes.

Design conditions for the reactor vessel are 650°F and 550 psi. In the design analysis of the vessel, extensive thermal and load stress calculations

were made to insure compliance with the pertinent codes. The results of detailed stress analyses of nonsymmetrical regions will be checked during tests of a force and pressure stress model of the vessel and during the startup of the prototype. Sets of strain gages, mounted at 20 stations on the full-scale vessel, will deliver data to a continuous plotter in the control room during initial tests.

3.6.2 Fuel Loader

A fuel loader, attached to the side wall of the pressure vessel (see Fig. 3.6.2.1), moves fuel elements through the core. In the first step of the loading operation, the fuel channel to be loaded is located by rotation of the core, which aligns the proper vertical row of channels with the loader, and by withdrawal to its rearmost position of the single loading ram that normally closes the entrance to the proper channel. Then a new fuel element drops through the gas lock at the top of the loader and stops on its side before a short steel ram. This ram moves the element through a tube to the fuel elevator, which is open on one side. In the elevator, the element moves slowly downward past slots in the loading tubes that are closed by the inactive Graph-i-tite G loading rams. When the elevator passes the slot opened by the withdrawal of a ram, the element drops out into the loading tube. The long, graphite loading ram then advances the element past the pressure vessel wall, through a graphite tube that extends to the graphite portion of the outer reflector, and into the core. In the core, the new element, driven by the ram, displaces the old elements along the fuel channel. When the new element is in place the ram withdraws until its inward end is flush with the inner surface of the stationary reflector.

Meanwhile, the innermost of the old elements, displaced into a vertical slot in the stationary central plug, drops down through a chute in the plug to the fuel discharge pipe at the bottom of the reactor vessel. The element, decelerated by contact with the discharge chute walls, comes to rest against the inlet valve to the bottom gas lock, where it can be held for sufficient time to cool below its combustion temperature in air. Should the element become wedged in the discharge chute, a Graph-i-tite G push rod rises up the chute to

Fig. 3.6.2.1 Section through fuel loader.

free the element. After a controlled passage through the gas lock, the element drops into a mechanical conveyor that terminates inside the fuel handling cell, described in Section 7. 2. 1.

Rack and pinion drives with sealed electric motors actuate the rams, elevator, and push rod in the fuel loader. Position indicators, mounted on each drive, transmit to display instruments in the control room. These indicators are used in the loading operation to determine how far the rams must be extended to properly position a new fuel element, since thermal expansion may change the insertion depth by as much as $5/16$ in. In the first step of the procedure used to find the exact distance to the surface of the core, the indexing mechanism is operated through only half its cycle to position the blank core surface (between fuel channel rows) before the loader. Then a ram is extended until contact is made, and the indicator reading is noted. The correct insertion depth, then, is the distance indicated plus the maximum calculated distance to the step inside the core channel. An element will seat satisfactorily in a channel if the outermost end of the element lies between the step and a point $1/2$ in. inside the step.

In tests with a full-scale model of the discharge chute, the discharged fuel element usually falls directly into the discharge pipe. On the rare occasions when an element lodges in the tapered portion of the chute, the push rod is used to dislodge the element into the discharge pipe.

Each gas lock, through which the fuel elements pass into the loader and out of the discharge pipe, consists of a series of ball valves. The outermost valves have leak-tight seats and shaft seals to prevent the passage of helium. The inner valves serve as mechanical stops and as radiation shields for the outer valves. A programmed sequence of pressurizations and evacuations of the inter-valve chambers assures safe entrance and exit of the elements against reactor coolant pressures.

3.6.3 Core Support Bearing and Indexing Mechanism

The rotating core of the reactor is mounted on a support plate, which in turn is supported by a combination ball-bearing gear that is 63 in. o. d. x 49-1/2 in. i. d. and rotates on 110 1-1/4-in. diam balls. The integral spiral bevel gear is at the edge of the top bearing race. Both the support plate and bearing-gear are fabricated of 17-7 PH stainless steel.

The core is rotated to each of the 24 radial fuel loading stations by an indexing mechanism which provides the necessary motion and positive index location. The indexing mechanism consists of a motorized reducer operating through a starwheel mechanism that rotates and locks the pinion drive shaft at 180° intervals. A 180° rotation of the pinion shaft, through the 12:1 ratio at the bearing gear, indexes the core one loading station. The time consumed in indexing the core from one station to the next is approximately 15 sec. To eliminate the necessity for a rotating helium seal, the complete mechanism is housed in a pressure vessel which is an integral part of the main helium containment. The mechanism extends outside of the reactor cell and is accessible for maintenance when the reactor is shut down.

The bearing, its size determined by mechanical stability considerations rather than by the weight of the core, is loaded to only a few per cent of theoretical capacity. A dry-film lubricant is used in the bearing. To evaluate the accelerated wear that is known to occur in bearings operated in a pure helium atmosphere, a long-term bearing test is being conducted with all operating conditions simulated except the radiation field. Results to date indicate that even an unlubricated bearing will survive for the life of the experiment.

4. COOLANT SYSTEM

4.1 General Description

Two closed-cycle helium loops remove, under full-power conditions, 3 Mw of heat from the reactor core and dissipate the heat to the atmosphere. A schematic diagram of the coolant systems is shown in Fig. 4.1.1. The primary coolant loop forms a closed system inside the secondary containment. Heat passes from the primary loop to a secondary cooling loop through the main heat exchanger. The secondary loop moves the heat to an air-cooled heat dump outside the reactor building. Helium flow rates in the two loops are matched; maximum flow in the primary loop is 10,250 lb/hr at 500 psi under full power conditions.

4.2 Primary Coolant Loop

The primary coolant loop components are illustrated in Fig. 4.2.1. Primary coolant helium enters the central plenum of the reactor at a temperature of 1600°F. As the coolant passes radially outward through the fuel channels, the helium gains heat and its temperature rises to 2400°F at the reactor outlet. The coolant stream then passes through the hot side of the recuperator, where the helium cools to 1400°F. From the recuperator, the helium flows to the main heat exchanger, where the stream temperature is dropped to 600°F by heat transfer to the secondary coolant loop. The coolant then passes through the blower, which can develop 9 psi pressure

Fig. 4.1.1 Schematic coolant flow diagram.

Fig. 4.2.1 Section through reactor-recuperator room.

rise, and into the cool side of the recuperator. Beyond the recuperator, the helium, its temperature restored to 1600°F, re-enters the reactor to complete the loop.

All components in the primary loop, expected to be contaminated with radioactive fission products carried in the helium stream, are built to rigid specifications that conform to pertinent codes.

4.2.1 Core

Heat transfer characteristics of the UHTREX reactor core were studied in an extensive series of calculations. These analytical investigations indicate the possibility of undesirable flow conditions that could seriously affect heat transfer in the core and create an intolerable temperature distribution across the reactor. Compensation was made for one of the simpler conditions, by-pass flow along the reactor vessel wall, by the installation of metal deflectors normal to the vessel walls in the postulated flow paths. Design improvements to rectify more complex problems, e. g. , unequal distribution of mass flow between various fuel channels, will be studied in a core flow and heat transfer model.

4.2.2 Recuperator

The recuperator is enclosed in a mild-steel pressure vessel that is joined, with bolted seal-welded flanges, directly to the reactor vessel at one end and to the main heat exchanger at the other. Direct connection with seal-welded flanges eliminates the need for any 2400°F and 1400°F helium piping, conserves space, and increases structural integrity. Weight of the recuperator is borne on spring-loaded hangers, supported on a steel frame that rests on the cell floor. The arrangement of reactor, recuperator, and main heat exchanger is shown in Fig. 4.2.1.

The recuperator, an internally-insulated, regenerative heat exchanger made of graphite, reduces the helium coolant temperature from 2400°F, too hot for heat exchangers made of ordinary materials, to 1400°F, tolerable

in the stainless steel main heat exchanger. Heat exchange in the recuperator is accomplished in a solid graphite block 12 ft long x 3 ft diam, penetrated by 437 holes, 1/2 in. in diameter, bored parallel to the long axis. The helium streams, which flow countercurrently through half of the holes in each direction, are separated at the ends of the block in headers. Pressure unbalance across the two sides of the recuperator is less than 5 psi during operation. Since the hot and cold sides of the recuperator are both in the primary loop, some cross leakage can be tolerated.

The graphite block, which rests on carbon blocks inside the pressure vessel, is insulated from the vessel with carbon felt. Temperature of the vessel walls, made of ASTM A-212, Grade B, Carbon steel, is maintained below 600°F by radiant heat transfer to water-cooled panels in the walls of the cell.

The recuperator exchanges a maximum of 1.28×10^7 Btu/hr between the hot, reactor-outlet helium and the cooler gas returning to the reactor. During operations, the cold side flow rate is varied by means of by-pass flow to obtain the desired hot side heat exchange. The heat transfer area is oversized by about 25% to allow for uncertainties in long term performance.

An experimental objective of UHTREX is the measurement of deposited fission product distribution throughout the primary loop. To aid in these measurements, wires are placed in flow passages in the heat exchanger and the recuperator. With the aid of remote-handling equipment, the wires will be removed for analysis of deposits during reactor shutdown periods.

4.2.3 Main Heat Exchanger

Joined directly to the recuperator and supported by it, a shell-and-tube heat exchanger, made of austenitic stainless steels, transfers heat from the primary to the secondary coolant loop. This main heat exchanger maintains primary containment of the helium stream that is contaminated with fission products by direct contact with the core. On the tube side of the exchanger,

the primary coolant stream drops in temperature from 1400°F to 600°F, a level compatible with the operation of electric blowers. In the shell, the temperature of the secondary coolant stream, in countercurrent flow, rises from 200°F to 1000°F.

To assure maintenance of primary containment, welded tube sheet junctions and rigorous overall inspection of the exchanger are specified. To allow for the differential thermal expansion and stress relief that may naturally occur at the higher operating temperatures, the exchanger design includes the assumption that the exchanger may be subjected to 1000 temperature cycles, any one of which might have a range as large as the difference between ambient and the maximum operating temperature. These individual cycles occur slowly (less than 100°F/hr) in heating and cooling, and the pressure differential between the hot and cold sides of the exchanger does not normally exceed 50 psi during the cycle. An additional design requirement is that the heat exchanger survive at least five events of an accidental, non-routine nature under circumstances in which the conditions in Table 4.2.3.1 prevail (i. e. , sudden accidental loss of secondary loop coolant).

4.2.4 Helium Blower

Two blowers are provided in parallel in the primary loop, and either is capable of meeting the system requirements at full flow and full power. One of the blowers serves as a standby to help assure long term operation of the experiment without the necessity of servicing a highly contaminated piece of equipment. Each blower can be isolated from the remainder of the loop by means of manually operated inlet and outlet valves. Interlocks prevent the operation of a blower unless its valves are in the full open position and also prevent the operation of both blowers simultaneously. Both blowers are suspended on spring-loaded hangers from an extension of the steel frame that supports the recuperator. Radiation shielding, mounted on the frame, forms a box around each blower.

TABLE 4. 2. 3. 1

MAIN HEAT EXCHANGER
MAXIMUM EMERGENCY CONDITIONS

Hot Side

Fluid	Helium containing fission products and probably some graphite powder or particles in suspension
Inlet Temperature	1400° F
Inlet Pressure	About 500 psia initially. Falls to 200 psia within one minute, then reduced to atmospheric in five hours
Outlet Temperature	600° F, increasing toward 1400° F
Flow Rate	Initially 10,250 lb/hr. Reduced to about 2000 lb/hr in 10 seconds and to zero flow in about 60 seconds total

Cold Side

Fluid	Helium (no solids content anticipated)
Inlet Temperature	200° F initially
Inlet Pressure	About 505 psia initially, assumed rapid decompression to ambient pressure (12 psia)
Flow Rate	10,250 lb/hr initially, decreasing to zero with decompression

The blower design provides a completely sealed unit whose only penetration to the surroundings is for electrical connections to the motor. The centrifugal impeller is driven at speeds up to 12,000 rpm by a squirrel-cage induction motor contained within the pressure housing. The blower shaft is supported on two self-acting hydrodynamic journal bearings and a hydrodynamic thrust bearing, which are expected to meet the need for long, maintenance-free operation. The lubricant for these bearings is the system helium.

For driving the blower motor, a 200-cps AC generator is coupled to a 60-cps AC motor by means of an eddy current clutch. To aid in the control of shaft speed, a direct function of power supply frequency, the output of the generator can be varied over the range from 60 cps to 200 cps by controlled slippage in the magnetic clutch. To avoid sudden changes in blower speed, which might lead to blower damage, the generator is driven by two motors which are powered from separate supply lines from the Los Alamos power distribution center. Either motor is capable of carrying the full generator load for about 30 minutes. In case of failure of one of the supply lines, automatic switch gear will immediately transfer the appropriate motor to the active supply line. Therefore, the 30-min restriction is of little consequence unless the cause of the failure is within the facility. A standby motor-generator set is provided for the added reliability of both the primary and secondary loop blowers.

Control of mass flow in the loop will be by a combination of blower speed control and operating pressure adjustment. By-passing of the reactor or throttling of the flow will not be used.

4.2.5 Recuperator By-Pass Valve

For controlling the reactor helium inlet temperature, a throttling valve is provided in a by-pass line around the low-temperature side of the recuperator. This valve is of solid body construction with a bellows or zero-leak stem seal. Seat leakage is not critical.

4. 2. 6 Piping

Since the primary coolant loop is a part of the primary containment for fission products volatilized from the fuel elements, a high degree of reliability and leak tightness is specified for all components. The number of joints and welds is reduced to a minimum and, wherever possible, piping assemblies are prefabricated to minimize the number of field welds.

To accommodate thermal expansion of the piping and pressure vessels, all primary loop components outside the reactor are suspended on spring-loaded hangers that effectively permit translational motion in any direction without introducing excessive stresses. In piping flexibility analyses, made according to the M. W. Kellogg general analytical method, the center of the reactor was considered to be fixed in relationship to the other components. All calculated stresses, under the anticipated operating conditions, fall well within permissible limits.

Heat loss from the loop is minimized by the use of either internal or external insulation. Internal insulation, made of carbonaceous materials, is used in the reactor, recuperator, and inlet end of the main heat exchanger. The pipe that connects the outlet end of the recuperator to the reactor, and the reactor inlet are protected by internal, all metal, reflective insulation. Other components have conventional external insulation.

4. 2. 7 External Cooling of Components

Water-cooled steel panels hang along three walls and across the floor of the reactor-heat exchanger cell, and one panel is suspended over the reactor. These panels form cool surfaces that increase the efficiency of radiant heat transfer from the steel vessel walls of components that are internally insulated and from the concrete walls of the cell. The panels have no material effect on the temperature of the air circulated in the cell. Cooling of vessels is effected primarily by radiation to the bare panel surfaces, maintained at 100°F by flowing water. The vessels are cooled to prevent surface temperature rises above 600°F, where allowable stresses in mild

steel begin to decrease rapidly, and to minimize the thermal stresses that develop between components that operate at different temperatures. Provision of a heat sink for the cell walls limits their ultimate temperature to a level lower than could be achieved by conduction of the deposited heat to the cool outer surfaces of the walls.

The water panels are designed to remove, at safe component surface temperatures, the total heat leak from the reactor and primary coolant loop. This design is conservative because 20% of the heat leak is normally removed in the air circulated by the ventilating system. Consideration was given to the removal of all of the heat leak with circulated air, but equipment reliability and fail-safe requirements raised more problems in an air system than in the water panels. Each panel has an individual flow monitor and control in the reactor control room. Alarms warn the operator of flow abnormalities in any panel, and a humidity monitor in the ventilation system gives early warning of water leaks.

4.3 Secondary Coolant Loop

In the secondary coolant loop, clean helium, circulated by a blower, carries heat from the main heat exchanger to the heat dump, where the thermal energy developed in the reactor is finally discharged to the atmosphere. The flow rate of helium in the secondary loop is matched with the primary loop flow rate, 10,250 lb/hr at maximum reactor power. Secondary loop helium pressure is 505 psia, at maximum power, to assure that any leakage in the main heat exchanger flows inward to the primary loop, operated at 500 psia. Barring equipment failures, the secondary loop helium contains no radioactive contaminants. Should a rupture occur in the primary coolant loop, the secondary loop serves as secondary containment. Therefore, all components in the secondary loop are built to the same standards specified for the primary loop.

4.3.1 Heat Dump

From the shell side of the main heat exchanger, helium flows, at a maximum temperature of 1000°F, to the heat dump, an air-cooled heat exchanger designed to dissipate heat to the atmosphere at a maximum rate of 3 Mw. Within the heat dump, the temperature of the helium stream drops to 200°F. The helium flows inside stainless steel, finned tubes, arranged in four horizontal banks inside a 13-ft cubical shroud. A 7-ft diam fan, located below the tube banks, forces 260,000 lb/hr of air past the tubes at maximum power. Changes in power demand are accommodated by controlling the pitch of the blades. The entire heat dump is located on a pad outside the reactor building.

4.3.2 Blower

After the 200°F helium leaves the heat dump, a centrifugal blower returns it to the main heat exchanger. This secondary loop blower is identical to the two primary loop blowers, and the standby motor generator set serves all three.

4.3.3 Piping

Eight inch, schedule 30, stainless steel pipe connects the secondary loop components. At wall penetrations, bellows joints, welded to the pipes and wall sleeves, maintain secondary containment. Thermal stresses in the pipe are mild. External insulation is used, primarily to protect personnel against accidental contact with the hot pipes.

To maintain secondary containment if the secondary loop should be ruptured, remotely controlled isolation valves are placed in the lines, just outside all the wall penetrations.

5. GAS CLEANUP SYSTEM

To preserve the chemical inertness of the coolant and to reduce the level of long-lived, volatile, fission product contamination in the primary loop, a portion of the helium stream is continuously purified in the gas cleanup system, a sidestream of the primary loop. In the cleanup system, metallic filters remove particulate matter from the stream, a copper oxide bed oxidizes chemical reducing agents, molecular sieve beds adsorb carbon dioxide and water, and cooled beds of activated carbon delay or trap volatile fission products.

Tritium, produced in the primary coolant loop by the He^3 (n,p) H^3 reaction, is removed with the reducing agents by oxidation in the oxide bed and adsorption in the sieve beds. Eventually, the tritium is vented to the stack, during sieve bed regeneration, without creating a radiation hazard.*

5.1 General Description

Figure 5.1.1 shows a simplified flowsheet for the gas cleanup and helium storage systems. The mass flow rate of helium through the gas cleanup system is 102 lb/hr, removed from the primary loop at a point near the downstream side of the primary loop blower and returned to the primary loop just upstream from the blower. An additional maximum flow of 10 lb/hr of helium may enter the cleanup system from the low pressure storage system concurrently with the flow from the primary loop. This additional gas is pumped into the cleanup system at a point upstream from the filter.

* Appendix D, this report.

Fig. 5.1.1 Schematic gas cleanup system flow diagram.

Remotely operated, stainless steel globe valves control the flow in all process lines. Valve stems are sealed with stainless steel bellows, welded to the valve body and to the stem. Valve body joints are welded wherever a seal against process helium is necessary. All valve actuators are pneumatic and readily removable without disturbing the valve body. The valves are welded into manifolds that are connected together with Conoseal joints to form the piping system. If part of a valve, other than the actuator, should fail, the entire manifold of which the valve is a part is removed and replaced.

All major process components in the cleanup system are suspended in a heavily shielded pit. The suspension system for a typical component is shown in Fig. 7.5. The inlet and outlet process and service lines for each component are brought to the top of the pit wall, where Conoseal joints are used to connect the component into the system piping manifolds. If a component should fail, it can be removed by lifting off the removable shielding overhead, breaking the Conoseal and other joints, and hoisting out the component and its attached lines.

For ease of access, the piping manifolds are hung on the outside face of the pit wall. No maintenance work will be done on actuators, valves, or lines, however, until the reactor is shut down, helium pressure is reduced to near atmospheric, and the piping manifolds are purged with clean helium.

All pipes, valves, and process equipment in contact with the helium stream are made of Type 304 or 316 stainless steel to limit the formation of scale, which could collect in valves or small-diameter lines.

5.2 Design Details

5.2.1 Filter

Helium from the primary loop and the low pressure storage system enters the gas cleanup system through a replaceable filter with porous stainless steel elements capable of removing particles as small as 1/2 micron in diameter. Initially, pressure drop through the filter is low, but, as particles

accumulate on the surfaces of the elements, pressure drop rises to an intolerable level. Then, after the system is depressurized and the filter is isolated between closed valves, the filter assembly is removed and replaced with a spare unit. The loaded assembly is disposed of in accordance with established procedures for radioactively contaminated equipment.

Filter elements are enclosed in a 36-in. vessel made of 3-1/2-in., schedule 10, Type 304 stainless steel pipe, closed with pipe caps. The helium inlet nozzle extends inside the lower head of the vessel to provide hold-up space for particles that fall off of the elements. The element assembly is welded to the outlet nozzle.

5.2.2 Preheater

After passing through the filter, helium enters the preheater at a temperature that ranges between the ambient level at startup and 600°F at full reactor power. In the preheater, the helium stream reaches 1200°F, the maximum operating temperature of the copper oxide bed. An electrical resistance heater that surrounds the preheater produces the heat. Power consumption at startup is 54 kw.

The preheater is fabricated from 40 ft of 1-in., schedule XX, Type 316 stainless steel pipe formed into a flat, four-pass coil.

The preheater is controlled by an instrument that senses the outlet temperature and then applies more or less electrical power to the electric heater. To protect the preheater, the controller is interlocked with a helium flow measuring device, thermocouples on the vessel wall, and thermocouples inside the heater jacket. If helium flow is interrupted or hot spots develop, electrical power to the heater is interrupted.

5.2.3 Copper Oxide Bed

From the preheater, the helium flows at 1200°F into the copper oxide bed. In the bed, carbon compounds in the helium are oxidized to carbon dioxide and hydrogen compounds to water. When the oxidation efficiency of the bed decreases, it is taken off stream and regenerated with oxygen.

The bed is a 14-ft length of 8-in. , schedule 120, Type 316 stainless steel pipe packed with 158 lb of knitted copper mesh, the surface of which is preoxidized at 1200°F with air until about 50% of the copper is converted to copper oxide. The copper oxide reacts with oxidizable impurities in the helium stream and is reduced to copper. Use of oxidized copper mesh, rather than some other form of copper oxide, minimizes pressure drop through the bed, yet makes a high proportion of surface area per unit mass available for reaction.

Performance of the bed is continuously monitored by an infra-red analyzer which determines the concentrations of carbon monoxide in the inlet and outlet streams. When CO is detected in the outlet stream, the analyzer sounds an alarm to warn the operator that the bed is becoming depleted.

Regeneration of the copper oxide can be delayed for a reasonable period of time after the initial warning until it is convenient to shut down the gas cleanup system. At that time, the preheater and bed are maintained at 1200°F while system helium flow is by-passed directly to the blower from the outlet of the water-cooled heat exchanger. From the blower, the helium, instead of returning to the primary loop, flows back to the preheater and copper oxide bed. Upstream from the preheater, oxygen is introduced into the helium stream which carries it to the copper oxide bed. The oxygen feed is continued at a carefully measured rate for a calculated period of time. During this reoxidation cycle, the loop containing the preheater, oxide bed, cooler, and blower is operated at a pressure slightly below that in the remaining, isolated portion of the cleanup system. This reduced pressure prevents the flow of oxygen-contaminated helium into the carbon beds. When the reoxidation is completed, the oxygen-helium mixture is vented through the stack facilities, the oxidation loop is purged with clean helium to remove traces of oxygen, and the loop is returned to the cleanup system.

A pilot plant model of the copper oxide and molecular sieve bed portion of the gas cleanup system is being operated to verify design data, determine optimum operating conditions, and test, under actual use conditions, the analytical instruments to be used in the prototype system.

5.2.4 Water-Cooled Heat Exchanger

After leaving the copper oxide bed, the helium stream enters a water-cooled heat exchanger, where the helium temperature is reduced to 100°F, the operating temperature of the molecular sieve beds. The only control on the exchanger is a temperature-actuated throttle valve that controls water flow.

5.2.5 Decay Heat Exchanger

From the water-cooled exchanger, the process helium flows into the decay heat exchanger, a shell-and-tube exchanger with 12 tubes, each a 20-ft length of schedule 80 pipe. The tubes are packed with 525 lb of activated carbon, and cooling water circulates through the shell. In the tubes, the activated carbon, by gas chromatography, delays the passage of krypton and xenon and, by absorption, quantitatively removes any iodine or bromine that has not yet deposited elsewhere. The fixed halogens and the delayed noble gases undergo radioactive decay and liberate approximately 12 kw of heat. To aid in conducting the heat from the carbon to the walls of the tubes, perforated aluminum baffles are spaced normal to the flow, on 2-1/4-in. centers, in the first two-thirds of the length of each tube. Water enters the shell at 80°F and leaves at 150°F.

Delay times in the bed are 1.2 hr for krypton and 20 hr for xenon. The rate of decay heat production through the bed diminishes from about 13 kw to 1.4 kw, the latter due primarily to long-lived krypton. By delaying the fission products while they give up a major part of their decay heat, the decay heat exchanger protects both the molecular sieve and the delay beds against localized overheating.

This exchanger is practically self-regulating and needs only a temperature-actuated valve at the cooling water exit, to adjust water flow rates.

5.2.6 Molecular Sieve Beds

From the delay bed, the helium stream flows to one of two beds, each filled with 145 lb of Linde 4A molecular sieves. The sieves quantitatively remove water and carbon dioxide from the helium. (Krypton is not adsorbed in significant quantities; xenon cannot enter the sieve pores.) Concentrations in the outlet stream are reduced to <1 ppm of carbon dioxide and water. After 24 hr onstream, at maximum outgassing rates in the reactor, the sieve bed becomes saturated and starts to pass carbon dioxide. An infra-red analyzer detects the carbon dioxide, at <1 ppm levels, in the outlet stream and sounds a warning device. The operator then valves off the saturated bed for regeneration. During the latter part of the 24-hr period, the two sieve beds are operated in series to prevent the liberation of carbon dioxide if premature breakthrough should occur.

To regenerate the sieves, the saturated bed is isolated from the system, evacuated, and heated to 600°F. Before heating starts, however, the bed is purged with clean helium to remove entrained krypton and xenon. At a flow rate of 10 lb/hr of helium, the purge period is 11 min. The purge gas passes to the low pressure receiver.

After the bed is purged, it is evacuated with the vacuum pump, and a flow of hot air is applied to the jacket. The bed is heated to 600°F over a 6-hr period while the internal pressure is held at 10 mm Hg. Then the bed is cooled, with cool air in the jacket followed by water, to 100°F in 11 hr. At 100°F the bed is repressurized with helium to 500 psia and returned to service.

5.2.7 Refrigerated Heat Exchanger

After leaving the molecular sieve bed, the process helium stream flows through a refrigerated heat exchanger, where the stream temperature is reduced from 100 to -20°F , the operating temperature of the carbon beds. The cooling medium is a water-ethylene glycol solution circulated through the exchanger shell.

5.2.8 Delay Beds

At -20°F , the process helium enters two delay beds, each 24 in. in diameter and 16.5 ft long, operated in series. Each bed is packed with 1431 lb of activated carbon, maintained at -20°F by refrigerated ethylene glycol solution circulated in external jackets. In these two beds, krypton is delayed 43.6 hr (10 half-lives of $\text{Kr}^{85\text{m}}$) and xenon for 1790 hr (14 half-lives of Xe^{133}). The beds, which remain onstream continuously at a constant temperature, need no control instrumentation.

5.2.9 Adsorption Beds

When the helium stream leaves the delay beds, its specific activity is so low that it has little effect on the overall activity of the primary loop to which the stream returns, via the primary loop blower. However, the helium that is diverted from the delay beds to the high pressure storage system (during system pressure reductions or when make-up helium is added) must be clean. To remove the last traces of krypton and xenon, this diverted helium passes through two adsorption beds, each 8 in. in diameter and 12 ft long and each packed with 131 lb of activated carbon. The beds, operated in series and cooled to -20°F by glycol solution circulated in external jackets, have the capacity to delay krypton for 44 hr at a helium flow rate of 10 lb/hr, the capacity of the high pressure compressor. As an alternative, the beds can purify the entire flow of the cleanup system for a period of 4 hr before krypton breakthrough.

In normal operations, the adsorption beds receive helium from the delay beds periodically at a maximum flow rate of 10 lb/hr and discharge to the high pressure system. When a gamma detector mounted on a line between the beds indicates that krypton has broken through the first bed, both beds are regenerated at the first opportunity, i. e. , during a period when no system pressure changes are planned or while the reactor is down for any reason. The saturated beds are regenerated by heating them to 500°F with hot air in the jackets and eluting the adsorbed gases with a clean helium purge. The purge helium can be discharged to the stack because it contains only low-activity, long-lived isotopes that have already been delayed in the decay heat exchanger and the delay beds. After regeneration, the beds are cooled to -20°F and returned to service.

5.2.10 Blower Preheater

As the main helium stream leaves the carbon beds, its temperature is -20°F. Before it enters the blower, the helium is heated to 100°F in an electrically heated exchanger.

5.2.11 Blower

As the helium stream flows through the various cleanup system components, it experiences a total pressure drop of about 5 psi. This loss in pressure is restored in a centrifugal gas-bearing blower, the last item in the normal flow path through the system. From the blower, the purified helium re-enters the primary loop at a point just upstream from the primary loop blower.

The cleanup blower is driven with variable frequency electric power produced by a motor-generator set. Two alternate methods are available for controlling the flow of helium through the blower: throttling the inlet stream to the blower, operated at constant speed; or regulating the speed of the blower by varying the frequency of the power supply. In either case, flow

through the cleanup system is measured by a meter whose output is used in an automatic controller that regulates the throttle valve or blower speed.

The blower is capable of developing at least three times the maximum expected pressure drop at the normal flow rate of 102 lb/hr. A spare blower is provided, connected electrically and piped into the helium stream.

5.3 Maintenance of Components

In anticipation of maintenance and repair problems, process units where radioactive isotopes will be concentrated are isolated behind substantial shielding, whereas process piping and valves are accessible. Valve actuators, which are expected to require the most servicing, are placed in positions that are as accessible as possible. A limited number of specially designed mechanical joints are included in the piping to facilitate the removal of pipe manifolds and process vessels.

Contact maintenance may be possible on parts of the system after waiting a reasonable time and flushing the system with clean helium. In order to provide a higher degree of assurance that maintenance operations can be performed, a remote maintenance machine (MINOTAUR) is provided for work that must be done in high radiation fields. The machine, shown in Fig. 5.3.1, is suspended from an overhead bridge crane on a telescoping vertical column. It includes two electro-mechanical manipulators, closed circuit TV cameras, lights, and a small hoist. **The machine can be operated** from either the control room or the remote maintenance corridor, which is shielded from, but has viewing windows into, the gas cleanup area.

Fig. 5.3.1 MINOTAUR remote maintenance machine.

6. HELIUM STORAGE AND PRESSURE REGULATION SYSTEM

6.1 General Description

As the reactor operates, the helium pressure in the primary and secondary loops is regulated to meet operating conditions. The source of helium added to increase primary loop pressure is the high pressure storage system, which also supplies the helium used for purging components of the gas cleanup and fuel handling systems. When helium is discharged to reduce the primary loop pressure, it goes to the high pressure storage system. New helium to make up losses flows from banks of cylinders on a transport trailer through the gas cleanup system to the high pressure storage system. The secondary loop is supplied directly from the trailer through a simple filter and adsorption bed system that removes contaminants. Figure 5.1.1 shows a simplified flowsheet for this and the gas cleanup system.

When pressure must be vented rapidly from the primary loop, a blow-down valve opens, and the helium discharges directly to the low pressure storage tanks, which are maintained (at 12 psia) for such emergency use. The pressure in the primary loop falls during blowdown to about 228 psia. The tanks are then valved off, and the low pressure system compressor empties the remaining helium from the primary loop over a 4-hr period. The compressor discharges at 500 psia into the gas cleanup system from which the now clean helium enters the high pressure system compressor. This compressor discharges the helium (at a maximum pressure of 1000 psia) into the high pressure storage system. The high pressure storage tank, held at 12 psia during normal operations, is the ultimate storage place for helium emptied out of the coolant loops.

After the primary loop is emptied, the gas from the low pressure storage tanks follows the same route as the primary loop helium to the high pressure storage tanks.

If the gas cleanup system should be inoperative, the low pressure storage tanks could contain all of the primary loop helium. In this case, the compressor evacuates the primary loop into the low pressure storage tanks which are designed to withstand a maximum operating pressure of 500 psia.

When the reactor is ready for startup, the helium stored in the high pressure system is discharged into the primary loop under its own pressure until equilibrium is reached. Then the low pressure system compressor completes the evacuation of the high pressure system to build the primary loop pressure to 500 psia. The secondary loop is repressurized concurrently with new helium.

When a controlled pressure reduction is made in the primary loop, helium is discharged from the cleanup system into the high pressure receiver, not to one of the storage tanks. From the receiver, helium flows wherever it is needed to adjust pressures or to be used as a purge.

During pressure reductions in the secondary loop, helium is discharged to a low pressure receiver. This receiver also takes the small quantities of helium periodically discharged when various components are purged and the sample gas continuously discharged from process control and analytical instruments. From the low pressure receiver, helium is returned to the gas cleanup system through the low pressure compressor and eventually is stored in the high pressure receiver.

6.2 Low Pressure System

The system consists of two storage tanks, each with a volume of 500 ft³; a receiver; and a two-stage diaphragm compressor with a precooler.

Under normal operating conditions, the storage tanks are empty; i. e., they contain helium at 12 psia. Their only function is to receive helium discharged from the primary loop during a blowdown. They are manifolded together.

As described above, the function of the receiver is to accept small quantities of helium discharged from various systems. Its maximum operating pressure is 25 psia, maintained by the compressor.

The compressor is a two-stage, V-style diaphragm compressor capable of pumping 10 lb/hr of helium at an intake pressure of 12 psia and a discharge pressure of 600 psia. It starts automatically when the pressure on its intake side reaches 25 psia, and stops when the pressure drops to 12 psia. The compressor, normally connected on its intake side to the low pressure receiver, discharges into the gas cleanup system just upstream from the preheater. When the high pressure tanks are to be evacuated, the intake side is connected to a line from these tanks. After a blowdown, the intake side is connected to the blowdown line until the primary loop is evacuated, and then to the low pressure storage tanks. A precooler on the intake side reduces the temperature of the incoming gas to 80°F.

Just upstream from the compressor discharge is an analyzer that detects hydrocarbons, released when a compressor diaphragm ruptures, and shuts down the compressor.

6.3 High Pressure System

The system consists of a storage tank with a volume of 350 ft³, a working tank of about 50 ft³ volume, and a single-stage compressor.

The storage tank is the ultimate storage place for all of the helium in the coolant and cleanup system. Ordinarily, with all systems operating at 500 psia, this tank contains helium at 12 psia. After a drastic reduction in coolant pressure, it is partially pressurized. When all of the helium is removed from the coolant and cleanup systems, the tank approaches its maximum design pressure of 1000 psia. The tank receives clean helium from the cleanup system through the compressor, and discharges to the primary loop directly or through the low pressure compressor.

Normal operating pressure of the working tank is 1000 psia. It is a working helium reservoir automatically maintained at this pressure by the high pressure compressor.

A single-stage diaphragm compressor serves the high pressure system. The compressor receives clean helium at 500 psia from the cleanup system and discharges it at 1000 psia into the receiver, during normal operations, or into the storage tank during a drastic pressure reduction. The compressor has a capacity of 10 lb/hr of helium. It is controlled by pressure-actuated switches and monitored by a hydrocarbon detector.

6.4 New Helium System

The system consists of a dock, pressure-reduction valves, and a manifold where a high pressure helium tank trailer can be connected.

Beyond the pressure-reduction valves, which drop the helium pressure from 2400 psia to the reactor coolant operating pressure, two lines carry the new helium either to the gas cleanup system or to the secondary loop. Helium flows under its own pressure from the trailer into either line automatically as leakage makes the addition of helium necessary. When the trailer pressure drops and flow stops, the residual helium in the trailer is drained into the low pressure receiver.

7. REACTOR BUILDING

A single, integrated, reinforced concrete structure, depicted in Fig. 7.1 through Fig. 7.6, houses the reactor and the associated experimental, operational, and support facilities. The building has two primary parts, separated by radiation shielding, gas-tight doors, and a gas-tight steel membrane. One portion of the building contains the reactor, the primary (contaminated) helium coolant loop, fuel handling systems, and the gas cleanup and storage systems, and serves as the secondary containment for volatile fission products carried in the helium coolant. The remainder of the building contains the control room, offices, fuel handling and gas sampling cells, utility distribution systems, and other auxiliary equipment.

7.1 Secondary Containment

The primary containment for volatile fission products consists of the reactor vessel and its attached auxiliaries, the primary coolant loop, and the gas cleanup and storage systems. To provide a second barrier to the uncontrolled release of fission products, a secondary containment is established. Its boundaries are the secondary coolant loop, the containment portion of the building, and the recirculating ventilation system that serves that part of the building.

The major part of the containment portion of the building extends above grade in a superstructure that has inside dimensions of 46 ft x 56 ft x 37 ft high and encloses a free volume of 94,000 ft³. This superstructure is designed

Fig. 7.1 General view of UHTREX facility.

Fig. 7.2 Building floor plan, ground level.

Fig. 7.3 Building floor plan, operating level.

Fig. 7.4 Building floor plan, basement level.

Fig. 7.5 Building section through reactor-recuperator room.

Fig. 7.6 Building section through control room.

to withstand an internal pressure 7.5 psi above the surrounding atmosphere with a leak rate no greater than 1% per day of the total enclosed volume (148,000 ft³) in the secondary containment portion of the building. To assure gas tightness of the superstructure, a liner made of welded steel plate covers the inside walls and roof. Below grade, this steel membrane extends within the walls to the base of the building on the two sides adjacent to habitable portions of the structure and to 5 ft below grade on the two outside walls.

All penetrations of containment boundary walls in the building are made through seals designed to meet the containment pressure and leakage standards. The only normally open discharge from the secondary containment is a 3-in line that bleeds filtered air to the stack from the recirculating ventilation system. This line passes through a solenoid valve that is tripped automatically in the event that monitors detect radioactivity above preselected levels in the circulated air. Open drains in this part of the building lead to a sump that is isolated by a normally closed valve from the remainder of the contaminated liquid waste disposal system for the building.

Operating within the superstructure are MINOTAUR, the remote maintenance machine described in Section 5.3, and a 30-ton bridge crane. Both machines can be controlled from within the secondary containment superstructure or outside it in the remote maintenance corridor.

7.1.1 Reactor and Recuperator Room

The reactor, primary coolant loop, and fuel loading equipment are located below grade in a shielded cell, approximately 18 ft x 48 ft x 38 ft deep, with a total free volume of 24,000 ft³. The top of this cell consists of removable concrete shielding slabs that permit access to the reactor and coolant system. Penetrations in the shielding exist for control rod drives, nuclear instrumentation, sampling and survey ports, fueling apparatus, coolant lines, ventilation, and utilities. Near the reactor and primary coolant loop, the walls of the cell are covered with water-cooled panels that remove heat radiated from the surfaces of hot containment vessels.

7.1.2 Helium Purification and Storage Rooms

The gas cleanup system and the associated storage system are housed in three other below-grade rooms inside the secondary containment. Over-all dimensions of these three rooms are 21 ft x 50 ft x 38 ft deep; total free volume is 30,000 ft³. Oxidation and adsorption units, heat exchangers, and storage tanks, all of which will accumulate deposits of fission products, are hung in the gas cleanup room, a long, narrow pit with magnetic concrete walls and removable, shielding slab covers. The interconnecting piping extends over the edge of the pit to valve manifolds and compressors located in the upper gas cleanup piping room. MINOTAUR has access to the upper part of this region after cover slabs are removed with the 30-ton crane. In the lower room are located clean helium, cooling water, refrigerant, and air piping manifolds and valves. Contact maintenance work can be performed in this lower room while the reactor is shut down.

7.1.3 Fuel Discharge Room

Fuel elements discharged from the reactor drop through a line that penetrates the bottom floor of the reactor cell at an angle and enters the well-shielded fuel discharge room in the subbasement level of the reactor building. In this room, the elements pass through the bottom gas lock valves and enter the bottom leg of a conveyor that travels vertically through the reactor drive room into the fuel charging room. There, the elements discharge into a horizontal conveyor that moves through the shield wall to the transfer cell.

Adjacent to the fuel discharge room are the duct space, through which ventilation conduits return to the filters, and the bottom of the pit in which rest the helium storage tanks.

Access to this entire region will be allowed only when the reactor is shut down and, even then, will be carefully controlled to limit exposure.

7.2 Operating and Control Areas

Building design provides for unlimited access to operating and control areas on three floors outside the secondary containment. On the ground floor, Fig. 7.2, are ventilation and electrical equipment rooms, a boiler room for site heating equipment, and the exhaust filter room of the operating and control areas ventilation systems.

The first subgrade floor, the operation level, Fig. 7.3, comprises a clean area and a hot area, where access is limited. In the clean area are the control room and operations office. Access to the hot area is through a hall that passes the health office to a locker room, where personnel must change clothes. From the locker room, personnel may enter the operating area serving the fuel transfer cell and the transfer lock, or may enter the remote maintenance corridor. The corridor provides access to the exhaust filter room, on the upper level, during reactor operation. When the reactor is shut down, the stairway at the end of the corridor may be used to go up or down to other levels where gas-tight doors open into the secondary containment enclosure.

On the basement level, Fig. 7.4, unrestricted access, by way of the western staircase, is limited to the office and electrical equipment areas. The mechanical equipment and sump rooms, where the possibility of radioactive contamination exists, are accessible through the maintenance corridor by way of the eastern staircase, entered from the operating level only.

7.2.1 Fuel Transfer Cell and Lock

Fuel elements move into and out of the reactor through the fuel transfer lock and cell. A schematic flow diagram for the fuel elements is presented in Fig. 7.2.1.1. New fuel elements may enter, and spent elements leave, inside casks that travel on rails between a loading dock and the transfer lock. The casks pass through a 20-in. thick cast Meehanite outer door that moves on

Fig. 7.2.1.1 Schematic fuel element flow diagram.

hydraulic lift and locking mechanisms operated from the basement. When a cask is inside the lock and the outer door is pneumatically sealed, an inner door between the lock and cell can be opened. Manipulators move elements between cask and cell under the control of operators who work before viewing windows in the cell operating area.

Inside the transfer cell is the end of a two-way horizontal conveyor that penetrates the biological shielding wall. This conveyor is closed, at its end inside the fuel-charging room, with gas-tight locks that maintain secondary containment. Spent fuel elements drop from the vertical leg of the discharge conveyor through a lock to the transfer cell conveyor, which carries them into the cell. New fuel elements are placed, with manipulators, on the cell conveyor which moves the elements to a chute inside the fuel-charging room. There, the elements fall from the conveyor, through a lock in the chute, to a vertical-charging conveyor, which elevates the elements to a gas lock above the fuel loader.

The geometry of the fuel-handling equipment prevents the accumulation of enough fuel elements to form a critical mass outside the reactor core. Sequential interlocks prevent the premature opening of cell doors, and special cell-ventilating systems prevent the spread of particulate matter outside the cell or lock when the doors are opened.

7.2.2 Remote Maintenance Corridor

Along the inner walls of the remote-maintenance corridor are control stations for MINOTAUR and the 30-ton crane. Set into the walls are three viewing windows through which can be seen the interior of the gas cleanup piping room, where the gas cleanup system valve manifolds, compressors, and analytical instruments are located. From the corridor, operators can carry out, with the aid of MINOTAUR and a Model A master-slave manipulator, adjustments and minor repair of components while the cleanup system is operating and more extensive repairs when the system is down. Through combined operation of MINOTAUR and the crane, entire piping manifolds, process vessels, and instrument components can be removed and replaced without exposure of the operators.

7.3 Shielding

Shielding requirements were specified for the UHTREX facility in accordance with the following criteria:

1. Operating and control areas and, in general, all areas outside the secondary containment regions should be fully accessible during normal full power operation of the reactor. The design radiation level for these regions is 1 mrem/hr. In addition, the control room proper is designated as a refuge for operating personnel in the event of a major disaster, such as the rupture of the primary containment system and consequent release of fission products to the secondary containment system. The shielding is specified to limit the dose rate in the control room to about normal working tolerance level.
2. Areas inside the secondary containment system will be accessible normally only when the reactor is shut down and the primary coolant loop has been purged and refilled with clean helium at low pressure. This operation is expected to require about 24 hr. After extended operation, radiation levels in most of these areas may be above the 40-hr week tolerance level, and working times will have to be adjusted accordingly. Radiation levels in the reactor cell proper will probably be so high as to preclude access to that room even after long cooling times.

7.3.1 Reactor and Gas Cleanup Rooms

The calculated fluxes incident on the reactor biological shield, together with some of the values used in specifying shield thicknesses, are tabulated in Table 7.3.1.1.

TABLE 7.3.1.1

DATA FOR SPECIFICATION OF SHIELD THICKNESSES

Source	Flux Incident on Bulk Shield (per cm ² -sec)	Decade Length in Ordinary Concrete (in.)	Decade Length in Magnetite Concrete (in.)	Decade Length in Tuff (in.)
Thermal neutrons	2.2×10^{10}	6.4	2.2	30
Non-thermal neutrons	2.4×10^9	12	8.7	23.4
Gamma rays $E_{av} = 2 \text{ Mev}$	3.1×10^{10}	12	8	26
Gamma rays $E_{av} = 8 \text{ Mev}$	2.0×10^{10}	18	12.5	48

Between the reactor room and the control room there are shield walls of 3.5 ft of magnetite concrete and 7 ft of ordinary concrete. Further protection for the control room is provided by the 4-ft magnetite concrete wall of the fuel transfer cell and transfer lock, which are situated between the control room and the biological shield. The change rooms and operating floor hallway are shielded from the reactor by 5 ft of magnetite concrete and 5 ft of ordinary concrete. The remote maintenance corridor, to which full-time access is not required, is shielded from the reactor by 5 ft of magnetite concrete and 2.5 ft of ordinary concrete. The exterior walls of the reactor room are of ordinary concrete, 5-ft thick. Because the reactor room is below grade, advantage is taken of the shielding properties of the surrounding earth and rock (tuff).

The components of the gas cleanup system, located in the gas cleanup room, are expected to accumulate a significant portion of the fission products generated in the reactor. The intensity of this source is difficult to determine. However, the position of the gas cleanup cell between two layers of the reactor biological shield provides a minimum of 2 ft of magnetite concrete and 5 ft of ordinary concrete between this source and the operating areas.

The top shield for the reactor room and the gas cleanup room consists of 5 ft of magnetite concrete, sufficient to maintain near-tolerance levels inside the secondary containment above-grade structure when the reactor is shut down. Radiation levels in areas immediately outside the structure will be held near tolerance values during reactor operation by the ordinary concrete walls of the structure, 1.5-ft thick on three sides and 3-ft thick on the side facing the control room. Control rod guide tubes which protrude through the top shield have necessitated a number of penetrations which will be shielded individually by stacked collars (alternate layers of steel and phenolic) to reduce neutron streaming.

7.3.2 Upper Gas Cleanup Piping Room

As noted above, this room is accessible only during shutdown. A 3-ft magnetite concrete shield between this area and the gas cleanup room, and a total of 5 ft of magnetite concrete between this room and the reactor vessel, should hold radiation levels low enough to permit reasonable working times even after long-term operation.

7.3.3 Reactor Drive Room and Fuel Discharge Room

The bulk shielding between these areas and the core consists of 4 ft of magnetite concrete. However, a 6-in. diam diagonal penetration for the fuel discharge tube will require special shielding to reduce streaming into the fuel discharge room.

7.3.4 Fuel Charging Room

Streaming through the large fuel loader slot will seriously restrict working times in this area.

7.3.5 Fuel Transfer Cell and Lock

Accessibility of these rooms to personnel should be limited by local sources within the rooms rather than by reactor power level. The 4-ft magnetite concrete walls will permit storage of at least 100 spent fuel elements.

7.3.6 Secondary Containment Structure (Above Grade)

Specifications for this structure call for ordinary concrete walls, 1.5-ft thick on three sides, and a 2-ft thick roof. In order to provide additional shielding for the control room and approach route in the case of the maximum credible accident, the fourth side is to be of ordinary concrete, 3-ft thick.

7.4 Health Physics Instrumentation

Gamma radiation levels at selected locations within and near the building are measured by a remote-area monitoring system. Local readout and alarm devices are installed at most stations, and all stations transmit data to level indicators and alarms located in the control room. A beta-gamma monitoring system continually surveys contaminants filtered from the air at automatic sampling stations. Beta-gamma levels at the stations are indicated in the control room on instruments with alarm capability. Filters from additional air sampling stations are removed routinely to be scanned for activity.

Effluents to be discharged to the stack are monitored by both types of continuous readout instruments, located in the secondary containment and hot cell ventilation systems.

Portal monitors are placed in suitable locations for general personnel monitoring. Individual monitoring and dosimetry are conducted in accordance with LASL radiation safety policies.¹⁷

7.5 Ventilation

A schematic air flow diagram for the building is presented in Fig. 7.5.1. There are four independent ventilation systems, one each for the following regions: the secondary containment enclosure, the fuel handling and gas sampling cells and their adjacent operating areas, the change rooms and other areas that might become contaminated during normal operations, and the

Fig. 7.5.1 Schematic building ventilation flow diagram.

control room and other clean areas. Except in the secondary containment enclosure and the control room, ventilation air is not recirculated. All intake air passes through filters designed to remove particles as small as 20 microns in diameter. During normal reactor operation, exhaust air from regions that might be contaminated is passed through absolute and activated carbon filters and vented through a 100-ft stack at a flow rate of 21,000 cfm.

7.5.1 Secondary Containment System

For ventilation of the secondary containment during reactor operation, an in-line blower moves the air in a closed cycle through a system designed to withstand 7.5 psi internal pressure. This system, an extension of the secondary containment, is depicted in Fig. 7.5.1.1. The full air flow, 10,000 cfm, passes through absolute and activated carbon filters and heating and cooling coils. Pressure inside the containment enclosure is maintained slightly below atmospheric by a small, continuous bleed of filtered air to the stack. Within the rooms, air flows from the regions least likely to be contaminated to those where contamination is more likely and then returns to the filters before recirculation. The flow rate is sufficient to provide 15 air changes per hour in the regions where equipment contains fission products.

When the reactor is shut down and the coolant and cleanup loops are at low pressure, personnel may enter the secondary containment enclosure. At that time, the ventilation system brings in fresh air, draws it through the 20-micron filters, passes the filtered air once through the rooms, and exhausts it through the absolute and carbon filters to the stack.

The exhaust filters in this system have a high probability for becoming contaminated should leaks develop in the helium systems. Therefore, these filters are located in a separate shielded pit outside the building.

After a day's operation of the reactor at design power, the A^{41} activity within the secondary containment will have reached an equilibrium level of approximately nine curies or an average concentration of $2 \times 10^{-3} \text{ C/m}^3$. As is shown in Appendix D, personnel would not be prevented from entering the

Fig. 7.5.1.1 Schematic ventilation flow diagram for secondary containment enclosure.

secondary containment by this source density. However, several complete changes of the secondary containment air will normally be made before personnel go in. Furthermore, access to the secondary containment is prohibited until the absolute pressures in the coolant and purification loops have been reduced to less than two atmospheres, a procedure that takes a minimum of about 24 hr following operation at design pressure.

During the period of venting and purging of the secondary containment enclosure before personnel entry, the A⁴¹ discharged to the stack produces in adjacent, uncontrolled areas maximum dose rates that are well within tolerance levels.

7.5.2 Other Ventilating Systems

The remaining three ventilating systems take in fresh air through a common 20-micron filter and move the air to various parts of the building through individual blowers.

The cell system ventilates the cell operating areas and the fuel transfer lock with a once-through flow of air that vents to the stack through sets of absolute and carbon filters. When cell doors are opened, the ventilation system provides an increased exhaust rate through the cells to prevent the movement of airborne contaminants through the cell doors to the operating area.

Ventilating air makes a single pass through the change room system to absolute filters and exhausts to the stack.

In the clean area ventilation system, the air is distributed in a more complex pattern. The major part of the flow passes through the offices and equipment rooms to rooftop vents. Air that ventilates the health office, operations office, and the adjacent hall passes into the change room system.

The control room is air conditioned to maintain close temperature control of the electronic racks. 85% of the air is recirculated through the electronic racks, and 15% is fresh air to provide proper room ventilation. The

control room also has a second, independent, system, for emergency use, that prevents in-flow from contaminated regions. Standby power for the blower in this auxiliary system is provided by the emergency power generator. Air is drawn in from outside the building, passes through an absolute filter, ventilates the control room, and vents directly to the atmosphere.

7.6 Utilities

Water, natural gas, and electricity are supplied from the Los Alamos mains, sanitary liquid waste flows to a local septic tank system, and contaminated liquid waste is pumped to the LASL contaminated waste disposal plant. Schematic diagrams of site utility and waste disposal systems appear in Fig. 7.6.1 through Fig. 7.6.4.

To provide reliable electrical power, the building is supplied from the Los Alamos power distribution center over two separately-routed, high tension lines. Normally, the building load is divided between the two lines, but either line is capable of handling the full load, and automatic switch gear transfers the load to the active line should one line fail. Should both lines fail, a 30-kw, diesel-driven generator provides emergency power for nuclear instruments, reactor controls, fueling operations, control room ventilation and pressurization, the secondary containment bleed blower, the pump that circulates water in the cooling panels, and a limited amount of lighting. This generator is electrically driven when normal power is available, and has sufficient inertia to provide continuous power during automatic switchover to the diesel drive. Should the power lines and the diesel generator fail, a battery-powered generator provides electrical power for the nuclear instruments and for reactor control.

Fig. 7.6.1 Schematic diagram of electrical power distribution system.

Fig. 7.6.2 Schematic waste disposal flow diagram.

Fig. 7.6.3 Schematic water flow diagram.

Fig. 7.6.4 Schematic diagram of cooling panel water system.

8. OPERATIONS

8.1 Administration and Planning

The startup and operation will be performed under the supervision of an UHTREX Planning Committee. The Committee will be selected from those members of the staff who have been intimately associated with the design and construction of the reactor; thus, the UHTREX Planning Committee will constitute a body possessing complete and detailed familiarity with every component in the reactor system. It will be the responsibility of the Committee to review and approve in detail all plans for starting up and operating the reactor, and to review and assess the results from the experiment as it progresses.

One member of the Committee will be selected to supervise the execution of the experimental programs. **This member usually will submit** operational proposals to the Committee for consideration; however, any member of the Committee may propose experiments or modifications to the program. Particularly during the startup phase of the program, those members of the staff having a special interest in the experiments currently in progress will serve as members of the reactor operating crew.

The operations and procedures devised by the UHTREX Planning Committee will be subject to review and modification by the LASL Reactor Committee.

8.2 Procedures

Safety regulations, operational procedures, and check lists will be formulated prior to the loading of any fuel into the core. Proposed regulations and procedures must be reviewed and approved by the UHTREX Planning Committee before becoming effective.

9. HAZARDS

The design of UHTREX includes both characteristics that mitigate hazards common to other reactors and characteristics that create unusual hazards. In the preceding descriptions of components, safety aspects of the designs have been mentioned in context. Below, the principal safety features of the over-all design are reviewed briefly, and the potential hazards are discussed in greater detail.

Hazards can arise from two different sets of circumstances: the inadvertent commission of operational errors, and the unforeseen mechanical failure of major system components. All UHTREX systems are designed to withstand the effects of operational errors until corrective actions are taken, usually automatically. The effects of credible mechanical failures are limited by design to acceptable proportions, or are controlled within tolerable limits by automatically actuated devices. The maximum credible accident, whose consequences are described in Chapter 10, is the result of a mechanical failure described in Section 9.3.1.

9.1 Safety in the Design

9.1.1 Reactor

Many hazards associated with conventional reactor fuel elements do not apply in UHTREX. The dispersion of uranium in a refractory matrix precludes a nuclear explosion due to rapid critical re-assembly following a core meltdown.

Small, discrete units of fuel, one added and another discharged simultaneously, make impossible the sudden addition of reactivity through failure of fuel-loading mechanisms or the inadvertent addition of excess reactivity in massive quantities. Fuel damage is immaterial to safety, in the usual sense, inside an already contaminated loop.

A solid moderator that encloses the fuel and serves as a structural part of the reactor limits the hazards that can arise from a sudden change in moderation. No experimental channels complicate the nuclear aspects of core design.

The ability to load and unload fuel at power makes possible the operation of the reactor, at any power level, with the control rods completely withdrawn. During full power operation, the rods are raised from the core into the upper, cooler part of the reactor. There, the rods do not need to be cooled to prevent thermal distortion and cannot initiate an excursion caused by their abrupt withdrawal. Since there is no need for coolant flow around the rods, buoyancy effects do not influence rod insertion or withdrawal.

All mechanical components that enter the reactor pressure vessel, and the mechanisms that drive those components, are enclosed in extensions of the pressure vessel. There are no penetrations of moving parts through seals. Junctions with the pressure vessel are made with welds, test-proved Conoseal couplings, or bolted flanges with seal welds.

9.1.2 Control Systems

Nuclear safety circuits initiate automatic rundown when pre-set conditions occur; operating procedures are limited by interlocks, and an ample number of annunciators give audible and visual indication of abnormal conditions. All control and analytical instrument components have been tested in previous installations; UHTREX operations do not depend on new or unique instruments. The control room instruments have indication and recording capabilities adequate for diagnosis of incidents to be expected in an experimental reactor.

Control system designs for auxiliaries, including the gas cleanup system, have the general safety characteristics of the nuclear controls described above.

Health physics requirements are met with ample instrumental systems tied into pertinent alarm systems.

9.1.3 Coolant Loops

Direct junction of the reactor vessel to the recuperator vessel and of the recuperator vessel to the main heat exchanger shell creates essentially one pressure vessel for these three major components, lowers thermal stresses among them, reduces the need for high temperature piping, and minimizes the number of joints in the loop. Again, all joints are welds, seal-welded bolted flanges, or Conoseals. A minimum number of sealed stem valves is used.

To compensate for operational uncertainties, recuperator heat exchange conditions can be varied with a simple by-pass. The main heat exchanger is designed to withstand multiple exposures to disaster conditions. Its design minimizes internal thermal stresses. Blowers are canned and have gas bearings to eliminate shaft seals. Primary loop blowers are duplicated, and all blowers have back-up power supplies.

Secondary containment is provided by the secondary loop, built to the same stringent specifications as the primary loop, and by a gas-tight portion of the building. The secondary loop dissipates its heat to an ever-present supply of air. There is no danger of a disastrous injection of water into the primary loop through failure of a coolant system heat exchanger.

Much of the equipment located in parts of the building that are inaccessible during reactor operation can be inspected by means of the television cameras mounted on MINOTAUR, the remote maintenance machine. MINOTAUR can be used to make minor repairs or adjustments found to be necessary on inspection, or to verify the existence of serious difficulties, indicated on control instruments, that justify a reactor shutdown.

9.1.4 Auxiliaries

Operation of the reactor is not strongly dependent on the gas cleanup system, whose function is to limit rather than to eliminate the concentrations of mobile oxidants and fission products in the primary loop. Under some conditions, the reactor can operate indefinitely without the gas cleanup system; under others, the reactor must be shut down within a reasonable period of time after loss of the cleanup system.

Ample capacity exists in the helium storage system to contain all of the coolant helium inventory after an emergency shutdown. Compressors are duplicated to insure a means for moving helium into storage.

All utilities have a reserve capability. Electrical power can be supplied from two independent lines or can be generated on the site. Cooling water, for auxiliary systems that include pumps, can flow under main pressure or by gravity at reduced flow rates.

9.2 Effects of Operational Errors

If UHTREX is to supply useful information, means must be provided for the operation of the experiment under a variety of conditions. This introduces the possibility that, inadvertently, a series of operations may produce an accident. Some of these accidents and their consequences are discussed below.

9.2.1 Insertion of Excess Reactivity

No mechanisms exist in UHTREX which can create a rapid rate of increase of reactivity (say $\sim 1\$/\text{sec}$). The ingredients in the core include graphite, fuel, fission product poisons, helium, control rods, and heat. The solid materials normally run at temperatures thousands of degrees centigrade below their melting or vaporization temperatures. Explosive ejection of control rods, or violent material relocation, due to the rapid expansion of a material undergoing a change of state, could only take place in UHTREX after some very large reactivity input, not before.

Discounting vaporization, a redistribution of the core materials cannot take place rapidly since both the fuel and poisons, even in the presence of large thermal gradients, can undergo only slow diffusion processes. A redistribution of the fuel can effect an increase in k only by spreading the fuel more homogeneously through the core. Perfect homogenization of the fuel would produce an increase in k of only 1 to 2% Δk . If, for example, fuel elements were to fall out of their channels and stick in the central plug, k would decrease since such a condition would ultimately lead to a less homogeneous fuel distribution.

The fission product poisons, if they were completely retained in the fuel elements at 1350°C, would represent a total inventory of at most 3% in k . While, at 2500°C, it is estimated that all the poisons would ultimately diffuse out of the fuel elements, the rate of diffusion is slow, even at this elevated temperature.

Thermal changes, likewise, cannot give rise to large reactivity change rates. The case of a loss of coolant flow leads to a relatively mild excursion as was shown in Section 3.5.8. A sudden by-passing of the recuperator would produce a transient of the same order as the loss of coolant flow, but in the opposite direction.

In all of the above, in order to obtain any effects at all, it must be assumed that the automatic rundown circuits do not work. A large transient would be induced if several of the control rods were withdrawn simultaneously at their maximum withdrawal rate. This is not a credible accident, but would represent, rather, an act of sabotage requiring rewiring of control circuits and jumping of rundown trips.

The effects of this case are assessed primarily to illustrate the relative indestructibility of the UHTREX core. For the analysis, a ramp addition rate of 10¢/sec is postulated, and the power is assumed to be 1 watt when the reactor becomes prompt critical. From the equation¹⁸

$$n_{\max} = \bar{n} \ln (n_{\max}/n_0)$$

a peak power of 2150 Mw is estimated. An average power of 100 Mw is

calculated from

$$\bar{n} = \frac{B C_f}{\alpha}$$

where

$$B = 0.007 \Delta k/\text{sec} = \text{ramp}$$

$$\alpha = -2.1 \times 10^{-6} \Delta k/^{\circ}\text{C} = \text{fuel temperature coefficient}$$

$$C_f = 0.3 \text{ Mw-sec}/^{\circ}\text{C} = \text{fuel heat capacity}$$

The ratio C_f/α is the same whether the heat is transferred to the moderator or the fuel, since both factors α and C_f are about 40 times larger in the moderator than in the fuel.

The duration of the initial excursion is 6.5 sec from

$$\tau = 2\sqrt{\frac{2\ell}{B}} \ln(n_{\text{max}}/n_0)$$

If a constant input power equal to $\bar{n}$ is assumed, the fuel temperature rise is

$$\Delta T_f \cong \bar{n} \left[\frac{\tau}{C_m} + \frac{1}{k} \left(1 - e^{-k\tau/C_f} \right) \right] = 2600^{\circ}\text{C}$$

the moderator temperature rise is

$$\Delta T_m = \bar{n} \left[\frac{\tau}{C_m} + \frac{C_f}{k C_m} \left(e^{-k\tau/C_f} - 1 \right) \right] = 180^{\circ}\text{C}$$

where

$$C_m = 12 \text{ Mw sec}/^{\circ}\text{C} = \text{moderator heat capacity}$$

$$k = 0.03 \text{ Mw}/^{\circ}\text{C} = \text{fuel to moderator heat transfer coefficient for fuel to moderator } \Delta T \sim 700^{\circ}\text{C}$$

While k is assumed constant in the calculations, it actually increases rapidly with temperature, since, for large fuel to moderator temperature differences, the heat would be transferred predominantly by radiation. Hence, the fuel temperature rise should be less than that calculated, but still one might expect the stresses in the fuel to produce cracking of the elements. Cracking would be tolerable as long as passageways for the fuel did not become clogged.

Prolonged application of this "incredible" ramp would undoubtedly lead to damage to the containment in the primary loop.

9.2.2 Coolant Blower Shutdowns

Heat moves from the core to the atmosphere in flowing helium that is circulated by electrically driven blowers and finally cooled with forced-draft air. When the operation of a single blower or fan is unwittingly interrupted, heat transfer throughout the coolant systems is affected. The effects and their consequences are discussed below for the primary loop blower, the secondary loop blower, and the heat dump fan.

Primary Loop Blower - A sudden interruption of the primary coolant flow can occur if power to the active blower is switched off or if either of its isolation valves is closed. Loss of flow causes a temperature transient in the core, with the nuclear consequences that are described in Section 3.5.8. Effects throughout the remainder of the primary loop, with or without a consequent rundown of the control rods, can be accommodated without equipment rupture.

In the recuperator, after flow interruption, the bulk temperature of the graphite heat exchanger block equilibrates at about 1500°F. The insulation around the block is sufficient to protect the shell against overheating at this internal temperature. None of the parts of the main heat exchanger depends on primary flow for cooling; the effect in that part of the loop is negligible. The blower coasts, unharmed, after a power interruption, but the sudden closing of a valve may cause the blower to surge, with consequent failure of the gas bearings. Although the bearings are damaged after an upset caused by surging, the blower case remains intact.

The reactor vessel, which does not depend on the helium stream for cooling, suffers no ill effects even if there is no rundown. There is no disturbance of the uncooled core support bearing. The initial temperature rise in the core forces a temporary 20% increase in heat transfer through the shell, but this greater heat flow can be achieved with an insignificant rise in pressure vessel temperatures. If a rundown is initiated by the flow loss, (n, γ) heating of the vessel wall diminishes rapidly, subtracting about 20% of the vessel wall heat dissipation load.

The most serious consequence of a primary loop blower shutdown, then, is effected by the closure of a blower isolation valve. To prevent the destruction of blower bearings, design and procedure limitations are established. The valves do not have remotely actuated operators; MINOTAUR turns the valve operator when a new setting is necessary. Changes in valve settings are made only when the reactor is shut down.

Secondary Loop Blower - The two means for stopping secondary loop flow are interruption of power to the blower and closure of either isolation valve. The principal effect of a flow stoppage is a rise in the internal temperature of the main heat exchanger, which continues to receive heat from the flowing primary loop coolant but cannot dissipate heat to the stagnant secondary loop. After secondary loop flow has stopped, several minutes elapse before the internal temperature of the main heat exchanger rises to its design limit. When the temperature rise starts, multiple temperature sensors located at the exchanger outlet trigger circuits that disconnect the power to the primary loop blower.

Another possible consequence of a stoppage caused by valve closure is the destruction of the blower bearings. Unlike the primary loop valves, the isolation valves in the secondary loop must be capable of rapid, remotely controlled closure initiated by the indication of a sudden coolant loss from the secondary loop. To prevent destruction of the blower bearings, a by-pass around the blower is opened by the controller that closes the isolation valves.

Heat Dump Fan - A general, moderately swift rise in temperature throughout the secondary loop is the immediate effect of a shutdown of the heat dump fan. The highly unlikely consequence is the rupture of some component when it reaches its yield temperature.

The heat capacity of the secondary loop components resists a rapid temperature rise and allows corrective measures to be taken in time to prevent ruptures. Detection of a temperature rise is assured, because careful temperature control in the secondary loop is a necessity for stable reactor operation. Time in which to restart the fan or take other corrective measures is provided for in the design of the heat dump.

9.2.3 Venting of Coolant Helium

For operating flexibility and safety, two sets of vent valves are installed in each coolant loop. One set provides a controlled pressure regulation and adjustment capability with widely variable vent flow rates, limited by the capacity of the high pressure compressor. The second set consists of blow-down valves, automatically opened in emergency, which dump coolant helium to the low pressure storage tanks at flow rates limited only by the pressure drop in the blow-down system. Either set can be opened accidentally, through operator error or control malfunctions.

Primary Loop - The sudden venting of coolant from the primary loop has two critical effects: a radical change in heat transfer rates and an abrupt increase in pressure difference across the tube sheets in the main heat exchanger. The thermal effects in the loop are similar to those experienced in the loss of coolant flow and are tolerable. Differential pressure effects in the main heat exchanger are accommodated in its design. Although the pressure differential imposes increased stresses, the decreased rate of heat transfer from the core allows a mitigating drop in temperature. A third effect of venting might be the displacement of fuel elements in the core as a result of increased coolant velocity through the fuel channels, but this effect is

prevented by design limitations on the maximum rate of venting through wide-open valves. The consequences of a sudden venting of the primary loop are not disastrous.

The possibility of an accidental venting through the pressure-regulating valves is remote because it requires the resetting or failure of two or three controlled valves in close sequence. A remotely actuated and controlled valve at the outlet of the adsorption beds in the gas cleanup system ordinarily vents primary loop helium to maintain the level set on a pressure controller in the control room. This valve and controller operate in conjunction with, but independently of, other controlled valves that add helium when the primary loop pressure is below control level. Immoderate pressure rises are prevented by redundant pressure controllers and rupture disks in the new helium valve manifold. If the gas cleanup system is inoperative, helium can be vented through the blow-down system.

Duplicate blow-down valves are designed to vent the primary loop automatically in case of rapid pressure rise or in response to other emergency situations that are included in the safety circuitry logic. One of the valves may be actuated by a manual control as a normal operational procedure. Ultimate blow-down capability is provided by a by-pass closed with a rupture disk. Rapid, but limited, venting can be effected unintentionally by failure of either valve or the rupture disk and by unwitting operation of the manual valve control. The probability of accidental venting is greater through the blow-down system, therefore, than through the pressure-control valves.

Secondary Loop - The arguments that are applied to the primary loop for the cause and effect of accidental venting can be extended to the secondary loop with the additional and overriding complication that the thermal effect amplifies rather than mitigates the stresses in the main heat exchanger. Design features of the exchanger make it capable of withstanding a loss of the secondary loop coolant under the conditions described in Section 4.2.3. A reduction of primary loop flow and pressure, specified in Table 4.2.3.1, is automatically initiated by sensors that detect a drop in secondary loop

pressure. If the secondary loop pressure falls more rapidly than that in the primary loop, differential pressure sensors initiate closure of the isolation valves in the secondary loop.

Rapid decompression of the secondary loop through the pressure control or blow-down systems is made almost impossible by their design. **Vent rate** capability is strictly limited to match that of the much more voluminous primary loop. Multiple interlocks between pressure control systems assure that a slight positive pressure differential is maintained in the secondary loop. The helium required to maintain the differential is supplied directly from cylinders independent of the remainder of the helium storage and pressure-regulation system. Should all these design conditions and interlocks fail to halt a rapid decompression, safety circuits close the isolation valves when the pressure in the secondary loop falls below the primary loop pressure.

9.2.4 Opening of Recuperator By-Pass

Heat exchange conditions within the primary loop are adjusted with a by-pass flow of coolant around the cool side of the recuperator. The flow is regulated by a throttling valve in the by-pass line. Valve position, normally partially open, is set by a controller that responds to changes in the inlet gas temperature at the main heat exchanger. **Through a controller malfunction or an operator error**, the valve can be opened wide in a few seconds; the rapid response is essential in normal operation if the valve is to perform its fine-trimming function. **The effects of a sudden opening of the nearly closed valve** are the admission of a 600°F stream of helium to the bottom of the reactor and the loss of the coolant in the recuperator. The consequences of both effects are limited by automatic controls.

A consequence inside the core of such an accident is a sudden rise in power due to cooling of the fuel elements, amplified by a slower cooling of the moderator. Any power rise is sensed by multiple detectors in nuclear control circuits and, if the rise continues, causes a rundown. Under moderate conditions, when the by-pass valve is nearly open, the rise is slow enough to be detected by an operator and counteracted with control rod movement.

When the cool side of the recuperator is completely by-passed, heat transferred from the hot side helium stream accumulates in the graphite heat exchanger body. As this one-tonne mass heats up, the outlet coolant temperature rises above 1400°F. Within several minutes after the total by-pass occurs, the metal components in the transition area between the recuperator and the main heat exchanger reach yield temperature. To prevent equipment damage, a high-temperature safety circuit decouples the blower. This preventive action is taken regardless of reactor conditions; a rundown will not prevent over-heating in the recuperator.

9.2.5 Malfunctions in the Gas Cleanup System

Coupling is loose between the primary loop and the gas cleanup system, which processes helium at a flow rate only 1% of the primary loop flow rate. When a malfunction occurs in the gas cleanup system, there is ample time to take corrective action and prevent the transmission of effects to the primary loop. To affect the primary loop, a malfunction or operator error must result in the rapid injection of a harmful material into the circulated helium stream at a point where subsequent removal is impossible. Such an accident is conceivable, but incredible, when the duplication of temperature, analytical and sensor elements and their interlocks is considered.

9.3 Effects of Component Failures

The possibility exists in any complex system that the mechanical failure of a single component may cause radical effects throughout the system. In UHTREX, the more vulnerable sub-systems are the coolant loops and the fuel-handling system. Failures, not necessarily credible, in these components are postulated below and the effects of the failures are discussed.

9.3.1 Rupture of the Primary Coolant Loop

Contaminated helium, released by a rupture of a primary loop component, is confined inside the secondary containment under the most rigorous set of credible conditions. For the analysis presented here, the postulated sequence of events is failure of the reactor vessel, quick release of primary loop coolant into the reactor room, reaction of the core graphite with the air in the containment structure, and dissipation of sensible and decay heat of the reactor to its surroundings. To increase the severity of the accident, the conditions include an improbable interruption of water flow to the cooling panels in the reactor room. Mobile fission products carried in the released helium create a radiation field around the superstructure. The consequences of this, the maximum credible accident, are analyzed in Section 10.

For the most severe case of rupture in the primary coolant loop, a ductile failure in the reactor pressure vessel is assumed to form a crack that expands into a hole. Coolant helium expands into the reactor-recuperator room through the hole, calculated to be elliptical in shape with an equivalent diameter of 3.5 in.¹⁹ The subsequent pressure rise in the room is plotted in Fig. 9.3.1.1. From the reactor room, the liberated helium and its load of mobile fission products move into the remainder of the secondary containment. The pressure inside the room rises rapidly to 2.3 psig after 0.6 sec when flows into and out of the room become equal. The internal force due to pressure differential is insufficient, by a factor of almost 3, to raise the loose shielding slabs that cover the room.

Meanwhile, the pressure inside the primary loop drops rapidly, within 1 sec, to 400 psi. The pressure drop initiates, 1 sec after the rupture, a reactor rundown, an electrical decoupling of the primary loop blower, and the closure of valves in the ventilation system bleed from the secondary containment. Immediately after rupture, a weak shock front forms in the flowing helium outside the reactor vessel close to the hole. There is no shock due to flow discontinuity inside the vessel.

Fig. 9.3.1.1 Gas pressure inside reactor room after rupture of reactor vessel.

About 10 sec after the rupture, the pressure inside and outside the reactor becomes equal, and the coasting blower begins to introduce a mixture of air and helium into the hot reactor core and recuperator. There, the oxygen reacts with graphite to form carbon monoxide at a reaction rate limited by the mass transfer of oxygen through the gas film on the graphite surfaces. Mass transfer decreases as the blower speed falls, until the rate is controlled by natural convection. The initial rate of combustion in the core produces heat much faster than convection carries the heat away. Therefore, the graphite temperature rises to 3200°F about one hour after rupture, when heat production and removal rates are equal and half of the available oxygen has reacted. The combustion is assumed to continue until all the available oxygen is consumed, three hours later.

The temperature and, consequently, the pressure of the gas inside the secondary containment are driven upward by heat convected and radiated from the hot surfaces of the reactor and auxiliary parts of the coolant system and by heat carried from the burning interior of the reactor in gas circulated initially by the blower and later by natural convection. However, a limitation on the temperature and resultant pressure rise is soon established through the increasing rate of heat transfer by convection from the gas to the massive concrete walls of the building. One minute after rupture, heat transfer into and out of the enclosed gas are equal, at a gas temperature of about 150°F. However, the gas pressure continues to rise as the combustion reaction, $2\text{ C} + \text{O}_2 \rightarrow 2\text{ CO}$, produces two moles of carbon monoxide from one of oxygen. Figure 9.3.1.2 shows the calculated pressure rise inside the secondary containment enclosure. Internal pressure rises to a maximum of 4.3 psi above atmospheric about 3 hr after rupture. The enclosure is designed to withstand without damage an internal pressure of 7.5 psi above atmospheric.

The reactor and primary loop subsequently lose sensible and decay heat to the building walls until an equilibrium is reached after about one month. The equilibrium conditions for the enclosed gas are 128°F and 4.3 psig. About 4.5×10^7 Btu of sensible heat, heat of reaction, and decay heat are transferred to 5000 tons of concrete.

Fig. 9.3.1.2 Gas pressure inside secondary containment after rupture of reactor vessel.

In the above analysis, it was assumed that the water flow into the reactor room cooling panels was stopped at the time of rupture. Under the assumed conditions, heat radiated by the reactor raises the water in the panels to its boiling point in about 30 min. Then relief valves open in the water piping manifold, and the water flashes down the drain. After the water is gone, the panel temperatures rise until heat input by radiation from the reactor equals the sum of heat outputs to the gas by convection and to the concrete walls by radiation. The equilibrium occurs 2 hr after rupture when the reactor vessel surface temperature is 550°F, the panel temperature is 225°F, and the concrete wall temperature is 100°F.

The net result of the whole chain of events is an enclosure filled with a mixture of contaminated gas at a pressure of 4.3 psig and a temperature of about 130°F. The consequences of this accident are discussed in Section 10.

9.3.2 Rupture of the Secondary Coolant Loop

Isolation valves close immediately to maintain secondary containment after a rupture of the secondary coolant loop. If the rupture occurs beyond the valves, the pressure regulating controller restores pressure in the shell side of the heat exchanger, now isolated from the remainder of the loop. The effects of this sequence of events are those described above for an operational loss of flow in the secondary loop. When a rupture occurs inside the isolation valves, the pressure in the main heat exchanger cannot be restored, and the effects are equivalent to those described for an emergency venting of the secondary loop.

Wherever the rupture occurs, safety circuits initiate action through the primary loop controls to prevent the overheating and consequent rupture of the main heat exchanger. If these circuits fail and a path develops between the primary and secondary loop inside the exchanger, the isolation valves maintain secondary containment.

Leakage in the main heat exchanger, by design, occurs inward from the secondary to the primary loop. Operational procedures require a higher pressure in the secondary loop; the pressure regulation system maintains

this differential. A differential pressure recorder sounds an alarm when the differential drops, and actuates the isolation valve operators when the differential becomes negative. As a backup, radiation monitors in the secondary loop also actuate the isolation valve operators when levels rise in the loop.

9.3.3 Failure in the Fuel Loader and Core Indexing Mechanisms

The principal effects of failures in the fuel loader and core indexing mechanisms are limitations on future reactor operations, rather than the creation of immediate hazards. When a component fails, in many cases, the ability to load or unload fuel is lost until the reactor can be shut down and depressurized to permit the correction of the failure. In other cases, all or part of the fueling capability is permanently lost.

For an example of a serious credible failure, the halting of a loading ram can be assumed to occur while the ram is moving a fuel element into the movable core. When the ram is partially inserted, it bridges the gap between the loader and the core, and a partially discharged element projects into the stationary core plug. The choice of what to do next lies between shutting down the reactor to effect immediate repairs or temporarily abandoning this loading position by rotating the core to shear off the protruding ram and element. If the choice is to proceed with loading, the graphite ram and the hollow graphite fuel element can be readily sheared with minimal damage to the core. The short piece of ram and the fragments of fuel element, in the worst case, fall into the plena at the edges of the core. There, the pieces either ride around the bottom of the plena as the core rotates, or are ground between the movable and stationary parts. The pieces do not have size or structural strength enough to bind the movable core.

Reparable components of the loader extend into rooms shielded from the reactor. In these rooms, repairs can be made under controlled conditions after a reactor shutdown and decay period. However, one vital component, the elevator inside the loader, is enclosed in an integral part of the pressure vessel. The elevator itself and all but the uppermost length of its actuating rod are inaccessible. For this reason, the elevator design is simple and conservative.

Three types of failures can happen in the core-indexing mechanism: drive breakdown, support bearing seizure, and sequence controller malfunction. If the drive stops while the core is in a position between index stations, loading operations are impossible until repairs are made under the conditions described above. A core support bearing seizure would result in permanent loss of loading capability. The sequence controller prevents the movement of the core unless all loading rams are withdrawn; a malfunction can allow movement of the core and consequent shearing of a loader ram and fuel element. If mobility of the core were lost, the reactor could continue to operate, but UHTREX experimental programs would be severely restricted.

A credible sequence of events that leads to the plugging of a fuel channel has not been postulated without structural failure of the core as a prerequisite. Heat transfer calculations show that the plugging of a channel does not, in itself, introduce additional hazards. When coolant flow is stopped in a single channel, heat generation continues and the fuel elements reach an equilibrium temperature of less than 3500°F. Heat flows from the elements to the moderator graphite by radiation and is conducted to adjacent active channels. In those channels, the additional heat flow from the moderator to the coolant reaches equilibrium after a 175°F rise in local moderator temperatures. A possible result of the increased temperature of the elements in the stagnant channel is some diffusion of uranium and additional fission products into the moderator. The local temperature increases in the moderator can be tolerated without mechanical failure of moderator parts.

9.3.4 Failure in the Fuel Handling System

Nuclear criticality hazards are eliminated in the fuel handling system design by two characteristics: the inability of the conveyors to handle more than one fuel element at a time, and the long operating cycle for the charging and removal of fuel. Failures in system components can limit future operation of the reactor, but the failures do not create nuclear criticality hazards. Hazardous conditions can arise, however, through failure of the gas locks at the entrance to the fuel loader or at the bottom of the reactor.

Each of the conveyors consists of a single vessel, large enough to contain a single fuel element or its fragments, moved inside a housing by continuous chain or cable drives. Electrical switching circuits and interlocks insure that the conveyors and their corresponding gas locks are operated in proper sequence. In the loading cycle, a single new element moves from a dry box in the fuel handling cell through conveyors and gas locks to the fuel loader. Progress of the element is indicated by sensors at each station where the element moves from one component to another. The fuel loader charges the single new element and concurrently discharges another single element. As this spent element is moved back to the fuel handling cell, its progress is indicated by sensors. Only when the cycle is completed by the visible arrival of the spent element in the cell, does a new loading cycle start. The cycle is not automatic; new fuel elements must be fed into the first conveyor with a hand-operated manipulator. If an equipment failure interrupts the loading cycle, no emergency is created, except in the case described below.

A series of ball valves, operated in proper sequence, closes the chambers in the gas locks that pass fuel elements into and out of the reactor pressure vessel. If, in an unlikely circumstance, a failure of the sequence controller should open concurrently all the valves in a gas lock, full coolant pressure would burst the flexible connector where the gas lock joins its adjacent conveyor. The result, provided that the valves could not be closed again, would be equivalent to that described above for rupture of the reactor vessel.

10. CONSEQUENCES OF THE MAXIMUM CREDIBLE ACCIDENT

The maximum credible accident, defined in Section 9.3.1, results in the release of the primary coolant helium to the secondary containment enclosure. After the accident, the enclosure is filled with a mixture of contaminated gas at a pressure of 4.3 psig and a temperature of about 130°F. The fission product activity carried in this gas mixture reaches the atmosphere through small leaks in the structure, which is designed for a maximum leak rate of 1% per day of the total enclosed volume. The significant hazards that are consequences of the maximum credible accident consist of direct gamma radiation from fission products inside the containment structure, internal radiation following inhalation of fission products in the airborne effluent, and external radiation from gamma-emitting fission products in the airborne effluent.

10.1 Basis for Activity Release

The fission product release to the secondary containment is estimated to consist of 100% of the inert gases, 10% of the halogens, and 1% of the remaining fission products. These figures are based on estimates of the activity, for each isotope, present in the primary coolant loop after one year's operation at 3 Mw. Taking into account the volatilities of the fission products, it is estimated that 40% of the total activity will be retained in the fuel elements. The lower temperatures, about 600°F, encountered by the helium coolant in its passage through the heat exchanger and blower are assumed to result in

deposition of approximately 30% of the total activity in this part of the system. Sensible, i.e., not necessarily continuous, operation of the gas cleanup system is expected to remove an additional 20% of the total activity. This latter mechanism should be especially effective in removing volatile fission products with half-lives greater than a few minutes.

10.2 Doses

Doses associated with airborne activity were computed* for two dose points: (1) the boundary of the 1000-ft radius exclusion zone, and (2) the nearest uncontrolled land, on which is situated a trailer court with approximately 200 residents, at a distance of one mile from the UHTREX site. A leak rate of 1% per day was assumed. Downwind concentrations of activity were obtained from Sutton's²⁰ equation for a continuous point source at ground level. The meteorological parameters used in the computation were appropriate to a typical inversion and a constant wind velocity of 5 mph.

10.2.1 Direct Gamma Radiation from the Secondary Containment Structure

Specifications for the secondary containment superstructure call for 1.5-ft thick ordinary concrete for the three side walls. In order to provide additional shielding for the control room and approach route, in the event of fission product release into the structure, the fourth side is 3-ft thick ordinary concrete. The roof is constructed of ordinary concrete 2-ft thick.

Shielding calculations were based on the release of fission product components postulated above. All fission products with yields and half-lives greater than 0.05% and 1 min respectively,²¹ were included in the dose rate computations. In order to simplify the calculations, the secondary containment superstructure was approximated by a non-absorbing cylinder, 50 ft in diameter and 37 ft high, with a uniformly distributed source. Dose points were located

* Appendix D, this report.

along a radial line at the mid-height of the cylinder. Attenuation of the gamma radiation in the air external to the cylinder was also included; build-up factors in air were assumed to be equal to those for water. Results of the calculation for 1.5 ft of ordinary concrete, and for various times after the fission product release, are shown in Fig. 10.2.1.1. For a 3-ft thick concrete side wall, the dose rates are reduced by at least a factor of 40.

10.2.2 Internal Doses

Calculations of internal doses to bone, thyroid, kidney, and muscle were based on the work reported by Burnett.²² Results of the calculations, presented in Appendix E, are shown in Table 10.2.2.1. The dose to the kidney and muscle can be neglected in comparison with that to the bone and thyroid.

TABLE 10.2.2.1
CALCULATED INTEGRATED DOSES (rem)

	Direct Radiation from Containment Structure (whole body)	Immersion (whole body)	Internal	
			Bone	Thyroid
2-hr dose at 1000 ft	0.1	0.4	1.7	5.9
24-hr dose at 1 mile	2×10^{-4}	0.1	1.4	4.1

10.2.3 Immersion Doses

The machine calculations from which the direct gamma-ray dose rates were obtained (Section 10.2.1) also provided information on the gamma-ray source intensity and effective energy for the specified fission product release. Figure 10.2.3.1 shows the variation of gamma source strength (source intensity x effective energy) with time after fission product release. Immersion doses were computed for a semi-infinite medium with a uniformly distributed volumetric source equal to the activity concentration at the point in question. Dilution factors were computed from Sutton's equation for a continuous point source. Results of the calculation (Appendix E) are shown in Table 10.2.2.1.

Fig. 10.2.1.1 Dose rate from radiation which penetrates 1.5-ft thick side walls of cylinder (height = 37 ft, diameter = 50 ft). Cylinder contains source (uniform distribution) resulting from a nuclear incident in which 100% of inert gases, 10% of halogens, and 1% of remaining fission products are released. T_s = time after fission product release.

Fig. 10.2.3.1 Time variation of gamma source strength in fission product release to secondary containment in maximum credible accident.

10.3 Consequences

10.3.1 Residential Areas

The calculational results listed in Table 10.2.2.1. are based on a postulated containment-shell leak rate of 1% per day. Based on an allowable accidental or emergency dose (AED) of 25 rem whole body, 25 rem to the bone, or 50 rem to the thyroid, and the assumption that residents at a distance of one mile could be evacuated within 24 hrs, the results indicate that a 7% per day containment shell leak rate would be tolerable

10.3.2 Technical Areas

The 1000-ft dose calculations apply to laboratory employees at nearby technical areas, all of which are located at distances greater than 1000 ft. Under the same considerations as those in Section 10.3.1, but assuming a 2-hr evacuation time, these results indicate that a 5% per day containment-shell leak rate would be tolerable.

10.4 Reduction of the Activity Release

During normal reactor operation, the air in the secondary containment enclosure is cooled and continuously recirculated through both absolute and activated charcoal filters. The locations of the blowers and filters render them relatively immune to damage in case of the maximum credible accident. Therefore, if the blowers and filters are in operating condition, decontamination of the gas mixture in the secondary containment will be possible. Significant and rapid reduction in the activity leak rate could be accomplished in this manner. In addition, controlled release of the filtered gas through the stack under favorable meteorological conditions might be an effective mechanism for reducing the internal overpressure and, hence, the uncontrolled release rate, without increasing the dose rate in sensitive areas.

APPENDIX A

METEOROLOGICAL DATA FOR LOS ALAMOS

TABLE A. 1
AVERAGE WIND DIRECTION, LOS ALAMOS
PERCENT OF TIME
1953 - 1957

SURFACE	Dir.	Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sep.	Oct.	Nov.	Dec.	Annual
	S	27.9	21.6	21.4	27.9	29.3	26.9	20.2	19.2	24.5	32.0	23.3	17.5	24.5
	SW	16.2	13.0	16.0	18.0	16.9	19.5	20.5	15.5	20.3	19.0	14.4	12.0	16.9
	W	13.8	16.0	21.3	21.7	18.5	22.3	16.6	21.0	20.2	15.2	15.3	15.5	18.7
	NW	7.7	10.5	10.9	8.2	7.4	7.3	8.5	8.5	7.3	7.2	9.4	10.6	6.9
	N	13.8	13.5	10.9	6.9	8.4	5.6	11.2	10.1	7.0	8.4	14.6	17.1	10.9
	NE	6.3	8.7	6.0	5.2	6.3	6.0	8.0	10.1	6.9	5.9	7.8	12.2	7.6
	E	7.7	8.9	8.5	7.8	8.9	7.6	9.2	10.1	8.2	6.6	8.7	7.4	8.6
	SE	6.6	7.8	5.0	4.3	4.3	4.8	5.8	5.5	5.6	5.7	6.5	7.8	5.9
* UPPER AIR	Dir.	Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sep.	Oct.	Nov.	Dec.	Annual
	S	11.9	5.7	10.7	13.8	18.4	30.6	17.7	26.7	22.8	20.9	8.9	6.5	16.2
	SW	18.6	16.1	22.1	14.4	25.0	20.2	24.2	20.0	18.8	22.1	15.0	15.9	19.4
	W	29.0	27.1	30.7	29.3	26.8	20.2	9.2	15.9	14.3	14.4	22.5	27.9	22.3
	NW	27.5	38.7	28.3	26.1	17.8	12.1	12.9	8.3	16.9	23.4	33.3	37.2	23.5
	N	8.6	9.2	4.5	8.6	7.0	8.9	6.9	10.1	8.6	8.6	15.0	9.4	8.8
	NE	1.1	0.3	0.0	2.8	0.0	3.2	4.6	3.7	4.0	1.6	2.7	0.3	2.0
	E	2.2	0.0	1.6	0.6	1.6	1.6	8.5	7.4	6.0	2.8	1.6	1.9	3.0
	SE	1.1	3.0	2.0	4.3	3.3	3.2	17.5	7.9	8.6	6.2	0.9	0.9	4.9

* About 1,000 ft above terrain; based on morning wind soundings at Albuquerque, data from three years of observation

TABLE A.2
AVERAGE SURFACE WIND SPEED, LOS ALAMOS
PERCENT OF TIME
1953 - 1957

MPH	Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sep.	Oct.	Nov.	Dec.	Annual
0	5.4	5.3	2.3	1.7	1.2	1.3	2.4	3.8	3.7	3.1	3.1	4.6	3.2
1-5	54.7	49.3	32.2	29.9	31.1	25.3	40.9	43.1	26.7	33.8	43.1	49.7	38.2
6-10	26.5	28.5	33.4	32.6	35.4	37.2	37.7	38.9	40.2	38.0	35.0	33.5	34.9
11-15	7.7	9.9	16.9	18.7	17.2	21.4	13.5	10.9	20.1	16.9	12.5	7.9	14.5
16-20	3.4	5.0	9.9	10.7	10.1	11.0	4.6	2.8	7.2	5.4	4.2	3.0	6.5
21-25	1.7	1.2	3.2	3.8	3.4	2.8	0.9	0.4	1.9	1.8	1.4	0.9	2.0
26-30	0.4	0.5	1.3	1.8	1.2	0.8	0	<0.1	0.2	0.8	0.6	0.3	0.6
31-35	0.1	0.2	0.5	0.5	0.3	0.2	0	0	<0.1	<0.1	<0.1	<0.1	0.1
36-40	<0.1	<0.1	0.2	0.1	0	0	0	0	0	<0.1	0	<0.1	<0.1
41-45	<0.1	<0.1	<0.1	<0.1	0	0	0	0	0	0	0	0	<0.1
46-50	0	0	<0.1	0	<0.1	0	0	0	0	0	0	0	<0.1
51-55	0	0	0	0	0	0	0	0	0	0	0	0	0
56-60	0	0	0	0	0	0	0	0	0	0	0	<0.1	<0.1

TABLE A. 3
PRECIPITATION, LOS ALAMOS
1952-1959*

	Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sep.	Oct.	Nov.	Dec.	Annual
Days of Precipitation													
Average	9.9	10.1	11.8	10.9	14.6	10.9	20.0	20.9	8.9	7.7	7.0	7.3	140.0
Maximum	18.0	13.0	23.0	16.0	23.0	18.0	24.0	27.0	19.0	20.0	14.0	11.0	
Minimum	5.0	4.0	2.0	3.0	7.0	2.0	16.0	16.0	3.0	0.0	2.0	3.0	
Days of Snow													
Average	9.6	9.6	11.1	6.0	1.6	0.0	0.0	0.0	0.0	1.1	5.9	6.7	51.6
Maximum	16.0	13.0	23.0	12.0	4.0	0.0	0.0	0.0	0.0	3.0	10.0	11.0	
Minimum	5.0	4.0	2.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.0	3.0	
Days of Thunderstorms													
Average	0.0	0.3	0.1	1.8	7.6	7.4	16.3	17.9	4.9	2.6	0.4	0.0	59.3
Maximum	0.0	1.0	1.0	4.0	12.0	15.0	25.0	23.0	11.0	8.0	2.0	0.0	
Minimum	0.0	0.0	0.0	0.0	3.0	2.0	7.0	16.0	1.0	1.0	0.0	0.0	
Snowfall (inches)													
Average	9.0	8.7	11.1	4.4	0.1	0.0	0.0	0.0	0.0	0.3	4.8	7.7	46.1
Maximum	18.8	18.0	35.5	33.6	4.0	0.0	0.0	0.0	0.0	0.3	34.5	11.5	100.0
Minimum	0.0	0.5	0.5	0.0	0.0	0.0	0.0	0.0	0.0	Tr	Tr	2.5	25.2
Precipitation (Since 1910) (inches)													
Average	0.90	0.66	0.95	1.07	1.46	1.34	3.24	3.64	1.95	1.44	0.67	0.83	18.15
Maximum	6.57	2.44	3.27	4.64	4.47	5.57	7.98	11.18	5.79	6.77	3.30	2.27	30.34
Minimum	0.0	Tr	Tr	0.0	0.0	0.0	0.72	0.51	Tr	0.0	0.0	0.05	6.80

* Except for last stub which covers the period 1910 to 1959.

TABLE A. 4
TEMPERATURES, LOS ALAMOS
1919-1958

	Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sep.	Oct.	Nov.	Dec.	Annual
Av Max	38	43	49	58	67	78	80	77	72	62	49	39	59
Av Min	16	21	25	33	42	50	54	53	47	36	26	18	35
Extreme Max	65	68	70	79	94	98	100	102	94	85	69	63	102
Extreme Min	-9	-14	-3	8	24	31	43	42	27	17	-4	-2	-14

APPENDIX B

GEOLOGY AND HYDROLOGY OF THE UHTREX SITE *

The following information was compiled from data collected during the course of ground-water investigations in cooperation with the Atomic Energy Commission.

The area selected for the UHTREX facility is about a half mile east-southeast of the site proposed for the new waste-treatment plant (TA-50), and is slightly more than 1,500 ft south-southeast of the present Ten-Site facilities (TA-35). The reactor facility is to be located south of Ten-Site Canyon, which is immediately south of the Ten-Site facilities and is a tributary of Mortandad Canyon. The topographic relief in the upper part of Ten-Site Canyon is about 70 feet in the vicinity of the proposed site. The area is on the western half of the Pajarito Plateau which has been dissected into a number of finger-like mesas by eastward flowing intermittent streams tributary to the Rio Grande. Mortandad Canyon and its tributaries drain the immediate area.

The proposed site of the UHTREX facility is in the area investigated recently by the Geological Survey for the proposed waste-treatment plant. This part of the Pajarito Plateau is capped by the Bandelier Tuff of Pleistocene Age. The Bandelier is a welded rhyolite tuff consisting of crystal fragments, tuff breccia, and pumiceous material. The Bandelier is about 700 ft thick near Ten-Site. The base of the Bandelier was encountered at an altitude of about 6,380 ft

* W. E. Hale, District Engineer, U.S. Geological Survey, Ground Water Branch, Albuquerque, New Mexico. Compiled by J. H. Abrahams and E. Baltz.

at Test well 8 in Mortandad Canyon, three-quarters of a mile east of the proposed UHTREX site which is at an altitude of about 7,160 ft. No bodies of water were found in the Bandelier at Test Well 8.

In the subsurface the Bandelier Tuff is underlain by the Puye Conglomerate of Pliocene (?) Age. The Puye consists of silt, sand, and gravel composed of pebbles to boulders of volcanic rocks. Basalt flows are interbedded with Puye sediments in the eastern part of the Pajarito Plateau. Test Well 8 was drilled through 575 ft of the Puye Conglomerate but did not reach the base of the formation. Water was not found in the upper part of the Puye, but was encountered in the lower part at a depth of about 985 ft below the floor of the canyon. The water rose about 25 ft in the well. This body of water, which is the main aquifer in the Los Alamos area, is at an altitude of about 5,960 ft in the vicinity of the proposed UHTREX facility.

Studies indicate that little of the precipitation falling on the mesa tops filters into the Bandelier Tuff where it is overlain by soil. Probably clay in the alluvium in the canyons also inhibits or retards infiltration into the Bandelier. Data concerning the movement of water through tuff are scanty, but because the upper part of the Bandelier is moderately welded and jointed, some infiltration may occur along joints when water flows on bare rock. These joints commonly extend across and along the canyons.

Generally there is flow in Ten-Site Canyon only during spring snow-melt or during summer thundershowers. Radioactive liquids discharged accidentally into Ten-Site Canyon would flow eastward across the Bandelier Tuff for several hundred yards, then presumably infiltrate into the alluvium in the lower part of Ten-Site Canyon and in Mortandad Canyon. However, if the accident occurred during spring runoff or during a heavy summer thundershower, the wastes could be carried as a "slug" into Mortandad Canyon or conceivably to the Rio Grande. Observation wells already constructed in Mortandad Canyon below the mouth of Ten-Site Canyon would serve to monitor the liquids in the alluvium.

APPENDIX C

UHTREX SOURCELESS STARTUP *

This appendix first presents a discussion of the neutron source strength in UHTREX fuel elements. For unburned fuel elements, the neutron source is shown to be mainly due to (α, n) reactions in which the α -particles come from U^{234} decay and produce neutrons when incident on C^{13} . The source strength should increase with burnup of the fuel. In UHTREX, the initial source strength will be about 10^3 neutrons/sec. It is then shown that, with this source strength, the departures of the neutron flux from its average value during a conservative startup are so slight as to be of no practical importance. Thus there is no necessity from these considerations for an additional source in the reactor.

1. Source Strength

UHTREX fuel elements are graphite impregnated with uranium which has been enriched to about 93.5% U^{235} , with 1.0% U^{234} , and 5.5% U^{238} . Possible neutron sources in such fuel elements arise from (α, n) reactions, spontaneous fission, and cosmic radiation. It is believed that the (α, n) reactions are by far the most important source.

Roberts²³ has reported neutrons resulting from bombardment of C by Po^{210} α -particles. These α -particles have 5.3 Mev energy and cannot produce

* Written by G. I. Bell, LASL, November 17, 1961.

neutrons from reactions with C^{12} . However, the $C^{13}(\alpha, n)O^{16}$ reaction is energetically possible for α -particles of energies >2.9 Mev, and has been observed. Thus, it is assumed that all of the neutrons reported by Roberts came from this reaction.

Roberts observed 0.11 neutron/ 10^6 5.3 Mev α -particles incident on thick carbon. In a UHTREX element, the principal α activity will be from the decay of U^{234} , which, with a half-life of 2.5×10^5 years, produces α -particles of about 4.8 Mev. The thick target yield of neutrons from carbon is not known as a function of α -particle energy. However, from measured yields versus energy for similar reactions (on B and F),²⁴ it is estimated that the thick target yield will be nearly a factor of two lower for 4.8 Mev α 's than for 5.3 Mev α 's. Since in a fuel element the uranium is dilute ($\rho_u \lesssim \rho_c/10$) and un-lumped (i.e., dimensions of uranium lumps $\ll$ range of α -particle), close to the thick target yield is obtained for each α -particle born in the fuel element. Thus, if the rate of α production is R/sec, about 0.06×10^{-6} R neutrons/sec are expected.

For one gram of U^{234} , the α decay rate is 2.25×10^8 sec⁻¹ so that the neutron source rate should be about 13.5 neutrons/sec/g of U^{234} . Since U^{234} is initially 1% of the uranium mass, 135 neutrons/sec/kg of uranium are expected.

These expectations have been confirmed by an experiment at this laboratory. With some actual fuel elements, 9.6 ± 0.7 neutrons/sec/kg of fuel element were observed. The quoted error is statistical only, and there is a comparable uncertainty due to spectral sensitivity of the detector system. Since the fuel elements contain about 76 g of uranium/kg of fuel element, it is seen that the above source rate corresponds to 126 ± 9 neutrons/sec/kg uranium, in good agreement with expectations.

If the reactor is to be critical with about 7 kg of uranium, the source from $U^{234}(\alpha, n)$ reactions is expected to be about 10^3 /sec. It can be seen that other source contributions are small for the initial loading. In particular, the spontaneous fission source is about 1% of the above, and the $U^{235}(\alpha, n)$ source is about 3% of the above.

The above figures correspond to clean uranium containing 1% U^{234} , such as will be used in the initial loading. In an equilibrium configuration, the situation will be somewhat different. First of all, if the uranium should tend to congregate into lumps, the source strength would decrease because of α -particle energy loss in the lumps. Other effects will make the source strength increase. For one thing, U^{234} has a thermal absorption cross section which is only about 15% as large as that of U^{235} , and thus the ratio U^{234}/U^{235} will steadily increase as U^{235} is burned up. If 50% of the uranium is burned per pass, it is expected that the equilibrium source from U^{234} will be roughly 30% larger than the initial source.

An additional source will appear as Pu^{239} and Pu^{240} isotopes are produced from neutron capture in U^{238} . At equilibrium, it is estimated that about 2.5% of the U^{238} will have been converted to Pu^{239} , assuming that the Pu remains in the fuel elements, and the atomic ratios $Pu^{239}/U^{234} \sim 0.07$, $Pu^{240}/U^{234} \sim 0.02$ are reasonable. Since α half-lives are in the ratios $\tau^{234}/\tau^{239} = 10$, $\tau^{234}/\tau^{240} = 38$, it is seen that these plutonium abundances would give an α decay rate about 150% as large as the U^{234} decay rate. Plutonium α 's are also of higher energy, so the Pu (α , n) rate would be about twice that due to U^{234} . A comparable contribution will arise from spontaneous fissions of Pu^{240} , for which the neutron production rate is about $10^3/\text{sec/g}$ of Pu^{240} .

Thus, if appreciable lumping of the uranium does not occur during burn-up, it is expected that the source strength will increase by about a factor of five during the transition from an initial clean loading to an equilibrium configuration. Some decrease in the (α , n) source strength would occur for uranium lump dimensions $\sim 10^{-4}$ cm, and the decrease would be serious (i.e., of the order of a factor of 10) for lump dimensions $\sim 10^{-3}$ cm. Such lumps may be unlikely, and discharged fuel elements could be examined to verify that they were not arising. Of course, as soon as equilibrium Pu^{240} has been formed, the spontaneous fission source will be present and unaffected by lumping.

2. Implications of the Source for Startup

The ordinary reactor kinetic equations will predict the average neutron population (in the normal mode) as a function of time during startup. However, if a reactor is brought to a supercritical state when only a weak source is present, then there is the possibility that a divergent neutron chain reaction will not start until some time appreciably after critical. Under these circumstances, the actual neutron population will, with some probability, depart appreciably from its average value. In particular, the actual population may be much less than the average population. This means that if the reactor startup is programmed on the basis of the average behavior, there is a chance that the neutron population will not be detected until the reactor has become embarrassingly supercritical. Evidently the instantaneous neutron population is not of great importance for these considerations. It can fluctuate quite erratically. However, it must be assured that the neutron population, when averaged over periods of time like seconds, will not depart dangerously from the corresponding average value. These considerations will now be investigated for the UHTREX source.

A theory has been developed for fluctuations during startup^{25, 26, 27}. In the theory, the prompt neutron lifetime is assumed to be zero (or at least very much less than the delayed neutron precursor lifetime), and only one group of delayed neutron precursors is allowed. Thus, the theory is not highly precise. However, it has given good qualitative agreement with Godiva fluctuations. Also, it will be shown that for UHTREX, the expected fluctuations are so small that the approximate theory should be good enough.

During UHTREX startup, the rate of rod removal is to be so low that the average neutron population, $\bar{n}(t)$, will have risen to detectable levels before the reactor period is shorter than about 20 sec. This means that the reactivity must not exceed delayed critical by more than about 30¢ before $\bar{n}(t)$ is detectable. The corresponding rod removal rate is $\sim 1\text{¢}/\text{sec}$.

Now, in the theory of fluctuations during startup, the fluctuations in neutron precursor population are considered rather than neutron population.

The reason is that the instantaneous neutron population will in general have a very erratic behavior and may fluctuate wildly in time. However, the precursor population and the time average of the neutron population (averaged over times comparable to the precursor lifetime) will be varying more slowly. The precursor population is thus both easier to compute and easier to relate to readings of detectors which have fairly long integrating times, of the order of seconds.

Precursor fluctuations are, in general, worse the higher the reactivity, other things being equal. Thus, to show that they are unimportant for UHTREX, consideration is given to fluctuations which would occur if all the multiplication took place 30¢ above delayed critical. For a step increase in reactivity from some low value to a high value, moments of the precursor population are readily computed.

Let $\bar{c}$ be the average precursor population (averaged over a large number of ensembles, or a large number of startups of the same system), $\overline{c^2}$ be the average of the square of the precursor population, β be the effective delayed neutron fraction, and k be the prompt reproduction number. Further, let S be the neutron source per precursor lifetime, $\bar{\nu}$ be the average number of neutrons per fission, and χ_2 be twice the average number of pairs of neutrons per fission. Then the theory shows that after a few e-folding times:

$$\left(\frac{c - \bar{c}}{\bar{c}} \right)^2 = \frac{\overline{c^2}}{\bar{c}^2} - 1 = \frac{\chi_2 (\beta k + |\beta k + k - 1|)}{2 \bar{\nu} S (1 - k)^2}$$

For U^{235} , $\chi_2 \cong 2 \bar{\nu}$ and $\beta = 0.007$. Then at 30¢ above delayed critical, $k = 0.9951$, so that

$$\frac{\overline{c^2}}{\bar{c}^2} - 1 = \frac{380}{S}$$

For this region of reactivity the effective delayed neutron lifetime is about 8 sec, so that $S \cong 8 \times 10^3$ and

$$\frac{\overline{c^2}}{\bar{c}^2} - 1 \cong 0.05$$

Thus for such a strong source, the fluctuations in precursor population are very small -- so small, indeed, that they need cause no concern. As mentioned above, the instantaneous neutron population will fluctuate much more, but the neutron population averaged over a few seconds will behave much as the precursor population.

It is concluded that for a source of 10^3 neutrons/sec there is no significant probability that the neutron population, averaged over a time of the order of a second, will depart materially from its mean value during startup. Indeed, the source could be nearly an order of magnitude smaller before the fluctuations would become serious.

APPENDIX D

A⁴¹ AND H³ PRODUCTION AND RELEASE

1. A⁴¹ within the Secondary Containment

The production rate of A⁴¹ is given by:

$$\frac{dN}{dt} = N_0 \sigma \phi A \ell$$

where

N = total number of A⁴¹ atoms

N₀ = number of atoms of A⁴⁰ per cm³ in the air = 3.02 x 10¹⁷ cm⁻³

σ = thermal neutron capture cross section = 0.53 b

φ = thermal neutron flux outside of the reactor vessel =
2 x 10¹⁰ cm⁻² sec⁻¹

A = area of the reactor vessel = 5 x 10⁵ cm²

ℓ = average neutron path length in the air = 2 x 10² cm.

Using these values for the variables, dN/dt = 3.2 x 10¹¹ sec⁻¹. The decay constant of A⁴¹ is λ = 1.05 x 10⁻⁴ sec⁻¹, so the source production rate is 9.1 x 10⁻⁴ C/sec.

For closed cycle operation of the ventilation system, the equilibrium source within the secondary containment will be

$$\frac{3.2 \times 10^{11}}{3.7 \times 10^{10}} = 8.65 \text{ C}$$

which gives a concentration in the total $4.2 \times 10^3 \text{ m}^3$ volume of $2.06 \times 10^{-3} \text{ C/m}^3$. To derive the equilibrium source dose rate, the superstructure is treated as a hemisphere of 10 m radius. The gamma flux at the center of a sphere is given by²⁸

$$\phi = \frac{BS_v}{\mu_s} (1 - e^{-\mu_s R_o})$$

which should be divided by two for a hemisphere. In the expression,

B = dose build-up factor $\cong 1$

S_v = source per unit volume

μ_s = macroscopic energy absorption cross section of air
for 1.3 mev γ -rays = $3 \times 10^{-5} \text{ cm}^{-1}$

R_o = radius of the sphere.

A^{41} emits a 1.3 mev γ -ray 99% of the time (none the other 1%), and the end point energy of the principal β -particle is 1.2 mev. Therefore, for calculating dose rate, the mean energy per disintegration is assumed to be 1.8 mev. The photon flux at the center of curvature of the hemisphere is $\phi = 3.8 \times 10^4 \text{ cm}^{-2} \text{ sec}^{-1}$, which gives a whole body dose rate of about 0.16 r/hr. Normally, the secondary containment will be flushed with outside air before personnel enter, but this dose rate would permit limited access even without flushing of the region.

2. A⁴¹ in Stack Effluent

The stack for the facility is 100 ft high. In the following, no credit is taken for increase in effective stack height due to effluent temperature or velocity nor is any credit taken for the irregularities in the surrounding terrain whose effects on air flow are described in Section 2.4.1.

Sutton's atmospheric diffusion equations as discussed in AECU-3066²⁰ and WASH-740²⁹ were used to estimate the dilution of the stack effluents and the dimensions of the plume. The dose rates were then computed using the methods outlined by Rockwell for a cylindrical source. Assuming typical inversion conditions for the Los Alamos area, continuous release of the A⁴¹ at its production rate would produce a maximum ground concentration at a distance of about 1 km, and the dose rate at that point would be about 10^{-4} r/hr. Consideration of the frequency and duration of such inversions (none has been recorded as lasting through a 24-hr period) and of the variability of wind direction over any extended operating time would reduce the average dose rate at any one point by at least an order of magnitude. The closed cycle operation of the secondary containment ventilation system, coupled with the short half-life of this isotope, would reduce the total activity released by an order of magnitude for venting following a 26-hr operating period compared to continuous release. This reduction factor increases linearly with operating time. The combination of these factors would reduce the average integrated dose from A⁴¹ to persons in unrestricted areas to no more than a few mr per year, and even this figure assumes unreasonably frequent shutdown and venting operations.

3. Tritium Production and Release

Tritium will be produced in the primary coolant by the $\text{He}^3 (n, p) \text{H}^3$ reaction. The fractional concentration of He^3 in natural He is about 1.3×10^{-6} . The (n, p) cross section of He^3 is about 2.1×10^3 b at the operating temperature of 0.14 ev. There are about 7×10^2 g He in and near the core in an average neutron flux of less than $3 \times 10^{13} \text{ cm}^{-2} \text{ sec}^{-1}$. Therefore, the tritium production

rate will be no greater than:

$$\begin{aligned}\frac{dN}{dt} &= 1.3 \times 10^{-6} \times \frac{7 \times 10^2}{4} \times 6 \times 10^{23} \times 2.1 \times 10^{-21} \times 3 \times 10^{13} \\ &= 7.4 \times 10^{12} \text{ sec}^{-1}\end{aligned}$$

which amounts to 3.5×10^{-7} C/sec.

Normally this isotope will be removed in the gas purification system by oxidation in the copper oxide bed and absorption in the molecular sieve beds. Periodically the molecular sieve beds will be regenerated and the tritium vented to the stack. Averaged over the production time, the release rate can be no greater than the production rate of 4×10^{-7} C/sec. Assuming a typical inversion for this area, the maximum ground concentration, averaged over the production time, would be about 3.5×10^{-11} C/m³, which is far below the concentration of 2×10^{-7} C/m³ permitted in uncontrolled areas.³⁰

APPENDIX E

CALCULATION OF RADIATION HAZARDS FROM MAXIMUM CREDIBLE ACCIDENT

1. Inhalation Doses Due to Activity Release from Secondary Containment

The following assumptions are made:

1. The release results from a rupture in the primary coolant loop.
2. Reactor has operated for one year at 3 Mw.
3. Activity release from primary loop to secondary containment consists of:
 - 100% of inert gases
 - 10% of halogens
 - 1% of remaining fission products.
4. Activity leakage from secondary containment is at a rate of 1% per day.
5. The leak constitutes a continuous, ground level, point source.
6. The accident occurs during a typical inversion; wind velocity is 5 mph in direction of nearest uncontrolled land. These conditions persist for 24 hrs.

Dose points considered are at 1000 ft and one mile, downwind. The 1000-ft dose is calculated over a 2-hr period, and the one-mile dose is integrated over a 24-hr period.

Dilution factors at the dose points are obtained from the relationship

$$K = \frac{2 Q_0}{\pi C^2 \bar{u} x^{2-n}},$$

a special case of Sutton's equation²⁰ (continuous point source, ground level) for concentrations of airborne contaminants,

where

K = The dilution factor, $\frac{C/m^3}{\text{leak source } C/\text{sec}}$

Q_0 = unit leak rate, 1 C/sec

x = downwind distance, m

$\bar{u}$ = wind velocity, m/sec

$C^2 = 0.01$ } Dimensionless meteorological parameters, values
 $n = 0.35$ } appropriate to a typical inversion.

The variation of K with downwind distance is plotted in Fig. E. 1.

1.1 Dose to Bone, F_b

Data for bone and thyroid dosage calculations are taken from Burnett.²² Neglecting source decay, bone dose is obtained from

$$F_b = 10^{-3} P E' K R T \sum_i (S_i E'_i D_i)$$

where

F_b = bone dose in rem from activity inhaled during T hrs

P = power level, 3×10^3 kw

$E' =$ fractional leak rate out of secondary containment, $1.16 \times 10^{-7} \text{ sec}^{-1}$

K = dilution factor

R = breathing rate, assumed to be $1 \text{ m}^3/\text{hr}$

S_i = source inventory, $\mu \text{ C/kw}$, for the i th bone-seeking isotope

E'_i = the fractional release to the secondary containment

D_i = the dose factor, $\text{mrem}/(\mu \text{ C of inhaled activity})$

Table E. 1. 1 lists the contributions to the summation in the equation.

Fig. E. 1 Source concentrations in curies per cubic meter for a release of one curie per second. Continuous release at ground level is assumed under a typical inversion with a wind velocity of 5 mph.

TABLE E. 1. 1
ISOTOPES CONTRIBUTING TO BONE DOSE

Isotope	S	E'	D	SE'D
Sr ⁸⁹	4.34 x 10 ⁷	10 ⁻²	3.12 x 10 ²	1.35 x 10 ⁸
Sr ⁹⁰ -Y ⁹⁰	1.45 x 10 ⁶	10 ⁻²	3.48 x 10 ⁴	5.05 x 10 ⁸
Sr ⁹¹ -Y ^{91m}	5.36 x 10 ⁷	10 ⁻²	3.3	0.02 x 10 ⁸
Y ⁹¹	5.32 x 10 ⁷	10 ⁻²	2.17 x 10 ²	1.15 x 10 ⁸
Zr ⁹⁵	5.73 x 10 ⁷	10 ⁻²	2.45 x 10 ¹	0.14 x 10 ⁸
Ba ¹⁴⁰ -La ¹⁴⁰	5.72 x 10 ⁷	10 ⁻²	1.09 x 10 ²	0.62 x 10 ⁸
Ce ¹⁴¹	5.45 x 10 ⁷	10 ⁻²	3.04 x 10 ¹	0.17 x 10 ⁸
Pr ¹⁴³	5.63 x 10 ⁷	10 ⁻²	1.14 x 10 ¹	0.06 x 10 ⁸
Ce ¹⁴⁴ -Pr ¹⁴⁴	3.47 x 10 ⁷	10 ⁻²	1.16 x 10 ³	<u>4.02 x 10⁸</u>
$\Sigma (SE'D) =$				1.22 x 10 ⁹

For 1000-ft downwind, first 2 hrs after accident:

$$F_b = 10^{-3} \times 1.22 \times 10^9 \times 3 \times 10^3 \times 1.16 \times 10^{-7} \times 1.9 \times 10^{-3} \times 2$$

$$= 1.6 \text{ rem}$$

For one mile and 24 hrs:

$$F_b = 1.6 \times \frac{1.3 \times 10^{-4}}{1.9 \times 10^{-3}} \times 12$$

$$= 1.3 \text{ rem}$$

1.2 Dose to Thyroid, F_t

The thyroid dose is calculated in the same manner as that for bone dose with the contributions to the summation as listed in Table E.1.2.

TABLE E.1.2

ISOTOPES CONTRIBUTING TO THYROID DOSE

Isotope	S	E'	D	SE'D
I ¹³¹	2.63×10^7	10^{-1}	9.63×10^2	2.53×10^9
I ¹³²	4.0×10^7	10^{-1}	3.66×10^1	0.15×10^9
I ¹³³	5.9×10^7	10^{-1}	2.27×10^2	1.34×10^9
I ¹³⁴	6.9×10^7	10^{-1}	1.77×10^1	0.12×10^9
I ¹³⁵	5.4×10^7	10^{-1}	6.3×10^1	0.34×10^9
				$\Sigma(SE'D) = 4.48 \times 10^9$

At 1000 ft, first 2 hrs after accident:

$$F_t = 10^{-3} \times 4.48 \times 10^9 \times 3 \times 10^3 \times 1.16 \times 10^{-7} \times 1.9 \times 10^{-3} \times 2$$

$$= 5.92 \text{ rem}$$

at one mile, first 24 hrs:

$$F_t = 5.92 \times \frac{1.3 \times 10^{-4}}{1.9 \times 10^{-3}} \times 12 = 4.86 \text{ rem.}$$

2. Whole Body Gamma Dose Due to Immersion in Cloud

From WASH-740,²⁹ the dose rate in a semi-infinite cloud, corrected for air density at a 7000-ft elevation, is

$$d = 0.296 \text{ CE}\gamma \text{ rad/sec}$$

where

$$\text{CE}\gamma = \text{AE}''\text{K}$$

and

C = concentration of activity, C/m^3

E_γ = effective gamma energy

A = a time dependent value of gamma-ray activity (Fig. 10.2.3.1)
inside the secondary containment shell

E'' = the leak rate, $1.16 \times 10^{-7} \text{ sec}^{-1}$

K = the dilution factor (Fig. E.1)

Then

$$d = 3.43 \times 10^{-8} AK$$

For 0-1 hr:

$$\bar{A}_{0-1} \sim \frac{A_0 + A_1}{2} = \frac{(1.99 + 0.56)}{2} \times 10^6 = 1.28 \times 10^6 \text{ C-Mev}$$

For 1-2 hr:

$$\bar{A}_{1-2} \sim \frac{A_1 + A_2}{2} = \frac{(5.56 + 4.11)}{2} \times 10^5 = 4.84 \times 10^5 \text{ C-Mev}$$

For 2-10 hr:

$$\bar{A}_{2-10} \sim \frac{A_2 + A_{10}}{2} = \frac{(4.11 + 1.09)}{2} \times 10^5 = 2.6 \times 10^5 \text{ C-Mev}$$

For 10-20 hr:

$$\bar{A}_{10-20} \sim \frac{A_{10} + A_{20}}{2} = \frac{(1.09 + 0.57)}{2} \times 10^5 = 0.83 \times 10^5 \text{ C-Mev.}$$

At 1000 ft,

$$K = 1.9 \times 10^{-3}$$

and the 2-hr dose

$$\begin{aligned} F_2 &= 3.43 \times 10^{-8} \times 1.9 \times 10^{-3} \times 3.6 \times 10^3 (1.28 + 0.484) \times 10^6 \\ &= 0.41 \text{ rad.} \end{aligned}$$

At one mile,

$$K = 1.3 \times 10^{-4}$$

and the 24-hr dose

$$\begin{aligned} F_{24} &= 3.43 \times 10^{-8} \times 1.3 \times 10^{-4} \times 3.6 \times 10^3 (1.28 + 0.484 + 8 \times 0.26 + \\ &\quad 14 \times 0.083) \times 10^6 \\ &= 0.080 \text{ rad.} \end{aligned}$$

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