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~~Health and Safety~~

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UCRL-51423

**SAFETY ANALYSIS REPORT
FOR
LIVERMORE POOL TYPE REACTOR**

MS date: September 19, 1974

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PREFACE

This Safety Analysis Report has been prepared to provide a comprehensive review of those considerations involving the Livermore Pool Type Reactor that could bear on the safety of the general public, personnel working at the Lawrence Livermore site, and persons working within this facility.

The report describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structure, systems and components, the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment systems, other engineered safety features, auxiliary and emergency systems, and of the system as a whole.

Specialists representing a number of different disciplines from LLL were consulted and have contributed to the preparation of this SAR. Special studies, analyses, and operational procedures that bear on this report and were prepared as separate documents are included as appendices. This analysis shows that the design and operation of the facility presents no undue hazard to the general public or employees and will have no adverse effect on the general environment.

FOREWORD

The International System of Units (SI) is used throughout this report in place of the mile, Btu/hr, psi, etc., customarily used in engineering. This has been done in support of the AEC decision to convert to SI units and in anticipation of the future general adoption of this system. The reader may find some units, such as the pascal, unfamiliar. He will also come across units that may be hard to visualize physically, such as those for pump flow rates: e.g. 250 gpm converts into the SI value of $160 \text{ dm}^3/\text{s}$. With continued exposure to these units, however, the reader will find them becoming more meaningful and easier to comprehend. The list below, adapted from the ASTM Metric Practice Guide, E 380-72, is presented as an aid in expediting that process.

<u>To convert from</u>	<u>To</u>	<u>Divide by</u>
<u>Acceleration</u>		
metre/second ² (m/s^2)	foot/second ²	3.048 000 E - 01
metre/second ² (m/s^2)	free fall, standard	9.806 650 E + 00
<u>Area</u>		
metre ² (m^2)	foot ²	9.290 304 E - 02
metre ² (m^2)	inch ²	6.451 600 E - 04
metre ² (m^2)	mile ² (U.S. Statute)	2.589 988 E + 06
metre ² (m^2)	yard ²	8.361 274 E - 01
<u>Energy (Includes Work)</u>		
joule (J)	British thermal unit (thermochemical)	1.054 350 E + 03
joule (J)	calorie (thermochemical)	4.184 000 E + 00
joule (J)	calorie (kg, thermochemical)	4.184 000 E + 03
joule (J)	foot-pound-force	1.355 818 E + 00
joule (J)	kilocalorie (thermochemical)	4.184 000 E + 03
joule (J)	kilowatt-hour	3.600 000 E + 06
joule (J)	ton (nuclear equivalent of TNT)	4.20 E + 09
joule (J)	watt-hour	3.600 000 E + 03
joule (J)	watt-second	1.000 000 E + 00

<u>To convert from</u>	<u>To</u>	<u>Divide by</u>
<u>Energy/Area Time</u>		
watt/metre ² (W/m ²)	Btu (thermochemical)/foot ² -second	1.134 893 E + 04
watt/metre ² (W/m ²)	Btu (thermochemical)/foot ² -minute	1.891 489 E + 02
watt/metre ² (W/m ²)	Btu (thermochemical)/foot ² -hour	3.152 481 E + 00
watt/metre ² (W/m ²)	calorie (thermochemical)/centimeter ² -minute	6.973 333 E + 02
watt/metre ² (W/m ²)	watt/centimetre ²	1.000 000 E + 04
<u>Force</u>		
newton (N)	kilogram-force	9.806 650 E + 00
newton (N)	pound-force (lbf avoirdupois)	4.448 222 E + 00
<u>Heat</u>		
watt/metre-kelvin (W/m · K)	Btu (thermochemical) · in./h · ft ² · deg F (k, thermal conductivity)	1.441 314 E - 01
watt/metre ² -kelvin (W/m ² · K)	Btu/h · ft ² · deg F (C, thermal conductance)	5.678 263 E + 00
joule/kilogram (J/kg)	Btu/pound-mass	2.326 000 E + 03
joule/kilogram-kelvin (J/kg · K)	Btu/lbm · deg F (c, heat capacity)	4.186 800 E + 03
joule/metre ² (J/m ²)	cal (thermochemical)/cm ²	4.184 000 E + 04
watt/metre ² (W/m ²)	cal (thermochemical)/cm ² · s	4.184 000 E + 04
joule/kilogram (J/kg)	cal (thermochemical)/g	4.184 000 E + 03
joule/kilogram-kelvin (J/kg · K)	cal (thermochemical)/g · deg C	4.184 000 E + 03
kelvin-metre ² /watt (K · m ² /W)	deg F · h · ft ² /Btu (thermochemical)(R, thermal resistance)	1.762 280 E - 01
metre ² /second (m ² /s)	ft ² /h (thermal diffusivity)	2.560 640 E - 05
<u>Length</u>		
metre (m)	foot	3.048 000 E - 01
metre (m)	inch	2.540 000 E - 02
metre (m)	micron	1.000 000 E - 06
metre (m)	mil	2.540 000 E - 05
metre (m)	mile (U.S. statute)	1.609 344 E + 03
metre (m)	yard	9.144 000 E - 01

<u>To convert from</u>	<u>To</u>	<u>Divide by</u>
<u>Mass</u>		
kilogram (kg)	gram	1.000 000 E - 03
kilogram (kg)	kilogram-force-second ² /metre (mass)	9.806 650 E + 00
kilogram (kg)	kilogram-mass	1.000 000 E + 00
kilogram (kg)	ounce-mass (avoirdupois)	2.834 952 E - 02
kilogram (kg)	pound-mass (lbm avoirdupois)	4.535 924 E - 01
kilogram (kg)	ton (short, 2000 lbm)	9.071 847 E + 02
<u>Mass/Area</u>		
kilogram/metre ² (kg/m ²)	ounce-mass/yard ²	3.390 575 E - 02
kilogram/metre ² (kg/m ²)	pound-mass/foot ²	4.882 428 E + 00
<u>Mass/Time (Includes Flow)</u>		
kilogram/second (kg/s)	pound-mass/second	4.535 924 E - 01
kilogram/second (kg/s)	pound-mass/minute	7.559 873 E - 03
kilogram/second (kg/s)	ton (short, mass)/hour	2.519 958 E - 01
<u>Mass/Volume (Includes Density and Mass Capacity)</u>		
kilogram/metre ³ (kg/m ³)	gram/centimetre ³	1.000 000 E + 03
kilogram/metre ³ (kg/m ³)	pound-mass/foot ³	1.601 846 E + 01
kilogram/metre ³ (kg/m ³)	pound-mass/inch ³	2.767 990 E + 04
kilogram/metre ³ (kg/m ³)	pound-mass/gallon (U.S. liquid)	1.198 264 E + 02
kilogram/metre ³ (kg/m ³)	ton (long, mass)/yard ³	1.328 939 E + 03
<u>Power</u>		
watt (W)	Btu (thermochemical)/second	1.054 350 E + 03
watt (W)	Btu (thermochemical)/minute	1.757 250 E + 01
watt (W)	Btu (thermochemical)/hour	2.928 751 E - 01
watt (W)	calorie (thermochemical)/second	4.184 000 E + 00
watt (W)	calorie (thermochemical)/minute	6.973 333 E - 02
watt (W)	foot-pound-force/second	1.355 818 E + 00
watt (W)	horsepower (electric)	7.460 000 E + 02
watt (W)	kilocalorie (thermochemical)/second	4.184 000 E + 03
watt (W)	watt, international U.S. (W ^{INT-US})	1.000 182 E + 00
watt (W)	watt, U.S. legal 1948 (W ^{US-48})	1.000 017 E + 00

<u>To convert from</u>	<u>To</u>	<u>Divide by</u>
<u>Pressure or Stress (Force/Area)</u>		
pascal (Pa)	atmosphere (normal = 760 torr)	1.013 25 E + 05
pascal (Pa)	bar	1.000 000 E + 05
pascal (Pa)	centimetre of mercury (0 C)	1.333 22 E + 03
pascal (Pa)	centimetre of water (4 C)	9.806 38 E + 01
pascal (Pa)	dyne/centimetre ²	1.000 000 E - 01
pascal (Pa)	foot of water (39.2 F)	2.988 98 E + 03
pascal (Pa)	gram-force/centimetre ²	9.806 650 E + 01
pascal (Pa)	inch of mercury (60 F)	3.376 85 E + 03
pascal (Pa)	inch of water (60 F)	2.488 4 E + 02
pascal (Pa)	kip/inch ² (ksi)	6.894 757 E + 06
pascal (Pa)	millibar	1.000 000 E + 02
pascal (Pa)	millimetre of mercury (0 C)	1.333 224 E + 02
pascal (Pa)	poundal/foot ²	1.488 164 E + 00
pascal (Pa)	pound-force/foot ²	4.788 026 E + 01
pascal (Pa)	pound-force/inch ² (psi)	6.894 757 E + 03
pascal (Pa)	psi	6.894 757 E + 03
pascal (Pa)	torr (mm Hg, 0 C)	1.333 22 E + 02

Speed (See Velocity)

Stress (See Pressure)

Temperature

kelvin (K)	degree Celsius	$t_K = t_C + 273.15$
kelvin (K)	degree Fahrenheit	$t_K = (t_F + 459.67)/1.8$
kelvin (K)	degree Rankine	$t_K = t_R/1.8$
degree Celsius (°C)	degree Fahrenheit	$t_C = (t_F - 32)/1.8$
degree Celsius (°C)	kelvin	$t_C = t_K - 273.15$

Time

second (s)	day (mean solar)	8.640 000 E + 04
second (s)	hour (mean solar)	3.600 000 E + 03
second (s)	minute (mean solar)	6.000 000 E + 01
second (s)	month (mean calendar)	2.628 000 E + 06
second (s)	second (sidereal)	9.972 696 E - 01
second (s)	year (calendar)	3.153 600 E + 07

<u>To convert from</u>	<u>To</u>	<u>Divide by</u>
<u>Velocity (Includes Speed)</u>		
metre/second (m/s)	foot/second	3.048 000 E - 01
metre/second (m/s)	mile/hour (U.S. statute)	4.470 400 E - 01
radian/second (rad/s)	revolution/minute (rpm)	1.04 E - 01
<u>Volume (Includes Capacity)</u>		
metre ³ (m ³)	acre-foot	1.233 482 E + 03
metre ³ (m ³)	foot ³	2.831 685 E - 02
metre ³ (m ³)	gallon (U.S. liquid)	3.785 412 E - 03
metre ³ (m ³)	litre	1.000 000 E - 03
metre ³ (m ³)	pint (U.S. liquid)	4.731 765 E - 04
metre ³ (m ³)	quart (U.S. liquid)	9.463 529 E - 04
metre ³ (m ³)	yard ³	7.645 549 E - 01
<u>Volume/Time (Includes Flow)</u>		
metre ³ /second (m ³ /s)	foot ³ /minute	4.719 474 E - 04
metre ³ /second (m ³ /s)	gallon (U.S. liquid)/day	4.381 264 E - 08
metre ³ /second (m ³ /s)	gallon (U.S. liquid)/minute	6.309 020 E - 05
<u>Work (See Energy)</u>		

SI PREFIXES:

<u>Multiplication Factors</u>	<u>Prefix</u>	<u>SI Symbol</u>
1 000 000 000 000 = 10 ¹²	tera	T
1 000 000 000 = 10 ⁹	giga	G
1 000 000 = 10 ⁶	mega	M
1 000 = 10 ³	kilo	k
100 = 10 ²	hecto*	h
10 = 10 ¹	deka*	da
0.1 = 10 ⁻¹	deci*	d
0.01 = 10 ⁻²	centi*	c
0.001 = 10 ⁻³	milli	m
0.000 001 = 10 ⁻⁶	micro	μ
0.000 000 001 = 10 ⁻⁹	nano	n
0.000 000 000 001 = 10 ⁻¹²	pico	p
0.000 000 000 000 001 = 10 ⁻¹⁵	femto	f
0.000 000 000 000 000 001 = 10 ⁻¹⁸	atto	a

*To be avoided except on units of higher order, e.g., dm³, cm³.

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SAFETY ANALYSIS REPORT FOR THE LIVERMORE POOL TYPE REACTOR

1.0 INTRODUCTION AND GENERAL PLANT DESCRIPTION

1.1 Introduction

Shortly after the formation of the Livermore branch of Lawrence Livermore Laboratory (LLL), the need for a high-flux research reactor facility became apparent. It was decided that a versatile facility should be acquired, one that would be suitable for general research by all departments of the laboratory. After much consideration, a tank-type pool reactor with a 1-MW power level and a peak flux of greater than 10^{13} n/cm²/sec was selected because of safety, economic feasibility, and flexibility of experimental facilities.

The general specifications were incorporated in a preliminary proposal formulated in late 1954. Preliminary planning continued for several months under the supervision of the San Francisco Operations Office of the AEC, culminating in the preparation of a final bidding document which was forwarded to 34 prospective bidders in March 1955. A contract to design, construct, and test the Livermore Pool-Type Reactor (LPTR) was signed by the Foster Wheeler Corporation in July 1955. The LPTR contract specified a reactor designed for 1-MW operation, with the shielding and all associated equipment being flexible enough to permit conversion to 5-MW operation without major structural modifications.

Construction of the LPTR began in October 1956, and was complete in April 1958. The reactor was first taken critical on December 13, 1958. For about four months thereafter, the reactor facility was thoroughly tested and the reactor operators were trained. The reactor was operated at 1 MW from May 1959 until June 1960, when a construction project was started to modify some of the confinement structure equipment. In October 1960, the LPTR was granted authority to operate at 2 MW and operated at that power level until February 1967. In October 1966, authority was granted for 3-MW operations.

The reactor operated at 3 MW from February 1967 until it was shut down for repairs in November 1970. Several primary coolant leaks were located within the tank boundary and concrete shield, and subsequently sealed. Water-damaged graphite in the east thermal column was cleaned and replaced. The west thermal-column nose piece, which suffered gamma overheating, was replaced with one of similar dimensions but of improved design. The LPTR was given authority to resume 3-MW operations in June 1971.

This report presents systems descriptions, systems data, operational information and analyzes all hazards resulting from the operation of the LPTR at 3 MW. All changes in the facility which have occurred since the original construction are included. It is intended that this report supersede the LPTR Safeguard Report and Addenda of November 1966.

1.2 General Description

The LPTR is a light-water-moderated-and-cooled tank-type pool research reactor located geometrically in the center of a cylindrical steel shell-type containment structure with controlled leakage.

The reactor is fueled with fully enriched ^{235}U fuel elements of the MTR type. Each control rod consists of boron carbide clad in a stainless steel jacket. Four such rods make up the shim safety rods, while one water-filled stainless steel rod serves as a regulating rod.

The LPTR typically carries approximately three and a half kilograms of fuel and is refueled about every thirty megawattdays (MWd) in order to preserve a reactivity margin large enough to allow routine scrams from power for short periods without being poisoned out. The reactor can be brought easily to power and scrammed instantly as desired. The reactor system contains no high-pressure or high-temperature stored energy. Several beam ports, pneumatic target handling systems and reactor core positions are available for target and experiment irradiations.

The core and structural components generate heat by fission and gamma absorption respectively. This heat is removed from the pool by a constant-flow primary coolant system. A variable-flow secondary coolant system carries the heat to a forced draft cooling tower and hence to the atmosphere.

An equipment pit below floor level contains auxiliary piping systems, heat exchanger, and primary pump. A floor storage area consisting of lined holes in the floor allows storage of hot fuel elements or radioactive items. Specific systems description will be found in detail in subsequent subsections.

1.2.1 Reactor Building

The reactor is housed in a controlled-leakage steel building designed to assure effective containment of radioactive effluents under all conditions. The building is a cylindrical shell of welded steel plates 6.3 mm thick (reinforced with angle rings) with a domed, self-supporting roof of 7.6-mm steel plates. The base of this structure is welded to large anchor bars which are buried in a massive reinforced-concrete ring wall. The

Fig. 1.2-1. Reactor building, East-West section view facing North.

ring wall is deep enough to prevent seasonal ground swelling. The building is 24 m in diam, 16 m high, with a working floor area of nearly 420 m² (volume of over 6000 m³). A section view of the building appears in Fig. 1.2-1.

A number of precautions have been taken to ensure a controlled-leakage structure when the reactor is operating. Normal personnel access is restricted to one airlock passage with the sealable doors interlocked so that only one can open at a time. The truck door and two emergency escape doors open inward, and are leak-resistant with gaskets inflated during reactor operation. Fail-closed butterfly valves in the air-conditioning system and in the cylindrical stack at the top of the steel dome (through which circulating air and waste gases are exhausted) are the only normal openings to the environment. The containment philosophy is based on sealing the building only when necessary. Because the building is not designed to withstand large external overpressures, two vacuum relief valves admit outside air when external pressure exceeds internal pressure by 370 Pa. To prevent internal overpressure, the building may be vented through a filtering system.

The building floor is a 150-mm-thick level concrete slab, with a 100-mm curb around the entire circumference. An antileak ring is welded to the steel ring wall below the concrete surface, and copper waterstop is used wherever there is a break in the concrete construction. To facilitate decontamination, the floor is surface hardened and covered with five coats of a strippable plastic paint. The floor has several drains and a number of flush, covered pits, including one large equipment pit. Other than this, the floor is free of breaks or obstructions.

In order to minimize floor clutter, fourteen 4-in. conduits were buried in the floor, running radially from the reactor faces to the building wall and capped at floor level for use with experimental equipment or for utilities extensions. A 0.3-m-diam conduit goes between the reactor face and the equipment pit, then from the equipment pit to the building wall.

1.2.2 Crane and Special Equipment

A 14-Mg traveling crane services the main area of the building, the reactor tank, and experimental facilities. The crane has an auxiliary 1-Mg hoist with the same floor coverage (15.5×10 m). Both hoists have a maximum hook height of 11 m, clearing the highest pool-top structure by about 1.2 m. A 2-Mg monorail hoist mounted above the main crane, with a 9.8-m run perpendicular to the main crane rail, has a hook height of 14 m. The crane trolley and all hoists have variable-speed motors, with a low-speed range and magnetic brakes. As an added precaution, special provisions prevent the long handling tools used in the reactor tank from touching the crane's electrical conductors.

An electric fork lift is used to move loads beyond the crane coverage (about 50% of the building area), for towing coffins, and positioning shielding material and experimental equipment. The laboratory keeps many cranes and forklifts of much higher capacity available for use as needed.

1.2.3 Storage Facilities

1.2.3.1 Beam Plug Handling and Storage - The beam plugs originally provided with the reactor can be removed using a beam plug storage coffin which is intended to shield the moderately activated nose sections of the plugs, and to provide an alignment platform for plug insertion and removal. The coffin contains up to 150 mm of lead shielding and weighs about 5 Mg. It can be used for temporary plug storage, but generally the plugs are towed to the external beam plug storage facility (Fig. 1.2-2) where plugs or activated experimental equipment can be kept safely for long periods.

Fig. 1.2-2. Plan view of reactor building.

1.2.3.2 Floor Storage Pit – A floor storage pit which will accommodate many fuel elements and a variety of other active items is situated close to the reactor in the area serviced by the crane (Fig. 1.2-2). Thirty 5-in.-diam aluminum pipes are sunk to a depth of 3.4 m. These are capped on the bottom and can be filled with water to accommodate as many as two fuel elements each. No special cooling system is provided. Similarly, half a dozen larger-diameter carbon-steel pipes provide storage space for other active items which require temporary or extended storage. Primarily designed for spent-fuel-element storage, this pit also provides a place to store the core whenever in-tank maintenance or modification requires its removal from the tank.

1.2.3.3 Outdoor Storage – Butler buildings are provided outside the reactor building for storing experimental and maintenance gear.

1.2.4 Equipment Pit

A mechanical equipment pit is located in the reactor building floor, just west of the reactor shield (Fig. 1.2-2). With dimensions of $4 \times 10 \times 4.9$ m deep, it houses most of

the mechanical support systems. The pit is covered with removable concrete slab sections 0.3 m thick, with personnel access provided by a covered manhole. If a rupture caused the pool to drain, the 45 m³ in the system would flow into the bottom of the equipment pit and fill it to a depth of about 1.2 m. Consequently, all equipment except the sump pump is mounted above this level. If water did collect in the pit, it could be pumped into the waste tanks, the storage tanks, or to external tanks (see Fig. 1.2-3). The equipment pit is routinely accessible whether the reactor is operating or scrammed. Radiation levels are only moderately above background.

1.2.5 Air Cleaning System (Filter-Scrubber System)

Twin filter systems ensure that controlled leakage emits only an absolute minimum of radioactive material during an accident situation. These systems are located in a Butler building outside the containment building (see Fig. 1.2-4). All portions of each system can be operated manually.

In case of an accident an air-lock door gasket inflates to seal the building (the emergency escape and truck door gaskets are normally inflated at all times). If diurnal pressure temperature effects raise the building pressure above ambient pressure, the

Fig. 1.2-3. Waste water system.

Fig. 1.2-4. Air cleaning (filter-scrubber) system, schematic.

filter-scrubber system starts to exhaust air from the building. This air, after passing through Cambridge absolute filters, activated charcoal filters, and scrubber exhausts through 12-m stacks into the atmosphere essentially free of particulate fission products and the radioiodines (see Fig. 1.2-5). The charcoal filters remove any methyl iodide formed with a measured efficiency of 99.9% (dry weather).

The use of duplicate air cleaning systems, each with its own filters, assures continuous availability. Each unit alone is capable of holding the building pressure down during the most severe diurnal temperature changes. The intake pipe starts high in the reactor dome to preclude taking moisture or foam into the system.

1.2.6 Air Decontamination System

A system of ducts and dampers permits air to be recirculated within the sealed building (see Fig. 1.2-5). In the event of a building seal, the three butterfly valves close. This diverts the flow out through the air-conditioning duct on the north wall of the building, through a bank of four Cambridge absolute filters, and in through the duct on the south

Fig. 1.2-5. Reactor plan view showing air flow and ducting for air conditioning and air cleaning systems.

wall. Thus a sweep through the building from south to north accompanies air cleanup. This system will filter $2.6 \text{ m}^3/\text{s}$, or the equivalent of the entire building volume in 2700 s. The filters are almost 100% efficient for particles larger than 0.3 microns (μm). The equipment for this recirculation system is located just outside the air lock (see left side, Fig. 1.2-1 and Fig. 1.2-5) where the filter is housed in a unit specially designed to facilitate replacement of contaminated filters. The decontamination system can operate continuously until the activity reaches a minimum level.

1.2.7 Containment Air Conditioning System

The air-conditioning unit is capable of maintaining the summer temperatures 11°C below ambient, with a maximum of 40% relative humidity at 40°C , and supplies sufficient fresh air to completely change the air at breathing level (below 4.5 m) twice every hour. This equipment is located in the east mechanical equipment room of the adjacent lab building with ducts penetrating the reactor building near the air lock (see Fig. 1.2-1 and 1.2-4). Cooled and heated air are delivered by two fans ($3.8 \text{ m}^3/\text{s}$) through equally-spaced registers in a duct running the circumference of the wall. Heated air is supplied by hot water coil sections installed in the ducts within the mechanical equipment room. Recirculating air leaves the building through a duct leading to the inlet of the air-conditioning system where it is replenished with about 30% fresh air. In addition, building air is exhausted through the stack on top of the reactor building by means of a $1 \text{ m}^3/\text{s}$ blower. A separate air duct with heating and cooling coils furnishes conditioned air to the reactor console and control room. The air conditioning compressors are cooled with the site's low-conductivity-water system.

Insulation is provided by a 9-mm layer of cork and mastic sprayed on the outside surfaces of the building and covered with a 25-mm layer of insulation foam coated with protective paint.

1.2.8 Reactor Building Communications

This facility uses many communication systems in addition to standard telephone and fire alarm systems.

- A public address system has a speaker in each room of the laboratory building (and hallways), and three speakers in the reactor building. Announcements originate from either the main office or the control room.
- An intercom system provides for communication between the control room, the reactor building, and the laboratory building. Sixty-channel master stations are located at the console and in the main office, with 12 intercom stations scattered strategically throughout the reactor building, including stations near each experimental facility, at the reactor top and balcony, and in the equipment pit. A speaker is also located in each room in the lab building.

- A public announcement system from the Security and Fire departments has speakers located along the lab hallways and in the reactor building.
- The telephone page system is interconnected with the Radiochemistry Department, Building 151.
- A closed-circuit TV system provides visual contact between the operations console and reactor. Two portable cameras with zoom lenses watch from opposite sides of the building at a height of about 9 m. The system incorporates remote scanning and focus control with a two-camera switch feeding a 17-in. monitor at the console. In addition, a second TV channel observes the core of the reactor with a camera mounted under the control structure at the top of the tank.

1.2.9 Fire Protection Systems

1.2.9.1 Alarms – Heat sensing detectors, located in the reactor building, control room, and laboratory building, send alarms to the Fire Department in case of fire.

1.2.9.2 Fire Extinguishing System – Various types of hand-held fire extinguishers are located along the hallways and in the reactor building. A manually triggered high-expansion foam system is capable of filling the reactor building with foam in 540 s. This system uses a detergent-water mixture which is sprayed onto a screen and formed into foam by a $12 \text{ m}^3/\text{s}$ fan. Tests show that such systems smother a fire quickly without injury to personnel or equipment.

1.2.10 Laboratory Building

The LPTR laboratory, a 1200 m^2 , one-story, concrete-block building, is adjacent to the reactor building (see Fig. 1.2-6). It is intimately connected to the reactor via the airlock, conduits, and pipes. It houses the control console and many of the mechanical utilities for the reactor. In addition it contains a chemistry, physics and biomedical lab, electronic shop, machine shop, and office space for reactor personnel.

Although air conditioning is provided, greatest emphasis is placed on providing close temperature and humidity control for the control room. The $8.2 \times 8.5\text{-m}$ control room contains a Leeds and Northrup console housing all of the reactor controls, with additional racks provided at the side of the console for experimental apparatus and controls for auxiliary equipment. Air conditioning for the reactor room and control room is centralized in the mechanical equipment room nearest the reactor building. Air conditioning for the rest of the facility as well as heating for the entire facility is located in the mechanical equipment space near the west end of the laboratory building. (See Fig. 1.2-6.)

Fig. 1.2-6. Floor plan of Building 281 and associated structures.

All mechanical utility lines and electrical power distribution panels for both buildings are centralized in the mechanical equipment room.

1.3 Comparison Tables

1.3.1 Similar Facility Design

Table 1.3 provides a summary of research reactor facilities with similar core characteristics, peak thermal flux levels or temperature ranges. The list is not

Table 1.3. Research reactor facilities with core characteristics, peak thermal flux levels, or temperature ranges similar to the LPTR.

	Primary core material	Power (MW)	Start-up date	Type	Designer	Containment	Reflector	Core coolant (temp., °C)		NTR fuel enrichment, in 2.35	Max thermal flux = 10 ¹⁵ n/cm ² -sec
								in	out		
BSR Bulk Shielding Reactor	Al	2	1950	Pool	ORNL	Corrugated metal bldg.	HeO	19 (20) 34(M)	90	95	1
BRR Battelle Research Reactor	Al	2	1956	Pool	AMF	None other than reactor bldg.	H ₂ O	31 (20) 34(M)	30	90	1.0
LPR Lynchburg Pool Reactor	Al	1	1958	Pool	B/W	Steel con. bldg. concrete block	H ₂ O	- ^b	- ^b	20	0.3
UVAR Univ. of Virginia	Al	2	1960	Pool	U/V/B/W	Reactor bldg. reinforced masonry	C ^b	34	42	93	2.3 ^c
MRR Brookhaven Medical Research Reactor	Al	1	1959	Tank	Nuclear Div., Assn. of Amer.	Steel cylinder with domed roof (14 kPa internal press. possible)	C ^b	38	44.5	90	1
FNR (Univ. of Mich) Ford Nuclear Research	Al	1	1957	Pool	B/W	Windowless concrete bldg. sealed doors/dampers (0.3 kPa overpress.)	H ₂ O C ^b	29 (20) 34(M)	32	90	1.4
HHLR Horace Hardy Lester Reactor For Material Research	Al	1	1960	Pool	USAMRA Denile Ac. Corp.	Steel cylinder (steel) with concrete (steel) press. of 14 kPa)	HeO	38	42	93	1 ^f
IRL Industrial Reactor Laboratories Inc., Research Reactor	Al	5	1959	Pool	AMF	Egg-shaped concrete bldg.	H ₂ O	38	42	90	1
OWR Omega West Reactor	Al	5	1956	Pool tank	Los Alamos	Concrete block bldg. plus reactor vessel (test press. 240 kPa)	He	40	46	95	3
UCNCR Union Carbide Nuclear Company Research Reactor	Al	5	1960	Pool	AMF	Reinforced concrete bldg.	H ₂ O	38	42	75	10
Buffalo Reactor Western New York Nuclear Research Center Reactor	Al	1	1961	Pool	AMF	Reinforced concrete cylinder air tight test 3.3 kPa overpress.	H ₂ O	-	-	90	1
ORR Oak Ridge Research Reactor	Al	20 (30 max)	1958	Tank	ORNL	Reactor bldg. Reactor contained in sealed tank	Be	49 ^d	55 ^d	90	40 ^d

^aNatural convection cooled.

^bGraphite.

^cAverage.

^dAt 20 MW.

intended to summarize all research reactors, as several have been decommissioned or are significantly different in safety considerations.

1.3.2 Comparison of Current to Previous Designs

The previous SAR¹ furnished a reactor description and safety analysis of the entire system for 3-MW operations as of November 1966. The current SAR is merely an update of that document.

Minor changes completed since the 1966 SAR include:

- a) Various experimental shielding arrangements around the reactor faces.
- b) Pneumatic rabbit system previously located in the east thermal column (E. T. C.) nose piece is now located in core reflector position A-7. The E. T. C. rabbit facility is now plugged.
- c) The west thermal column (W. T. C.) previously contained an integral 6-in. lead nose piece. It now has a separate lead nose piece of similar dimensions but with much better heat removal.
- d) The E. T. C. now contains a void in the graphite on a line parallel to the center line of the thermal column. This cavity along with a 0.55-m hole in the thermal column door permits full size neutron radiography. The neutron radiograph facility is decoupled from the reactor core sufficiently to eliminate damage to the reactor core by shock waves or projectiles caused by an object exterior to the graphite. Refer to Ch. 10.0 for an amplified treatment of this facility.
- e) Recirculation lines (1/2 in. Al) connecting outer zones of beam tubes and thermal columns to upper portions of the reactor tank were sealed with epoxy under pressure.

1.3.3 References

1. J. W. Bolstad, G. E. Cummings, J. L. Eagan, and H. F. Finn, Safety Analysis Report for the Livermore Pool Type Reactor, Lawrence Livermore Laboratory, Rept. UCRL-50147 (1966).

1.4 Identification of Agents and Contractors

The LPTR is owned by the Atomic Energy Commission and operated by the University of California under Contract No. W-7405-ENG-48 at Lawrence Livermore Laboratory (LLL). The LPTR was constructed for the Atomic Energy Commission under Contract AT(04-3)-70 which was awarded to Foster Wheeler Corporation in July 1955.

Major subcontractors and vendors were:

Airmatic Systems, Inc.	Pneumatic system, fabrication and installation
Austin Co. or Utah Construction Co.	Building construction
Babcock & Wilcox ^o	Fuel elements and control rods
Chicago Bridge and Iron Co.	Steel reactor building
Detroit Hoist and Machine Co.	Hoist
Illinois Water Treatment Co.	Demineralizer
Knapp Mills Inc.	Lead
Leeds and Northrup Co.	Control actuator instrumentation
Moffett Engineering Co.	Crane
Speer Carbon Co.	Graphite
Reactor Experiments, Inc.	Fast pneumatic system components

1.5 Requirements for Further Technical Information

Since this document merely updates an existing SAR, and covers operation at a power level already licensed, no further technical information is required.

1.6 Material Incorporated by Reference

J. W. Bolstad, G. E. Cummings, J. L. Eagan, and H. F. Finn, Safety Analysis Report for the Livermore Pool Type Reactor, Lawrence Livermore Laboratory, Rept. UCRL-50147 (1966).

*Several vendors have since supplied fuel elements for the LPTR.

CHAPTER 2.0 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Location

The Livermore Pool Type Reactor (LPTR) is located in the northwestern corner of the grounds of the Lawrence Livermore Laboratory in the eastern end of the Livermore Valley near the eastern edge of the city of Livermore. This valley, one of those generally referred to as the California coastal valleys, lies in Alameda County approximately 64 km east-southeast of San Francisco, California. More specifically, the LPTR building is located at Long. $121^{\circ} 42' 33''$ W, and Lat. $37^{\circ} 41' 27''$ N. The Universal Transverse Mercator Coordinates are: 4 172.1 km north and 613.8 km east in Zone 10. The reactor building is at an elevation of 180 m above sea level.

2.1.2 Site Description

The grounds of the Lawrence Livermore Laboratory, which comprise the exclusion area, contain 2.6 km^2 (640 acres) located as the whole of Section 12 of Township 3S, R2E, MDBM. During World War II the site was a Naval air station known as Waggoner Field.

The site is surrounded by ranges of hills and mountains. Those to the west, separate it from the San Francisco Bay Area, and those to the north, are up to 600 m high. Those to the south, however, climb to heights of 1200 m. The eastern ranges, separating the Livermore Valley from the central San Joaquin Valley, rise to 460 m.

Prominent geographical features near the site include Del Valle Dam and Reservoir located 14 km due south, South Bay Aqueduct which runs from northeast to southwest at a closest distance of 1.8 km to the southeast, the Clifton Court Forebay about 24 km to the northeast, Mt. Diablo about 31 km northwest, and Mt. Hamilton about 39 km to the south.

The center of the city of Livermore is about 5 km west of the site. The cities of Pleasanton and Dublin-San Ramon Village are approximately 16 and 19 km distant respectively from the site in a westerly direction. The cities of Danville and Alamo are about 34 km west-northwest of the site. To the west, beyond a range of hills, lie the cities of Fremont, Hayward, Castro Valley and San Leandro. These cities all lie within a radius of between 30 and 40 km of the site. Figure 2.1-1 shows the relation of Livermore to the surrounding area. Figure 2.1-2, a larger scale map of the area, shows the prominent local features, locations of LPTR structure, and exclusion boundaries.

Fig. 2.1-1 Relationship of LLL to surrounding urban areas, transportation routes, and other geographical features. Arcs are at 16 and 80 km from the LLL site.

Fig. 2.1-2 Location of LPTR with respect to prominent local features and the LLL exclusion boundaries.

2.1.2.1 Exclusion Area Control – The Laboratory has a well developed Security Force with authority to control access to the site, to exclude any personnel and property, and to determine the nature and conduct of any activity or operation carried out on the site. No portion of this site is penetrated or traversed by any public right of way or public transportation facility. In the event of an emergency occurring on the site, the Security Force is authorized and directed to block off and clear unnecessary personnel from affected areas as indicated in the Emergency and Disaster Control Plan in effect at the Laboratory. Local security measures are described in Sec. 13.7.

2.1.2.2 Boundaries for Establishing Effluent Release Limits – Figure 2.1-3 shows the location of the exclusion area boundaries which are to be taken as the airborne effluent release boundaries. As may be seen, the nearest open water supply facility is the South Bay Aqueduct located approximately 140 m from the southeastern corner of the exclusion area or approximately 1.8 km southeast of the LPTR. The reservoir and water treatment plant of the Zone 7 Water District is located 1.1 km east of the northeastern corner of the exclusion area or approximately 2.4 km east-northeast of the LPTR location. Otherwise there are no lakes, reservoirs, or flooded gravel pits providing ready access to the underground aquifer within 8 km of the reactor site.

Figure 2.1-3 also shows part of the drainage of the Livermore Valley. Two dry creeks, Arroyo Seco and a branch of Arroyo Las Positas, cross the site at the southwest and northeast corners respectively. These are dry except in periods of heavy rainfall. There are, however, several small creeks flowing through the area contained within the 8 km radius. These ultimately combine to flow into Alameda Creek and thence into the Niles Cone aquifer, the water supply source for the city of Fremont. However, the amount of flow supplied by these creeks is small in comparison to the total flow into this aquifer and is accomplished over distances of 16 to 32 km before the combination discharges into Alameda Creek. There is adequate dilution and dispersion so that release of radioactive air borne waste from the LPTR would not be detectable in the water flowing to the Niles Cone aquifer. Refer to Ch. 11.0 for discussion of radioactive waste and in particular to Sec. 11.3 for gaseous wastes.

2.1.3 Population and Population Distribution

Population figures in this section are based upon the 1970 Census. Data on population and its distribution for areas more than 16 km distant from the LPTR (Sec. 2.1.3.2) have been omitted because the LPTR operates at a low power (3 MW) hence possesses a small fission product inventory.

Projections of population have not been carried beyond the year 1990 since it is believed that the reactor will have ceased operation before that date. These projections are based upon an annual average growth rate for Livermore and environs of 5%.

Fig. 2.1-3 LLL airborne effluent release boundaries.

Further it is assumed that there will be no war or other catastrophe, natural or man-caused, disruptive enough to affect the basic conditions of life. This annual rate of 5% is believed to be reasonably accurate and the projected population totals for the census districts involved match well with similar totals for these districts projected in other studies made by Alameda County and the State of California.^{1,2}

2.1.3.1 Population Within 16 km — Figure 2.1-2 shows the LPTR Site and its environs. It has been divided into radial zones of radii 1.6, 3.2, 4.8, 6.4, 8.0, and 16.0 km centered on the LPTR and further subdivided into angular zones of 0.393 rad each, centered on the cardinal points of the compass. The number of each zone on the map is keyed to the population distribution table (Table 2.1-1) so that population data may be easily determined for each zone. Total area population for 1970 was 37 552. The projected populations for 1980 and for 1990 are 63 000, and 109 000, respectively.

Table 2.1-1 Population distribution within 16 km radius of LPTR.

Segment No.	Total residents			Rate %		Segment No.	Total residents			Rate %
	1970	1980	1990				1970	1980	1990	
1	0	0	0	—		17	15	39	101	10
2	0	0	0	—		18	28	34	42	2
3	0	0	0	—		19	10	16	27	5
4	0	10	16	5		20	20	33	53	5
5	0	6	10	5		21	10	16	27	5
6	4	7	11	5		22	0	10	16	5
7	2	3	5	5		23	50	81	133	5
8	3	5	8	5		24	18	2 ^a	48	5
9	3	8	20	10		25	0	20	33	5
10	30	49	80	10		26	50	81	133	5
11	25	32	41	2,5		27	60	98	159	5
12	8	21	54	10		28	3 322	5 411	8 814	5
13	35	91	235	10		29	1 622	2 642	4 304	5
14	0	24	39	5		30	24	62	161	10
15	0	0	0	—		31	353	916	2 375	10
16	0	0	0	—		32	449	731	1 191	5
Totals	110	256	519							

Table 2.1-1 (continued)

Segment No.	Total residents			Rate %		Segment No.	Total residents			Rate %
	1970	1980	1990				1970	1980	1990	
33	118	192	313	5		65	10	16	27	5
34	15	24	40	5		66	15	24	40	5
35	12	20	32	5		67	20	33	53	5
36	15	24	40	5		68	10	16	27	5
37	24	39	64	5		69	5	8	13	5
38	22	36	58	5		70	0	10	16	5
39	90	147	239	5		71	10	16	27	5
40	40	65	106	5		72	0	10	16	5
41	12	20	32	5		73	35	57	93	5
42	70	114	186	5		74	4	7	11	5
43	50	81	133	5		75	55	90	146	5
44	3 321	5 410	8 812	5		76	6 736	10 972	17 873	5
45	2 749	4 478	7 294	5		77	4 562	7 431	12 104	5
46	135	220	358	5		78	24	149	920	20
47	1 314	3 408	8 840	10		79	15	24	40	5
48	762	1 241	2 022	5		80	40	65	106	5
49	28	46	74	5		81	2	3	5	5
50	0	10	16	5		82	15	24	40	5
51	24	39	64	5		83	20	33	53	5
52	12	20	32	5		84	100	163	265	5
53	12	20	32	5		85	6	10	16	5
54	9	15	24	5		86	10	16	27	5
55	105	171	279	5		87	25	41	66	5
56	6	10	16	5		88	30	49	80	5
57	135	220	358	5		89	10	16	27	5
58	0	10	16	5		90	30	49	80	5
59	82	213	552	10		91	430	701	1 143	5
60	3 086	5 027	8 188	5		92	737	1 912	4 958	10
61	6 711	10 932	17 806	5		93	1 120	2 905	7 535	10
62	43	266	1 649	20		94	45	73	119	5
63	24	62	161	10		95	15	24	40	5
64	26	42	69	5		96	10	16	27	5
Grand totals							37 552	63 152	108 798	

2.1.3.2 Population Between 16 and 80 km — Not pertinent, see Sec. 2.1.3.

2.1.3.3 Low Population Zone — In analyzing the effects of a Design Basis Accident (DBA), Sec. 15.4, dose rates were calculated for specific distances from the reactor.³ The nearest population center was taken as 1.6 km to the west. The number of residents shown in segments 1-16, Table 2.1-1, a 1.6 km radius easily satisfies the definition of a low population zone for several years hence.

Fig. 2.1-4 Cumulative population vs distance.

Figure 2.1-4 shows a plot of cumulative population with respect to distance. These curves compare favorably to those of power reactors.⁴ In addition it should be recognized that the LPTR fission product inventory is about 0.01% that of a power reactor.

Integrated doses to the thyroid from radioiodines are a small fraction of the allowed 300 rem dose for a low population zone,⁵ see Table 15.4-6. Integrated whole body immersion doses were also a small fraction of the 25 rem allowed for a low population zone, see Table 15.4-7.

2.1.3.4 Transient Population — There are no major seasonal shifts in population in this area. There are, however, significant changes in the population of the exclusion area and low population zone as a result of the day shift at LLL and Sandia arriving for work at 0800 and leaving at 1645. The extent of this change is indicated in Fig. 2.1-5, a large scale map of the area encompassed in the low population zone, showing the zones formed by the 1.6 km radius and the 0.393 rad sectors with the reactor as a center. Each zone thus formed has a number keyed to Table 2.1-2 which shows the day vs night population.

Table 2.1-2 Day vs night population in the low population zone.

Zone No.	Day				Night			
	Residents	LLL	Other	Total	Residents	LLL	Other	Total
1	0	0	0	0	0	0	0	0
2	0	0	0	0	0	0	0	0
3	0	0	0	0	0	0	0	0
4	0	5	0	5	0	0	0	0
5	0	0	0	0	0	0	0	0
6	4	0	0	4	4	0	0	4
7	2	609	0	611	2	32	0	34
8	3	953	675	1 631	3	238	75	316
9	3	2 474	0	2 477	3	275	0	278
10	30	868	0	898	30	46	0	76
11	25	37	0	62	25	5	0	30
12	8	18	50	68	8	0	0	8
13	35	12	0	47	35	6	0	41
14	0	22	0	22	0	2	0	2
15	0	15	0	15	0	5	0	5
16	0	0	0	0	0	0	0	0

Fig. 2.1-5 Transient population in the Low Population Zone (LPZ).

2.1.3.5 Population Center Distance – The nearest population center is the City of Livermore with a 1970 census population of 37 703. The nearest boundary is 1.2 km WSW and the city center about 5.6 km west of the LPTR.

2.1.3.6 Public Facilities and Institutions – Public facilities and institutions located within 16 km of the LPTR have been grouped into three categories: schools, other public facilities, and parks. Table 2.1-3 lists the public schools and major private

Table 2.1-3 Livermore school population (est. 1972-1973).

School name	Location		Staff (total)	Student (A.D.A.) ^a	Total popu- lation	Classification
	Dist. (km)	Dir.				
Christensen	5	NNW	15	361	376	Elem.
Green	3	N	15	194	209	Elem.
Jackson Avenue	3	W	29	663	692	Elem.
East Avenue	4	WSW	37	674	711	Jr. High
Almond Avenue	3	WSW	29	576	605	Elem.
Fifth Street	7.7	WSW	31	539	570	Elem.
Junction Avenue	7.7	W	49	943	992	Jr. High
Portola Avenue	5.6	W	38	660	698	Elem.
Rincon Avenue	6.8	W	45	687	732	Elem.
Marilyn Avenue	7.2	W	32	718	750	Elem.
Joe Michel	7.2	WSW	35	775	810	Elem.
Sonoma Avenue	8.0	WSW	34	722	756	Elem.
Mendenhall	8.4	WSW	53	1 000	1 053	Jr. High
Emma C. Smith	8.7	WSW	43	717	760	Elem.
Arroyo Seco	2	W	11	252	263	Elem.
Henry	8	W	15	200	215	Elem.
Arroyo Mocho	6	WSW	23	402	425	Elem.
Livermore High	5	W	114	1 971	2 085	High
Granada High	7.6	WSW	114	1 835	1 949	High
Del Valle High	8	W	6	90	96	High
St. Michael's	5.3	W	13	330	343	Parochial
Seventh Day Adventist	6.0	SW	2	28	30	Parochial
Ladd Avenue	5.2	W	6	16	22	Special (M.R.)
Pentecostal Bible College	6.0	WSW	12	92	104	College
UC Davis Advanced Studies	(on site)		13	84	97	Colleg Grad. (study)
Ark School	2.6	WSW	7	45	52	Private
Ark Preschool	3.5	WSW	7	59	66	Private
					15 343	

^aA.D.A. = average daily attendance.

schools. This table shows the estimated 1972-73 average daily attendance figures for student population, the instructional and service staff, and the total population for each school, as well as its location with respect to the LPTR.

Table 2.1-4 lists the other facilities such as hospitals, libraries, and prisons and gives their location, average population of users or inmates, number on staff and total population for 1972-73. Table 2.1-5 gives similar data for parks. The Livermore City Jail is used only as a detention facility. Since its average population is nearly zero it has been omitted from Table 2.1-4.

Table 2.1-4 Population data for hospitals, prisons, and libraries.

Institution	Location		Staff	Users	Total pop.
	Dist. km	Dir.			
Valley Memorial Hospital	6.8	WSW	260 ^a	60	320
Casa Linda Convalescent Home	5.0	W	19 ^a	20	39
Hacienda Convalescent Hospital	6.8	WSW	60 ^a	83	143
Pines Guest Home	4.5	WSW	6	20	26
Livermore Valley Convalescent Hospital	6.8	WSW	18 [?]	37	55
Veterans Administration Hospital	8.5	SSW	630 ^a	164	794
Livermore Public Library	4.8	WSW	4	100	104

^aStaff divided approximately as follows: 60% day, 20% swing shift, 20% night shift.

Table 2.1-5 Population data for parks and recreational facilities.

Park or facility	Location		Operating agency	Day pop.	Night pop.
	Dist. km	Dir.			
Livermore Recreational Center	6.0	WSW	LARPD	200	130
Del Valle Park ^a	8.9	S	EBRPD	1000	200
Shadow Cliffs Park	11	WSW	EBRPD	400	0
May Nissen Park ^b	7.2	W	LARPD	600	0

^aVaries from winter to summer, e.g., 100/day in winter to 1000/day in summer. The table gives maximum summer value.

^bSummer maximum value. For winter use the figure drops to ~430.

2.1.4 Uses of Adjacent Lands and Waters

In applying the requirements of this subsection to the LPTR it should be noted that the LPTR releases no liquid effluents from either reactor cooling water or process water. All contaminated or potentially contaminated liquids are handled by special procedures described under Liquid Waste Systems, Sec. 11.2.

Examination of the maps will show that the only bodies of water that lie within an 8 km radius of the LPTR are the South Bay Aqueduct, the Patterson Pass water treatment facility, and Frick Lake. Frick Lake is dry most of the year. Hence only the aqueduct and the treatment plant represent ways for airborne radioactivity to contaminate potable water supplies.

These facilities are located almost directly east and about 2.4 km from the LPTR. The prevailing winds are essentially from the west. This water treatment plant supplies from 0.13 to 0.39 m³/s to the City of Livermore water distribution system. Another treatment plant is planned to be located on the South Bay Aqueduct just west of the Veteran's Administration Hospital in the near future. It will supply an average of 0.53 m³/s of water to the cities of Livermore and Pleasanton. The South Bay Aqueduct itself carries about 82.1 m³/s total.

Del Valle Reservoir is a part of the State of California Water Project serving as a storage and surge reservoir for the South Bay Aqueduct. Its waters are ultimately treated for human consumption and are also used for recreational purposes. Approximately 200 000 fish of all types are taken annually with about 55% being trout and 45% of other species. About 200 000 trout fingerlings and 70 000 catchable size trout are planted in the lake each year.

Shadow Cliffs Park has a number of flooded, abandoned gravel pits whose waters are used for swimming, fishing, and small boating. Species of fish found in the pits are the same as in Del Valle Reservoir except that there are no striped bass. Estimated yield from these pits was approximately 5000 fish in 1971 (first year of operation) with an estimated rise to a level of 7000 to 8000 fish annually of all species.

Commercial fishing is not allowed in either of these parks.

Principal agricultural products of the area near the LPTR are grapes and wine, cattle, poultry and eggs, and nuts. The principal products and amounts produced annually are summarized from the 1971 Agricultural Crop Report for Alameda County in Table 2.1-6.

Table 2.1-6 Livermore Valley agricultural crop report, 1971.

Item	km ²	Annual production	Estimated valuation (\$)
Wine grapes	4.9	2 820 Mg	466 200
Wine		1 582 m ³	2 508 000
Eggs		78 000 doz.	25 740
Poultry (fryers)		1 018 000	65 083
Beef cattle	142	15 000 head	1 955 624
Sheep and lambs		6 690 head	187 460
Feed grains	194	4 570 Mg	138 096
Alfalfa	0.12	150 Mg	5 890
Almonds	0.05	3.6 Mg	1 680
Walnuts	0.40	50 Mg	23 100
Hogs		50 head	2 255
Total agricultural valuation (1971)			5 969 128

2.2 Nearby Industrial, Transportation, and Military Facilities

2.2.1 Locations and Routes

Figure 2.2-1 is a map of the area surrounding the LPTR showing all industrial, transportation, and military installations within a radius of 8 km. There are no military bases, missile sites, military firing ranges, or large tank farms within this radius.

A glider field is located approximately 1.6 km to the northwest of the LPTR. Its landing approach runs east-west past the LPTR at a distance of about 0.8 km. Meadowlark Field, a small private airstrip, lies 3.5 km to the south-southeast. Its landing approach passes about 3 km to the south of the LPTR. The only other flight facility in the area is the Livermore Municipal Airport located about 10 km to the west of the LPTR. Its landing pattern is indicated in Fig. 2.2-1.

Three major transportation routes serving the San Francisco Bay Area pass within 8 km of the LPTR: Interstate Route 580, and tracks of the Western Pacific and the Southern Pacific railroads. Interstate Route 580, an eight-lane major east-west route to and from the Bay Area, runs about 2.4 km to the north of the LPTR. The tracks of the Southern Pacific and Western Pacific railroads also pass to the north of the reactor at distances of 1.4 and 0.8 km respectively.

Fig. 2

Fig. 2.2-1 Locations of industrial, transportation, or military installations near the LPTR site.

Two major gas transmission lines belonging to the Pacific Gas and Electric Co. run north and south at a distance of 2 km west of the LPTR. An east-west line runs about 1.6 km south of the reactor.

2.2.2 Descriptions

Traffic on Interstate Route 580 flows at an average rate of 30 500 vehicles daily (1972). This estimate was arrived at by dividing the California Division of Highways one-year traffic count at the First Street intersection by 365. Seasonal and daily variations in this number are to be expected since this is a major route for recreational travel to and from the San Francisco Bay Area. The two railways accommodate an average of 15 freight trains per day. Materials transported on these three routes could include explosives, fuels, chemicals, toxic gases, radioactive substances, and any other material allowed by I.C.C. regulations to be transported by rail or truck.

The natural gas transmission lines are 0.41 and 0.61 m diam and operate at pressures of 2.4 and 3.45 MPa, respectively.

Livermore is not a highly industrialized area at present. A list of principal enterprises, their products or services, and a map key are given in Table 2.2-1.

The nearest oil and gasoline storage facility is that of the Phillips Petroleum Co. located about 3.2 km west-southwest of the LPTR. Here 95 m^3 of gasoline and 72 m^3 of diesel fuel are stored in aboveground tankage. The Standard Oil Co. and Union Oil Co. bulk storage facilities are located adjacent to each other about 5 km west of the LPTR. At the Standard Oil facility 150 m^3 of gasoline, 110 m^3 of diesel fuel, and 38 m^3 of heating oil are stored in aboveground tankage with 11 m^3 of gasoline in underground tankage. At the Union Oil facility 90 m^3 of gasoline and 120 m^3 of diesel fuel are stored in underground tankage. The Hansen Fuel Co. facility stores 150 m^3 of propane and 100 m^3 of heating oil at a location about 3 km west-northwest of the LPTR in aboveground tankage.

Of the organizations listed in Table 2.2-1 only L.I.L. and G.E. Vallecitos Laboratory present much of a hazard to the LPTR.

The hazard at G.E. Vallecitos stems mainly from the operation of nuclear reactors and the consequent accumulation of fuel and radioactive wastes. At L.I.L. more hazards, such as high explosives, flammable and explosive gases, toxic gases, fissionable materials, and radioactive materials are present. The hazards come about due to the proximity of these materials to the LPTR. Sandia Laboratories has similar types of materials but in smaller quantities and farther away from the LPTR. A minor hazard

Table 2.2-1 Industrial enterprises, Livermore area, 1972.

Name	Map key	No. of employees	Products or services
Hexcel Corp.	1	83	Fiberglass reinforced plastic products
Kaufman and Broad Ind.	2	75	Prefab. housing components
Livermore Data Systems	3	40	Information processing systems
Mobat Tire and Rubber Co.	4	20	Recapping of large off-road tires
Sandia Laboratories	5	1000	Nuclear research and development
Lawrence Livermore Laboratory	6	5100	Nuclear research and development
Sunshine Art Studio	7	50	Manufacture of greeting cards
Control Data Corp.	8	9 ^a	Computer service
Lox Equipment Co.	9	75	Cryogenic containers
Unimark Corp.	10	5	Printing equipment
Concannon Vineyards	11	30	Winery
Wente Bros.	12	45	Winery
Schenley Distillers	13	3	Winery
Wagner and Sons, Inc.	14	15	Diesel engine parts
G.E. Vallecitos Atomic Laboratories (16 km SW)	15	550	Nuclear reactor research and development
Associated Professions Inc.	16	20	Subdivision planning and engineering
Pacific Telephone and Telegraph Co.	17	255	Communications
Pacific Gas and Electric Co.	18	24	Energy generation and distribution

^aPlus 17 located at LLL and 5 located at Sandia Laboratories.

could come from the combustion of considerable quantities of resins, plastics, plasticizers, and catalysts kept at Hexcel Corporation.

2.2.3 Evaluations

The close proximity (0.8 km) of the glider landing pattern, we feel, constitutes only a slight hazard since the field is used mainly on weekends when the LPTR is shut down. Further, the light weight and construction of the aircraft using the field renders it highly unlikely that they could inflict serious damage on the LPTR containment building.

The nearest scheduled commercial air service is from San Jose, Oakland, and Stockton airports. Nonscheduled commercial aircraft operate from the Livermore Municipal Airport, using a controlled approach and departure pattern. The low-level flight and

landing pattern is far enough from the LPTR to constitute only a very minor hazard to it from aircraft striking the containment building.

The evaluation of possible hazards to the LPTR from the proximity of Interstate 580 is extremely difficult because of the almost constantly-changing amounts and types of hazardous materials transported through the vicinity. The most one can say is that the violence of chemical, fuel, or munitions explosions occurring on the route is expected to be restricted by the limited amount transportable on a single vehicle. The distance of 2.4 km should mitigate the blast effect considerably. Fires occurring there would not affect the LPTR.

The explosion of chemicals, fuels, or munitions on either of the two railway lines could pose a hazard to the LPTR because of both the capacity for much greater amounts and the proximity (1.4 and 0.8 km) to the LPTR. In this regard a fire and explosion such as the one involving 15 tank cars of propane at Laurel, Miss. on January 25, 1969, which damaged buildings about 1.5 km away would almost certainly have an adverse effect on the LPTR. However, it is felt that events of this magnitude are rare enough to make the relative proximity of the railway lines a small hazard to the LPTR. Also, the I.C.C. regulations and revised safety rules of the railroads should limit the amounts per car and the proximity to other cars containing flammable or potentially explosive chemicals or gases.

At Hexcel Corporation, fire resulting in the combustion of plastics, resins, plasticizers, and catalysts could release toxic gases which could then be blown toward the LPTR. This is not regarded as more than a slight hazard since the distance of 3.2 km separating the sites allows atmospheric diffusion to reduce the concentration to a low value by the time the toxic gases arrive at the LPTR. Also the availability of self-contained breathing apparatus at the reactor should allow its safe shutdown in case of such an event. No explosion hazard large enough to affect the LPTR exists at Hexcel Corporation.

LLL itself presents the largest potential hazard to the safe operation of the LPTR because of the wide range of hazardous materials kept and used in the normal operation of the Laboratory. The hazards at LLL will be broken down into three categories: explosives, radiation and toxic substances, and fire.

Explosive exposure results from the storage of kilogram quantities of high explosives as indicated on the map of Fig. 2.2-2. The nearest of these is located near Bldg. 165 approximately 0.4 km south-southwest of the LPTR. The other is located near Bldg. 222 about 1.2 km south of the LPTR. The explosives are stored in well protected and managed magazines under stringent safety procedures. They should constitute only a slight hazard to the LPTR in case of an accident. Gram quantities of high explosives

Fig. 2.2-2 Explosive material storage and use at LLL.

in the form of detonators are stored in a magazine in Bldg. 344 but this should not constitute a hazard to the LPTR.

Possible explosive exposure could result from accident to any of three hydrogen gas storage banks on site: One near Bldg. 131, one near Bldg. 241, both located about 0.8 km south of the LPTR with each storing 1000 m³ of hydrogen gas. Another located at Bldg. 321 about 1.1 km south-southeast of the reactor also stores 1000 m³. A bank at Bldg. 410 about 1.3 km south-southeast of the LPTR stores 2000 m³ of oxygen. Damage occurring to this bank with the consequent release of large quantities of oxygen could affect the LPTR. Other flammable and potentially explosive gases are stored in

Bldg. 518 (Gas Dock). A storage facility holding 190 m^3 of propane for emergency fuel supply is also located near here. Because of the distance of approximately 1.6 km separating this area from the LPTR it is felt that the effect on the LPTR of explosion or fire occurring here would be small.

The nearest bulk storage facility for gasoline is 3 km west of the LPTR, thus fire or explosion occurring in these facilities should have little effect on the LPTR.

Exposure to radiation and toxic substances results from the fact that many different types of radiation sources and kinds of toxic materials are stored, handled, and used at LLL in the course of normal laboratory operation. The map of Fig. 2.2-3 shows the

Fig. 2.2-3 Potential sources of exposure to radioactive and toxic materials.

locations of and major types of these hazards at LLL. Potentially toxic gases are stored in and distributed from Bldg. 518 (Gas Dock). These gases would disperse in event of accidental release and sufficient warning coupled with the use of self-contained breathing apparatus would allow the LPTR to be safely shut down until decontamination of the area was completed.

A small amount of plutonium is kept and used in Bldg. 194 immediately adjacent to the LPTR. Kilogram amounts of plutonium are stored and handled in Bldg. 332 about 0.8 km to the southeast of the LPTR. Building 231 located about 0.8 km south of the LPTR contains our central radioactive materials storage facility where kilogram quantities of uranium and plutonium as well as large quantities of other radioactive materials are stored. Hazards to the LPTR from these two buildings do exist. However, stringent procedures and safety measures are employed to prevent accidents and to minimize the effects if an accident should occur. The major hazards to the LPTR from these sources would be airborne radioactive contamination, but the reactor could be safely shut down and personnel evacuated until cleanup was completed. Small amounts of plutonium are stored and used in chemical experiments in Bldg. 222 but the limited quantities make this a negligible hazard to the LPTR.

In Bldg. 251 large (megacurie) amounts of transuranic elements are stored and handled making it possible for the LPTR to become contaminated in case of accident. In this building also, however, strict operating and safety procedures are in force to prevent and limit the consequences of accidents.

In Bldg. 331 large quantities of tritium are stored and handled presenting a possible contamination exposure. This could result either from direct accidental release or from accidental ignition and combustion with subsequent release of contamination to the atmosphere. Stringent control procedures are used in this facility to prevent and to limit the results of accidents.

Exposure from radioactive sources could come from Bldg. 412 (Hot Cells) where large quantities of radioactive materials are frequently handled, also from several large ^{60}Co sources located as follows: a 5 kCi source located in Bldg. 164, a 2 kCi source in Bldg. 239, and a 100 kCi source in Bldg. 255. However, we feel that these ^{60}Co sources are well enough contained (Class IV sealed sources) and operated conservatively enough to make them negligible hazards to the operation of the LPTR. Building 412 does present some hazard since some of the radioactive materials handled there are in powder form which can be spread easily in case of accidental escape. Here again, strict procedures and safety measures are employed to prevent and limit the consequences of accidental spills or releases. In event of accident, sufficient warning could be given to enable the LPTR to be shut down safely.

Fire is considered a minor hazard due to the fire resistant construction of the reactor building itself and its relative isolation from significant sources of combustible materials. The nearest appreciable sources of combustible materials, e.g. gases, gasoline, etc., are 0.8 km or more distant and the quantities stored seem insufficient to pose a hazard to the reactor. The reactor and its laboratory building are surrounded by a large expanse of paving, thus limiting the possibility of damage arising from brush or grass fires in adjacent fields.

Airborne pollutants are not expected to have any appreciable effect on the LPTR system except for shortening the life of rubber and plastic components and making necessary their testing and replacement on a regular schedule.

2.3 Meteorology

2.3.1 Regional Meteorology

2.3.1.1 General Climate – The general area has a "Mediterranean scrub woodland" climate⁶⁻⁸ which is characterized by mild, rainy winters (about 0.38 m of rain from November to April) and warm, dry summers. Sunshine is abundant throughout the year since the winter rains are generally showery. Snow is rare.

Winter storms are a result of migratory low pressure systems that become detached from the semipermanent Aleutian Low and move over or north of the area. Following the passage of the migratory low, skies typically clear as the Eastern Pacific High builds inland. Occasionally under these conditions strong northerly surface winds with gusts up to 27 m/s are observed for a day or two.

The summer is consistently warm and dry. A sea breeze typically develops during the afternoon when modified ocean air moves through the passes to the west; although the effect upon maximum temperatures is slight, the breeze persists into the early evening and brings cool nights. The strength of this sea breeze rarely exceeds 13 m/s in Livermore. The spring and autumn seasons are typically transitional periods when no exceptional meteorological phenomena occur.

2.3.1.2 Severe Weather – The Livermore Valley rarely experiences severe weather. The greatest recorded daily rainfall is about 88 mm. Thunderstorms occur less than 5 d/yr and are not intense; hail occurs even less frequently. The probability of tornadoes has been estimated as about 1 in 300 000 yr.⁹ Strong winds with gusts to about 27 m/s occur a few times each fall and winter, usually following the passage of a low pressure system. Damage from such winds is rare. Dust devils occur at times on warm summer afternoons. They are small and usually pass almost unnoticed.

2.3.1.3 Air Pollution Potential – Because of its location, surrounded by hills, the Livermore Valley has more days of high photochemical smog levels (oxidant) than most of the other air pollution sampling stations in the San Francisco Bay Area. Such smog is generated from automobile emissions; it is estimated that about one-half of the ambient Livermore smog is imported into the Valley from metropolitan areas.¹⁰

2.3.2 Local Meteorology

2.3.2.1 Station Summary – Table 2.3-1 is a copy of the National Weather Service compilation for Livermore¹¹ based on data observed from 1931-1960. Temperature and precipitation data observed during 1969-1971 at the County Fire Department, about 5.6 km to the west-southwest of the Laboratory, provide a check on these normals, Table 2.3-2. The data source was the Climatological Data for California, published monthly by the National Oceanic and Atmospheric Administration (NOAA). It is apparent that these two tables are similar enough to rule out significant climatic change. The Lawrence Livermore Laboratory is similar to the NOAA substation at the Fire Department in topography, soil type, etc., so that the values in Table 2.3-1 may be considered representative of the site.

Table 2.3-1 Climatological summary for Livermore, Alameda County, 1931-1960: elevation 166 m, latitude 37° 39', longitude 101° 47'.

	No. years	Jan	Feb	Mar	Apr	May	June	July	Aug	Sept	Oct	Nov	Dec	Year
Temperature, °C														
Highest	30	24	25	27	34	39	43	44	44	46	39	32	26	46
Mean daily maximum	30	13.5	15.5	17.9	21.4	24.7	28.2	32	31.3	30.6	25.6	19.5	14.6	22.9
Mean daily	30	7.8	9.4	11.3	13.8	16.6	19.4	22.1	21.6	20.8	17.0	11.9	8.6	15.1
Mean daily minimum	30	2.1	3.3	4.7	6.3	8.5	10.6	12.1	11.6	11.0	8.4	4.3	2.7	7.2
Lowest	30	-7.2	-5.0	-3.8	-1.1	0	4.4	5.0	5.0	1.7	-1.6	-3.0	-5.6	-7.2
Degree days ^a	30	586	448	391	264	133	33	0	0	12	90	348	543	2890
Days above 32°C	19	0	0	0	+	3	8	17	14	12	4	+	0	58
Days below 0°C	19	15	9	4	+	+	0	0	0	0	+	4	13	46
Precipitation, mm														
Greatest monthly	30	208	172	129	95	70	15	6	4	48	47	151	258	258
Mean monthly	30	76	68	49	29	12	3	0.2	0.5	3	15	37	73	366
Least monthly	30	5	1	0	1	0	0	0	0	0	0	T ^b	12	0
Greatest daily	30	88	37	51	46	18	10	4	4	39	21	77	83	88
Days 0.3-3 mm	30	11	10	9	5	3	+	+	+	1	3	6	10	59
Days 3 mm or more	7	7	6	4	4	2	+	0	0	+	1	2	5	31
Days 13 mm or more	10	2	2	1	1	+	0	0	0	+	+	1	3	10
Snowfall	30	3	T ^b	T ^b	0	0	0	0	0	0	0	T ^b	T ^b	3
Wind	Calm 24%, W 19%, SW 16%, NW 7%, NE 12%.													

^aDegree day = No. of days $65^{\circ}\text{C} - \left(\frac{T_{\text{low}} + T_{\text{high}}}{2}\right)$.

^bT = trace.

Table 2.3-2 Summary of temperature and precipitation observations at County Fire Department, Livermore, California, 1969-1971.

	Month												Annual
	J	F	M	A	M	J	J	A	S	O	N	D	
Temperature (°C)													
Mean maximum	14.2	16.1	19.3	20.4	25.8	27.4	33.4	33.7	32.4	24.8	19.7	14.2	23.5
Mean minimum	3.0	3.1	3.4	4.0	8.4	10.8	11.9	11.4	10.7	6.5	4.3	2.3	6.7
Average	8.6	9.6	11.4	12.3	17.2	19.1	22.7	22.6	21.6	15.7	12.0	8.3	15.1
Absolute maximum	21	24	28	32	39	37	42	42	43	37	20	21	43
Absolute minimum	-5.5	-3.3	-5.5	-1.6	2.2	4.4	7.2	6.7	2.0	-1.6	-3.8	-4.4	-5.5
Precipitation (mm)	109	53	31	25	6	3	0	0	1	13	52	92	386

2.3.2.2 Wind Direction and Speed – Winds have not been measured at NOAA's climatological substation in Livermore. Wind observations were taken at LLL for several years by instruments mounted atop a telephone pole. No maintenance records nor calibration reports of these instruments are available. Furthermore, trees gradually grew up around the pole during the past ten years. Consequently, these measurements are not deemed to be reliable for obtaining a representative wind summary. In 1971 an instrumented 40 m mast was installed in an open area of the Laboratory, about 280 m to the north of the reactor building, Sec. 2.3.3. Continuous readings at 10 m (about 33 ft) above the surface are available for 12 months and are summarized in Tables 2.3-3a to 3c giving the annual, dry season (May-October), and wet season (November-April) summaries for all atmospheric stability categories. The annual wind rose data in Table 2.3-3a agrees favorably with the annual wind rose data presented in the previous Safety Analysis Report¹² and the older data taken by the Livermore Civil Aeronautics Authority and the Navy Station.⁶

The predominant wind direction throughout the year and especially during the dry season is from the southwest through west. During the wet season, post-frontal anticyclonic flow occurs often enough to cause north-northeast and northeast winds of comparable frequency to the southwest through west directions. The most common windspeeds during all seasons are 5-7 m/s from the southwest through west. This relatively high speed is caused by channeling of the winds through passes in the hills to the west. The wet season winds from the north-northeast and northeast are most common in the 2-3 m/s range (3.5%), but winds of 11-16 m/s occur with nearly half that frequency. In general, the strongest winds blow during the wet season from the north-northeast and northeast.

2.3.2.3 Atmospheric Stability – The wind records mentioned above were used to estimate Pasquill-Gifford stability categories using the method described by Slade.¹³ These stabilities are summarized in like fashion in Table 2.3-4. In the dry season, D and E stabilities are of nearly equal prominence (about 70% combined frequency); during the wet season E stability is observed nearly 40% of the time. The wind

Table 2.3-3a Annual wind rose data, Lawrence Livermore Laboratory (April 1971-March 1972), frequency %.

Direction wind blows from	Wind speed m/s							Total	Av speed
	0.2-1.6	1.6-3	3-5	5-7	7-11	11-16	>16		
NNE	0.2	0.9	0.7	0.5	0.1	0.4	0.1	2.9	5.5
NE	0.2	1.4	0.9	0.9	0.3	0.2	0.1	4.0	4.8
ENE	0.2	1.7	0.9	1.5	0.2	0	0	4.5	4.0
E	0.2	2.4	1.1	1.3	0.3	0	0	5.3	3.8
ESE	0.2	1.8	0.7	0.4	0	0	0	3.1	3.0
SE	0.1	1.7	0.6	0.2	0	0	0	2.6	2.8
SSE	0.1	1.5	0.4	0.1	0.1	0	0	1.8	3.6
S	0.2	1.8	0.5	0.2	0.1	0	0	2.8	2.9
SSW	0.1	2.2	1.4	0.9	0.4	0.1	0	5.1	4.0
SW	0.4	4.4	4.7	5.3	2.3	0.2	0	17.3	4.7
WSW	0.2	4.4	5.8	6.2	1.9	0.1	0	18.6	4.7
W	0.3	3.4	3.9	5.0	2.3	0.1	0	15.0	4.9
WNW	0.1	1.8	1.3	1.2	0.4	0	0	4.8	4.1
NW	0.1	1.1	0.5	0.3	0.1	0	0	2.1	3.4
NNW	0.1	1.0	0.2	0.3	0.4	0.2	0.1	2.3	5.6
N	0.2	1.3	0.5	0.6	0.4	0	0	3.0	4.0
Calm	4.4	—	—	—	—	—	—	4.4	0
Total	7.3	32.8	24.1	24.9	9.3	1.3	0.3	100	4.2

Table 2.3-3b Dry season wind rose data, Lawrence Livermore Laboratory (May-October 1971), frequency %.

Direction wind blows from	Wind speed m/s							Total	Av speed
	0.2-1.6	1.6-3	3-5	5-7	7-11	11-16	>16		
NNE	0.2	0.3	0.6	0.2	0	0	0	1.3	3.4
NE	0.3	0.7	0.6	0.2	0	0	0	1.8	3.0
ENE	0.2	0.8	0.2	0.2	0	0	0	1.4	2.7
E	0.4	1.1	0.2	0	0	0	0	1.7	2.1
ESE	0.3	1.0	0.1	0	0	0	0	1.4	2.2
SE	0.2	0.5	0.1	0	0	0	0	0.8	2.1
SSE	0.1	0.7	0	0	0	0	0	0.8	2.1
S	0.4	1.6	0.2	0	0	0	0	2.2	2.1
SSW	0.2	2.8	1.5	1.1	0.6	0.1	0	6.3	4.0
SW	0.6	6.4	7.2	8.2	3.9	0.5	0	26.8	4.8
WSW	0.2	6.8	8.5	8.9	1.8	0	0	26.2	4.4
W	0.5	4.1	4.6	6.1	2.6	0	0	17.9	4.7
WNW	0.2	1.2	0.9	0.6	0	0.1	0	2.9	3.5
NW	0.2	0.7	0.1	0.1	0.1	0	0	1.2	3.0
NNW	0.1	0.5	0.1	0.4	0.8	0.3	0.2	2.4	8.0
N	0.2	0.8	0.5	0.6	0.5	0	0	2.6	4.5
Calm	1.9	—	—	—	—	—	—	1.9	0
Total	6.2	30.0	25.4	26.5	10.3	1.0	0.2	100	4.3

Table 2.3-3c Wet season wind rose data, Lawrence Livermore Laboratory (Nov. 1971-March 1972 plus April 1971), frequency %.

Direction wind blows from	Wind speed m/s						Total	Av speed
	0.2-1.6	1.6-3	3-5	5-7	7-11	11-16		
NNE	0.2	1.5	0.9	0.9	0.3	0.8	0.2	4.8
NE	0.2	2.0	1.2	1.5	0.7	0.5	0.1	6.2
ENE	0.2	2.5	1.7	2.8	0.5	0.1	0	7.8
E	0.1	3.7	2.0	2.5	0.7	0	0	9.0
ESE	0.2	2.7	1.2	0.7	0	0	0	4.8
SE	0.1	2.9	1.1	0.4	0	0	0	4.5
SSE	0.1	2.3	0.7	0.2	0.2	0	0	3.5
S	0.1	2.1	0.9	0.3	0.2	0.1	0	3.7
SSW	0.1	1.7	1.3	0.7	0.2	0	0	4.0
SW	0.1	2.7	2.6	3.0	1.0	0	0	9.4
WSW	0.1	2.3	3.6	4.0	2.1	0.2	0	12.3
W	0.1	2.8	3.6	4.2	2.2	0.1	0	13.0
WNW	0.1	2.3	1.8	1.8	0.7	0	0	6.7
NW	0.1	1.5	0.9	0.5	0.1	0	0	3.1
NNW	0.2	1.6	0.4	0.2	0.1	0	0	2.5
N	0.2	1.8	0.5	0.5	0.2	0	0	3.2
Calm	1.9	—	—	—	—	—	—	1.9
Total	4.1	36.4	24.4	24.2	9.2	1.8	0.3	100

Table 2.3-4. Percentage frequency of stability categories (1971-72) at Lawrence Livermore Laboratory.^a

Pasquill stability type	Annual	Dry season (May-October)	Wet season (November-April)
A	10.2	8.2	12.0
B	6.3	5.8	6.7
C	12.6	13.5	11.8
D	27.4	37.5	22.0
E	35.2	32.3	37.8
F	6.3	2.6	9.6

^aBased on measurements every 1.8 ks.

observations have been summarized by directions and speeds for stability categories A through F in Tables 2.3-5a to 2.3-7f. Annual tables are presented, together with those for dry and wet seasons. A summary of the dry and wet season tables is given in Table 2.3-8.

2.3.2.4 Humidity and Fog — Long term records of humidity and fog are not available in the Livermore Valley.

Table 2.3-5a "A" stability annual wind rose percentage frequencies.

Direction wind blows from	Windspeed, m/s							Total
	0.2-1.6	1.6-3	3-5	5-7	7-11	11-16	>16	
N	0.4	5.8	1.3	0.1	0	0	0	7.5
NNE	0.5	3.0	0.7	0.1	0	0	0	4.3
NE	0.5	4.7	1.1	0.2	0	0	0	6.4
ENE	0.2	3.2	0.4	0.1	0	0	0	3.8
E	0.4	4.8	1.1	0.2	0	0	0	6.6
ESE	0.3	1.9	0.1	0.1	0	0	0	2.4
SE	0.5	2.8	0.3	0	0	0	0	3.7
SSE	0.2	2.4	0.2	0	0	0	0	2.7
S	1.0	5.8	0.4	0	0	0	0	7.2
SSW	0.6	3.8	0.4	0.1	0	0	0	4.9
SW	0.7	6.6	1.3	0.1	0	0	0	8.7
WSW	0.4	7.8	2.8	0.2	0	0.1	0	11.3
W	0.6	8.2	2.4	0.3	0.1	0	0	11.5
WNW	0.7	5.4	1.9	0.2	0	0	0	8.0
NW	0.3	3.6	1.1	0.4	0.1	0	0	5.5
NNW	0.4	4.3	0.7	0.2	0	0	0	5.6
Total	7.5	73.9	16.1	2.3	0.2	0.1	0	100

Direction wind blows from	Windspeed, m/s							Total
	0.2-1.6	1.6-3	3-5	5-7	7-11	11-16	>16	
N	0.8	3.0	0.4	0.3	0	0	0	4.5
NNE	0.4	1.8	1.6	0.4	0	0	0	4.2
NE	0.3	4.1	1.5	0.4	0.1	0	0	6.3
ENE	0.4	2.8	0.8	0.3	0	0	0	4.3
E	0.3	4.5	2.1	0.3	0	0	0	7.1
ESE	0.1	3.2	1.1	0.1	0	0	0	4.5
SE	0.4	3.2	0	0	0	0	0	3.6
SSE	0.5	2.4	0.1	0	0	0	0	3.0
S	0.5	3.6	0.3	0.1	0	0	0	4.5
SSW	0	2.9	0.4	0.5	0	0	0	3.8
SW	0.8	5.3	0.7	0.3	0	0	0	7.0
WSW	0.5	7.2	6.3	2.1	0	0	0	16.2
W	0.7	6.7	6.7	0.9	0	0	0	15.0
WNW	0.3	4.5	2.9	0.8	0	0	0	8.4
NW	0.7	2.5	0.8	0	0	0	0	3.8
NNW	0.3	2.5	0.8	0.3	0	0	0	3.9
Total	6.8	59.8	26.5	6.7	0.1	0	0	100

2.3-6

Table 2.3-5c. "C" stability annual wind rose percentage frequencies.

Direction wind blows from	Windspeed, m/s							Total	Av speed
	0.2-1.6	1.6-3	3-5	5-7	7-11	11-16	>16		
N	0.1	1.4	0.7	1.0	0.3	0	0	3.4	4.2
NNE	0.4	1.6	1.0	0.3	0	0	0	3.3	2.9
NE	0.8	2.0	1.6	0.3	0.1	0	0	4.8	2.9
ENE	0.2	2.9	1.0	0.4	0	0	0	4.5	2.9
E	0.5	3.4	1.0	0.5	0.1	0	0	5.6	2.8
ESE	0.1	2.3	0.2	0.2	0	0	0	2.7	2.6
SE	0.3	2.5	0.3	0.2	0	0	0	3.3	2.5
SSE	0.3	1.8	0.5	0.1	0	0	0	2.7	2.5
S	0.4	1.9	0.7	0.2	0	0	0	3.1	2.7
SSW	0.2	2.7	0.3	0.1	0	0	0	3.3	2.3
SW	0.4	3.9	3.5	1.6	0.7	0	0	10.1	3.7
WSW	0.2	7.1	9.7	8.0	1.0	0	0	25.9	4.2
W	0.3	4.5	6.0	3.3	0.3	0	0	14.5	3.8
WNW	0.2	3.5	2.0	1.0	0.2	0	0	6.9	3.3
NW	0.1	2.0	0.9	0.2	0.1	0	0	3.3	3.0
NNW	0.2	1.6	0.4	0.3	0.1	0	0	2.5	3.2
Total	4.8	45.0	29.7	17.6	2.9	0	0	100	3.5

Table 2.3-5d "D" stability annual wind rose percentage frequencies.

Direction wind blows from	Windspeed, m/s							Total	Av speed
	0.2-1.6	1.6-3	3-5	5-7	7-11	11-16	>16		
N	0.1	0.9	0.4	1.0	0.7	0	0	3.0	5.2
NNE	0.3	0.6	0.7	0.5	0.2	0	0	2.2	4.1
NE	0.1	0.9	0.7	1.0	0.1	0.1	0	2.9	4.4
ENE	0.3	1.5	1.0	0.7	0.1	0	0	3.5	3.5
E	0.2	1.9	1.2	0.8	0	0	0	4.1	3.3
ESE	0.2	2.0	0.3	0.4	0	0	0	2.8	2.9
SE	0.1	1.4	0.7	0.1	0	0	0	2.3	2.8
SSE	0	0.9	0.4	0	0	0	0	1.3	2.7
S	0.1	1.3	0.4	0.3	0.1	0	0	2.2	3.3
SSW	0.2	2.1	1.2	0.6	0.4	0	0	4.5	3.7
SW	0.4	5.2	6.3	6.6	2.8	0	0	21.3	4.6
WSW	0.1	4.8	8.0	9.8	3.2	0.1	0	25.8	5.0
W	0.2	2.4	3.8	7.0	2.5	0	0	15.9	5.2
WNW	0.1	1.3	1.5	1.9	0.4	0	0	5.2	4.5
NW	0.1	0.8	0.4	0.2	0	0	0	1.6	2.9
NNW	0	0.6	0.1	0.3	0.4	0.1	0	1.5	5.6
Total	2.3	28.6	27.2	31.0	10.7	0.3	0	100	4.5

Table 2.3-5e "E" stability annual wind rose percentage frequencies.

Direction wind blows from	Windspeed, m/s							Total	Av speed
	0.2-1.6	1.6-3	3-5	5-7	7-11	11-16	>16		
N	0.1	0.3	0.4	0.4	0.4	0	0	1.5	5.5
NNE	0	0.5	0.5	0.9	0.2	0.5	0.1	2.7	6.9
NE	0.1	0.5	0.7	1.1	0.8	0.4	0	3.6	6.3
ENE	0.1	1.0	1.1	2.9	0.4	0.1	0	5.5	5.1
E	0.1	1.7	0.9	1.8	0.3	0	0	4.8	4.3
ESE	0.1	1.6	1.2	0.3	0.1	0	0	3.2	3.4
SE	0	1.1	0.7	0.2	0	0	0	2.0	3.1
SSE	0	1.2	0.2	0.2	0.2	0	0	1.8	3.5
S	0.1	1.0	0.7	0.2	0.2	0.1	0	2.3	4.0
SSW	0	2.0	2.3	1.8	0.8	0.2	0	7.0	4.8
SW	0.2	3.9	6.1	9.0	3.8	0.6	0	23.6	5.4
WSW	0.1	2.6	5.0	6.5	2.4	0.2	0	16.8	5.2
W	0.1	2.6	4.2	7.2	4.1	0.1	0	18.3	5.5
WNW	0	0.6	0.8	1.2	0.6	0.1	0	3.3	5.5
NW	0.1	0.4	0.4	0.6	0.1	0	0	1.5	4.6
NNW	0	0.3	0.1	0.5	0.7	0.3	0.2	2.1	8.5
Total	1.2	21.0	25.3	34.6	15.1	2.6	0.3	100	5.2

Table 2.3-5f "F" stability annual wind rose percentage frequencies.

Direction wind blows from	Windspeed, m/s							Total	Av speed
	0.2-1.6	1.6-3	3-5	5-7	7-11	11-16	>16		
N	0.4	0.9	0.1	0.1	0	0	0	1.6	2.1
NNE	0.1	0.4	0.9	0.7	0.4	4.2	1.1	7.8	11.3
NE	0.3	0.1	0.8	3.0	0.4	1.5	0.9	7.0	8.7
ENE	0.1	1.2	1.3	4.6	1.5	0.3	0	9.0	5.8
E	0.4	2.2	1.8	5.7	3.4	0	0	13.6	5.5
ESE	0.9	1.7	2.1	2.2	0	0	0	7.0	3.6
SE	0.3	2.0	1.8	1.2	0	0	0	5.3	3.5
SSE	0.3	4.5	2.0	0.3	0.5	0	0	7.5	3.2
S	0	1.3	0.7	0.1	0	0	0	2.1	2.9
SSW	0	0.7	2.8	1.2	0.1	0	0	4.7	4.3
SW	0.1	2.6	4.4	4.0	3.2	0.3	0	14.5	5.4
WSW	0.1	1.5	1.2	2.2	0.8	0.8	0	6.6	5.9
W	0.5	1.3	1.7	2.1	3.4	0	0	9.1	5.8
WNW	0	0	0.5	0.1	0.4	0	0	1.1	5.5
NW	0	0.4	0	0	0	0	0	0.4	2.2
NNW	0.4	0.4	0.1	0	1.2	0.3	0.5	2.9	8.7
Total	4.0	12.2	22.3	27.5	15.3	7.3	2.5	100	5.8

Table 2.3-6a "A" stability wind rose percentage frequencies for dry season, May to October.

Direction wind blows from	Windspeed, m/s							Total	Av speed
	0.2-1.6	1.6-3	3-5	5-7	7-11	11-16	>16		
N	0.9	2.5	1.9	0	0	0	0	5.3	2.5
NNE	0.4	1.5	1.3	0	0	0	0	3.2	2.7
NE	0.9	2.5	1.5	0	0	0	0	4.9	2.5
ENE	0.4	1.9	0.2	0	0	0	0	2.5	2.1
E	1.1	2.3	0.4	0	0	0	0	3.8	2.0
ESE	0.4	1.3	0.0	0.2	0	0	0	1.9	2.3
SE	0.6	1.1	0.2	0	0	0	0	1.9	2.0
SSE	0.2	1.3	0	0	0	0	0	1.5	2.1
S	2.3	4.9	0.2	0	0	0	0	7.4	1.9
SSW	0.6	3.8	0.4	0	0	0	0	4.8	2.2
SW	1.1	6.6	1.5	0.2	0	0	0	9.4	2.4
WSW	0.2	12.5	4.7	0.4	0	0	0	17.8	2.7
W	1.3	11.2	3.2	0.6	0	0	0	16.3	2.5
WNW	0.9	4.7	3.4	0.4	0	0	0	9.4	2.8
NW	0.9	4.0	0.9	1.1	0	0	0	6.9	2.8
NNW	0.2	2.5	0.4	0.2	0	0	0	3.3	2.5
Total	12.4	64.6	20.2	3.1	0	0	0	100	2.5

Table 2.3-6b "B" stability wind rose percentage frequencies for dry season, May to October.

Direction wind blows from	Windspeed, m/s							Total	Av speed
	0.2-1.6	1.6-3	3-5	5-7	7-11	11-16	>16		
N	1.5	2.1	0.3	0.3	0	0	0	4.2	2.1
NNE	0.9	0.3	1.8	0.6	0	0	0	3.6	3.3
NE	0.6	1.8	1.8	0.3	0	0	0	4.5	2.9
ENE	0.3	1.2	0.6	0	0	0	0	2.1	2.5
E	0.6	1.5	0.6	0	0	0	0	2.7	2.3
ESE	0.3	1.8	0.6	0	0	0	0	2.7	2.4
SE	0.6	0.9	0	0	0	0	0	1.5	1.7
SSE	0.9	0.9	0	0	0	0	0	1.8	1.6
S	0.6	2.4	0	0	0	0	0	3.0	2.0
SSW	0	2.7	0.6	0.9	0	0	0	4.2	3.2
SW	1.5	6.5	1.5	0.3	0	0	0	9.8	2.4
WSW	0.9	11.6	10.1	3.6	0	0	0	26.1	3.3
W	1.5	9.2	9.2	1.5	0	0	0	21.4	3.1
WNW	0.3	3.3	2.4	1.2	0	0	0	7.1	3.3
NW	1.2	1.5	0	0	0	0	0	2.7	1.7
NNW	0.3	1.5	0.6	0.6	0	0	0	3.0	3.1
Total	11.9	49.0	30.0	9.2	0	0	0	100	2.9

Table 2.3-6c "C" stability wind rose percentage frequencies for dry season, May to October.

Direction wind blows from	Windspeed, m/s							Total	Av speed
	0.2-1.6	1.6-3	3-5	5-7	7-11	11-16	>16		
N	0.1	1.5	1.0	1.3	0.4	0	0	4.3	4.3
NNE	0.3	0.9	1.0	0.1	0	0	0	2.3	2.9
NE	0.8	1.4	1.2	0	0	0	0	3.3	2.5
ENE	0.1	1.4	0.1	0	0	0	0	1.7	2.2
E	0.4	0.9	0.3	0	0	0	0	1.5	2.3
ESE	0.1	1.0	0	0	0	0	0	1.2	1.9
SE	0.4	0.8	0	0	0	0	0	1.2	1.8
SSE	0.4	1.5	0.1	0	0	0	0	2.0	2.2
S	0.8	2.2	0.1	0	0	0	0	3.1	1.9
SSW	0.4	2.3	0.3	0.1	0	0	0	3.1	2.3
SW	0.5	4.3	4.9	2.4	1.3	0	0	13.4	4.1
WSW	0.4	11.2	13.9	12.4	1.2	0	0	39.0	4.1
W	0.6	4.7	6.8	3.7	0.4	0	0	16.2	3.8
WNW	0.4	2.4	1.7	0.4	0	0	0	4.9	3.0
NW	0.1	1.0	0.1	0.1	0	0	0	1.4	2.4
NNW	0.3	0.6	0	0.5	0.1	0	0	1.5	3.6
Total	6.0	38.3	31.4	21.1	3.3	0	0	100	3.6

Table 2.3-6d "D" stability wind rose percentage frequencies for dry season, May to October.

Direction wind blows from	Windspeed, m/s							Total	Av speed
	0.2-1.6	1.6-3	3-5	5-7	7-11	11-16	>16		
N	0.1	0.6	0.3	1.0	0.9	0	0	2.8	5.9
NNE	0.2	0.1	0.4	0.3	0.1	0	0	1.0	4.6
NE	0.1	0.5	0.2	0.4	0	0	0	1.2	3.6
ENE	0.2	0.8	0.3	0.2	0	0	0	1.5	3.2
E	0.4	0.8	0.3	0	0	0	0	1.4	2.4
ESE	0.3	0.9	0	0.1	0	0	0	1.2	2.4
SE	0.1	0.4	0.1	0	0	0	0	0.5	2.7
SSE	0.1	0.6	0.1	0	0	0	0	0.7	2.6
S	0.2	1.0	0.1	0	0	0	0	1.3	2.1
SSW	0.2	2.6	1.2	0.7	0.5	0	0	5.1	3.8
SW	0.6	6.6	8.6	9.0	4.2	0	0	29.0	4.7
WSW	0.1	6.3	10.4	13.6	3.0	0.1	0	33.3	4.8
W	0.3	2.3	3.7	8.1	2.5	0	0	16.9	5.3
WNW	0.1	0.8	0.3	0.8	0.1	0	0	2.2	3.9
NW	0.1	0.4	0	0	0.1	0	0	0.5	3.8
NNW	0	0.3	0	0.4	0.6	0.2	0	1.5	7.4
Total	2.9	24.9	25.8	34.4	11.9	0.3	0	100	4.7

Table 2.3-6e "E" stability wind rose percentage frequencies for dry season, May to October.

Direction wind blows from	Windspeed, m/s						Total	Av speed
	0.2-1.6	1.6-3	3-5	5-7	7-11	11-16	>16	
N	0.1	0.2	0.4	0.1	0.4	0	0	1.1
NNE	0	0.1	0.3	0.2	0	0	0	0.6
NE	0.1	0.1	0.3	0	0	0	0	0.5
ENE	0	0.3	0.1	0.2	0	0	0	0.5
E	0.2	0.9	0	0	0	0	0	1.0
ESE	0.2	0.9	0.3	0	0	0	0	1.3
SE	0	0.3	0.1	0	0	0	0	0.3
SSE	0	0.4	0	0	0	0	0	0.4
S	0.2	1.1	0.3	0	0	0	0	1.6
SSW	0.1	3.2	2.7	2.5	1.2	0.3	0	10.0
SW	0.5	7.4	9.6	13.8	6.4	1.4	0	38.9
WSW	0.2	4.2	5.7	6.4	1.7	0.1	0	18.2
W	0.1	3.9	4.8	7.6	4.7	0	0	21.0
WNW	0.1	0.2	0.4	0.2	0	0.2	0	1.0
NW	0.1	0.1	0.2	0	0.1	0.1	0	0.4
NNW	0	0.1	0.1	0.4	1.5	0.7	0.4	3.2
Total	1.6	23.2	25.0	31.2	15.9	2.7	0.4	100

Table 2.3-6f "F" stability wind rose percentage frequencies for dry season, May to October.

Direction wind blows from	Windspeed, m/s						Total	Av speed
	0.2-1.6	1.6-3	3-5	5-7	7-11	11-16	>16	
N	0	0.7	0	0	0	0	0	0.7
NNE	0.7	0	0.7	0.7	0	0	0	2.0
NE	0.7	0	0	2.7	0	0	0	3.3
ENE	0	1.3	0	0.7	0	0	0	2.0
E	1.3	5.3	1.3	0	0	0	0	8.0
ESE	2.0	4.0	0.0	0	0	0	0	6.0
SE	0	0.7	0.7	0	0	0	0	1.3
SSE	0	0	0	0	0	0	0	0
S	0	0.7	0.7	0	0	0	0	1.3
SSW	0	2.7	5.3	1.3	0.7	0	0	9.9
SW	0.7	6.6	6.6	8.6	6.0	0.7	0	29.1
WSW	0	2.7	2.0	0.7	0.7	0	0	6.0
W	2.0	1.3	4.0	4.6	6.0	0	0	17.9
WNW	0	0	1.3	0.7	0	0	0	2.0
NW	0	0.7	0	0	0	0	0	0.7
NNW	0	0	0	0	6.0	1.3	0.7	10.0
Total	7.3	26.5	22.5	15.9	19.2	2.0	0.7	100

Table 2.3-7a "A" stability wind rose percentage frequencies for wet season, November to April.

Direction wind blows from	Windspeed, m/s							Total	Av speed
	0,2-1,6	1,6-3	3-5	5-7	7-11	11-16	>16		
N	0,1	7,8	0,9	0,1	0	0	0	9,0	2,4
NNE	0,5	4,0	0,4	0,1	0	0	0	5,0	2,3
NE	0,3	6,1	0,8	0,3	0	0	0	7,4	2,5
ENE	0	4,0	0,5	0,1	0	0	0	4,6	2,5
E	0	6,3	1,6	0,4	0	0	0	8,3	2,7
ESE	0,3	2,2	0,1	0	0	0	0	2,6	2,2
SE	0,4	4,0	0,4	0	0	0	0	4,7	2,3
SSE	0,1	3,0	0,3	0	0	0	0	3,4	2,3
S	0,1	6,5	0,5	0	0	0	0	7,1	2,3
SSW	0,5	3,8	0,4	0,1	0	0	0	4,9	2,2
SW	0,5	6,6	1,2	0	0	0	0	8,3	2,4
WSW	0,5	4,9	1,6	0,1	0	0,1	0	7,2	2,7
W	0,1	6,3	1,8	0,1	0,1	0	0	8,6	2,6
WNW	0,5	5,8	0,9	0	0	0	0	7,2	2,3
NW	0	3,3	1,3	0	0,1	0	0	4,7	2,8
NNW	0,5	5,4	0,8	0,2	0	0	0	7,0	2,5
Total	4,6	79,7	13,6	1,7	0,2	0,1	0	100	2,5

Table 2.3-7b "B" stability wind rose percentage frequencies for wet season, November to April.

Direction wind blows from	Windspeed, m/s							Total	Av speed
	0,2-1,6	1,6-3	3-5	5-7	7-11	11-16	>16		
N	0,2	3,8	0,5	0,2	0	0	0	4,7	2,5
NNE	0	3,1	1,4	0,2	0	0	0	4,7	2,9
NE	0	5,9	1,2	0,5	0,2	0	0	7,8	2,9
ENE	0,5	4,0	1,2	0,5	0	0	0	6,1	2,8
E	0	6,8	3,3	0,5	0	0	0	10,6	2,9
ESE	0	4,3	1,4	0,2	0	0	0	5,9	2,7
SE	0,2	5,0	0	0	0	0	0	5,2	2,2
SSE	0,2	3,5	0,2	0	0	0	0	4,0	2,2
S	0,5	4,5	0,5	0,2	0	0	0	5,7	2,4
SSW	0	3,1	0,2	0,2	0	0	0	3,5	2,5
SW	0,2	4,3	0	0,2	0	0	0	4,7	2,3
WSW	0,2	3,8	3,3	0,9	0	0	0	8,3	3,2
W	0	4,7	4,7	0,5	0	0	0	9,9	3,2
WNW	0,2	5,4	3,3	0,5	0	0	0	9,4	3,0
NW	0	3,1	1,4	0	0	0	0	4,7	2,6
NNW	0,2	3,3	1,2	0	0	0	0	4,7	2,6
Total	2,8	68,4	23,8	4,7	0,2	0	0	100	2,7

Table 2.3-7c "C" stability wind rose percentage frequencies for wet season, November to April.

Direction wind blows from	Windspeed, m/s							Total	Av speed
	0.2-1.6	1.6-3	3-5	5-7	7-11	11-16	>16		
N	0.1	1.2	0.3	0.7	0.1	0	0	2.4	3.7
NNE	0.5	2.4	1.1	0.4	0	0	0	4.4	2.8
NE	0.8	2.7	2.1	0.5	0.3	0	0	6.4	3.2
ENE	0.3	4.5	1.9	0.8	0	0	0	7.5	3.0
E	0.7	6.0	1.9	1.1	0.3	0	0	9.9	3.1
ESE	0	3.6	0.4	0.4	0	0	0	4.4	2.7
SE	0.3	4.4	0.5	0.4	0	0	0	5.6	2.5
SSE	0.3	2.0	0.8	0.3	0	0	0	3.3	2.9
S	0	1.6	1.2	0.4	0	0	0	3.2	3.3
SSW	0	3.1	0.4	0.1	0	0	0	3.6	2.5
SW	0.3	3.3	2.0	0.8	0.1	0	0	6.5	3.2
WSW	0	2.7	5.2	3.3	0.9	0	0	12.2	4.3
W	0	4.3	5.2	2.9	0.3	0	0	12.7	3.8
WNW	0	4.7	2.4	1.6	0.4	0	0	9.1	3.6
NW	0.1	3.1	1.7	0.3	0.1	0	0	5.3	3.0
NNW	0.1	2.5	0.8	0	0	0	0	3.5	2.5
Total	3.5	52.1	27.9	14.0	2.5	0	0	100	3.3

Table 2.3-7d "D" stability wind rose percentage frequencies for wet season, November to April.

Direction wind blows from	Windspeed, m/s							Total	Av speed
	0.2-1.6	1.6-3	3-5	5-7	7-11	11-16	>16		
N	0.1	1.3	0.6	0.9	0.4	0	0	3.3	4.2
NNE	0.4	1.2	1.3	0.9	0.4	0	0	4.1	4.1
NE	0.1	1.7	1.4	1.9	0.2	0.1	0	5.5	4.2
ENE	0.3	2.6	2.2	1.4	0.1	0	0	6.6	3.5
E	0	3.7	2.7	2.0	0	0	0	8.4	3.6
ESE	0.1	3.6	0.7	0.9	0	0	0	5.2	3.1
SE	0.1	3.1	1.7	0.2	0	0	0	5.0	3.0
SSE	0	1.4	0.9	0	0	0	0	2.4	2.7
S	0	1.9	0.9	0.7	0.2	0	0	3.7	3.7
SSW	0.1	1.4	1.3	0.5	0.3	0	0	3.6	3.8
SW	0	2.9	2.7	3.0	0.5	0	0	9.1	4.2
WSW	0	2.4	4.2	3.8	3.4	0.1	0	14.0	5.4
W	0	2.7	4.0	5.2	2.4	0	0	14.3	5.1
WNW	0.1	1.9	3.4	3.7	0.9	0	0	10.0	4.7
NW	0.1	1.6	1.1	0.6	0	0	0	3.3	3.4
NNW	0.1	1.1	0.3	0.1	0.1	0	0	1.7	3.0
Total	1.4	34.4	29.3	25.8	9.0	0.2	0	100	4.2

Table 2.3-7e "E" stability wind rose percentage frequencies for wet season, November to April.

Direction wind blows from	Windspeed, m/s							Total	Av speed
	0,2-1,6	1,6-3	3-5	5-7	7-11	11-16	>16		
N	0.2	0.3	0.4	0.6	0.3	0	0	1.8	4.7
NNE	0	0.7	0.8	1.4	0.4	0.9	0.1	4.3	7.0
NE	0.1	0.7	1.1	1.9	1.4	0.7	0	6.0	6.5
ENE	0.1	1.6	1.8	5.1	0.8	0.1	0	9.5	5.1
E	0.1	2.4	1.6	3.1	0.6	0	0	7.8	4.5
ESE	0.1	2.2	1.8	0.5	0.1	0	0	4.7	3.3
SE	0	1.8	1.2	0.4	0	0	0	3.3	3.3
SSE	0	1.7	0.4	0.3	0.3	0	0	2.8	3.5
S	0	0.9	1.0	0.3	0.4	0.2	0	2.8	4.9
SSW	0	1.1	2.0	1.3	0.4	0	0	4.7	4.5
SW	0	1.2	3.4	5.2	1.8	0	0	11.6	5.3
WSW	0	1.4	4.4	6.6	3.1	0.3	0	15.8	5.7
W	0.1	1.5	3.7	6.9	3.7	0.3	0	16.1	5.9
WNW	0	0.9	1.1	2.1	1.1	0	0	5.2	5.4
NW	0	0.6	0.6	1.0	0.1	0	0	2.3	4.5
NNW	0	0.4	0.2	0.5	0.1	0	0	1.2	4.6
Total	0.7	19.3	25.5	37.2	14.4	2.6	0.1	100	5.2

Table 2.3-7f "F" stability wind rose percentage frequencies for wet season, November to April.

Direction wind blows from	Windspeed, m/s							Total	Av speed
	0,2-1,6	1,6-3	3-5	5-7	7-11	11-16	>16		
N	0.5	1.0	0.2	0.2	0	0	0	1.8	2.5
NNE	0	0.5	1.0	0.7	0.5	5.3	1.3	9.3	11.6
NE	0.2	0.2	1.0	3.1	0.5	1.8	1.2	7.9	9.2
ENE	0.2	1.2	1.7	5.6	1.8	0.3	0	10.7	5.8
E	0.2	1.5	2.0	7.1	4.3	0	0	15.0	6.0
ESE	0.7	1.2	2.6	2.8	0	0	0	7.2	4.1
SE	0.3	2.3	2.1	1.5	0	0	0	6.3	3.5
SSE	0.3	5.6	2.5	0.3	0.7	0	0	9.4	3.2
S	0	1.5	0.7	0.2	0	0	0	2.3	3.1
SSW	0	0.2	2.1	1.2	0	0	0	3.5	4.4
SW	0	1.7	3.8	2.8	2.5	0.2	0	10.7	5.6
WSW	0.2	1.2	1.0	2.6	0.8	1.0	0	6.7	6.3
W	0.2	1.3	1.2	1.5	2.8	0	0	6.9	6.0
WNW	0	0	0.3	0	0.5	0	0	0.8	7.0
NW	0	0.3	0	0	0	0	0	0.3	2.2
NNW	0.5	0.5	0.2	0	0	0	0	1.2	1.9
Total	3.1	19.9	22.2	29.4	14.3	8.6	2.5	100	6.0

Table 2.3-8 Summary of wind frequency tables by stability categories.^a

Dry season	Stability category					
	A	B	C	D	E	F
Most freq. dir.	WSW,W	WSW,W	WSW,W	SW,WSW	SW,WSW	SW
Frequency (%)	34	48	55	62	57	29
Speed (m/s)	3	3	4	5	5	5
Least freq. dir.	ESE,SE,SSE	SE,SSE	ESE,SE	SE,NW	SE	SSE
Frequency (%)	5	3	2	2	<1	0
Speed (m/s)	2	2	2	3-4	4	0
Highest av speed (m/s)	3	3	4	7	10	8
Direction	WNW,NW	WNW	N	NNW	NNW	NNW
Frequency (%)	7	7	4	2	3	10
Lowest av speed (m/s)	2	2	2	2	2	0
Direction	S	SSE	SE	S	E	SSE
Frequency (%)	7	2	1	1	1	0
Wet season	A	B	C	D	E	F
Most freq. dir.	N	E	WSW,W	WSW,W	WSW,W	E
Frequency (%)	9	11	25	28	32	15
Speed (m/s)	2	3	4	5	6	6
Least freq. dir.	ESE	SSW	N	NNW	NNW	NW
Frequency (%)	3	4	2	2	1	<1
Speed (m/s)	2	3	4	3	5	2
Highest av speed (m/s)	3	3	4	5	7	12
Direction	NW	WSW,W,WNW	WSW	WSW	NNE	NNE
Frequency (%)	5	28	12	14	4	9
Lowest av speed (m/s)	2	2	3	3	3	2
Direction	ESE	SE,SSE	SSW,NNW	SSE	ESE,SE	NNW
Frequency (%)	3	9	3	2	8	1

^aValues rounded to nearest whole number.

Data summarized by the Weather Bureau¹¹ indicate that heavy fog (visibility less than 0.4 km) occurs on some 27 d/yr. The monthly frequency is rather uniform at 2 or 3 d/mo, except for 4 d in January and 1 d in March. Radiation fogs are more prevalent in late fall and early winter. High fog (coastal fog that is lifted in passing over the western hills) is prevalent on many spring and summer mornings but rarely reaches the ground in the vicinity of the LLL site.

2.3.3 On-Site Meteorological Measurements Program

The Laboratory maintains a meteorological program for rapidly assessing critical meteorological parameters that may be used to estimate potential doses to individuals in the off-site environment in the event of an accidental release of hazardous materials into the atmosphere. The program includes meteorological instrumentation near the Laboratory's north perimeter where the measurements accurately represent the overall site meteorology and are not influenced by building structures. The instruments are attached to a 40-m telescoping tower; measurements of wind direction and speed and temperature are recorded at both 10 and 40 m.

The sensors consist of a lightweight Climet bivane for measuring horizontal and vertical wind direction fluctuations within accuracies of ± 50 mrad and ± 20 mrad respectively, and a cup anemometer for measuring the wind speed to an accuracy of 0.07 m/s above the 0.27 m/s threshold speed. In addition, a thermistor temperature sensor is placed in a wind-aspirated shield to prevent exposure of the sensor to direct solar radiation and to prevent air from stagnating about the thermistor. Ambient temperatures may be measured to within 0.2°C with this type of sensor. The associated electronics are located in a van 100 m from the tower. Both standard chart recorders and punched tape are used currently for recording and digitizing the collected data for computer processing.¹³

The weather station also includes an M33 radar system, by which upper-level winds can be measured by tracking a weather balloon. The balloon's trajectory, as determined by the radar, is used to compute the wind directions and speeds at various altitudes. These measurements are made only upon programmatic need.

2.3.4 Short Term (Accident) Diffusion Estimates

Arithmetic average and probability maps of ground level concentrations exceeding specific probability values were obtained from the continuous point source (CPS) code developed by K. Peterson, T. Crawford, and L. Lawson of LLL. The input data were the same 10-m tower winds mentioned earlier.

The generalized diffusion equation¹⁴ for surface concentration under the plume centerline is

$$\frac{x}{Q} = \frac{1}{\pi \sigma_y \sigma_z u} \exp - \left(\frac{h^2}{2\sigma_z^2} \right)$$

where

χ/Q = relative concentration, s/m^3

σ_y = cross-wind horizontal standard deviation, m

σ_z = vertical standard deviation, m

u = horizontal windspeed, m/s

h = plume height above ground, m (stack height)

An accidental release will likely have a duration of minutes to hours; sector averaging, therefore, is not warranted.

Wind observations at 10 m above the ground are adjusted to the 15.8-m height of the reactor stack using the power law relationship:

$$u_{\text{stack}} = u_{\text{obs}} \left(\frac{h_{\text{stack}}}{h_{\text{obs}}} \right)^p$$

where p is a variable exponent, depending on the atmospheric stability category. The variation in p is from 0.14 for A stability to 0.45 for F stability.

Estimates of χ/Q for various probabilities of a specific value being exceeded between 1 and 50% of the time are shown for annual, dry, and wet seasons by Figs. 2.3-1a to 2.3-1c. The assumption is made that the wind is blowing toward a specific site boundary.

Figures 2.3-2a through 2.3-7b present the relative diffusion (χ/Q) contour maps under annual average, dry season, and wet season conditions. For each condition separate maps show the arithmetic average concentration (50% probability), and the 5% probability that specific concentrations will be exceeded. Two maps are presented for each condition and probability; (a) a close-in isopleth map out to about 5 km, (b) a distant map extended to about 50 km.

The LLL site boundaries are oriented in the four cardinal directions. The northern boundary is 450 m from the reactor, while the east, south, and west boundaries are 1140, 1150, and 440 m distant, respectively.

All stability categories are included in the probability maps. Hence, for example, the annual relative diffusion value (χ/Q) of the arithmetic average of the observations is 10 ns/m^3 at about 1 km to the northeast of the reactor site; this value includes stability categories A through F weighted by their relative frequencies.

Fig. 2.3-1a Probability of dilution factor (χ/Q) being exceeded at east and west site boundaries during the year.

Fig. 2.3-1b Probability of dilution factor (χ/Q) being exceeded at east and west site boundaries during the dry season.

Fig. 2.3-1c Probability of dilution factor (χ/Q) being exceeded at east and west site boundaries during the wet season.

Fig. 2.3-2a Annual close-in ground-level plume-center relative diffusion estimates (arithmetic av) — $\mu\text{B}/\text{m}^3$.

The short term accident diffusion estimates arrived at in this chapter differ from the diffusion estimates used in Sec. 15.4, Design Basis Accident. That more conservative calculation neglects the effect of stack elevation.

Fig. 2.3-2b Annual distant ground-level plume-center relative diffusion estimates (arithmetic av) — $\mu\text{s}/\text{m}^3$.

2.3.5 Long Term (Routine) Diffusion Estimates

Diffusion estimates of routine stack emissions released during normal operation are obtained in a similar manner to short term (accident) estimates, except that sector averaging is allowed. This means that the winds during any particular hour are assumed to vary over a 0.393 rad sector width; the resultant effect is to reduce the

Fig. 2.3-3a Annual close-in ground-level plume-center relative diffusion estimates (exceeded 5% of the time) — $\mu\text{s}/\text{m}^3$.

relative diffusion (χ/Q) contours. The basic equation¹⁴ used to calculate sector-averaged relative diffusion estimates is

$$\frac{\chi}{Q} = \sqrt{\frac{2}{\pi}} \left(\frac{8\sigma_y}{x} \right) \left(\frac{\chi}{Q} \right)_1,$$

Fig. 2.3-3b Annual distant ground-level plume-center relative diffusion estimates (exceeded 5% of the time) — $\mu\text{s}/\text{m}^3$.

where $(\chi/Q)_1$ is the nonsector-averaged value used in the short term discussion and x is distance downwind. For the average stabilities typically observed in the Livermore Valley, the ratio of $\sigma_y/x \approx 0.05$; hence to a good approximation (within $\pm 20\%$ from 400 m to 100 km), the above equation may be written as

$$(\chi/Q)_{\text{sect.}} \approx \frac{1}{3} (\chi/Q)_1$$

Fig. 2.3-4a Dry season close-in ground-level plume-center relative diffusion estimates (arithmetic av) — $\mu\text{s}/\text{m}^3$.

Thus dilution factors for long-term (routine) releases may be estimated from the short-term (accident) contour plots (Figs. 2.3-2 to 2.3-7) by reducing the contour values by one-third.

The patterns of the relative concentration plots are consistent with the surface wind summaries presented earlier. Two exceptions are the annual and wet season arithmetic plots which at 20 to 50 km and beyond show higher concentrations extending

Fig. 2.3-4b Dry season distant ground-level plume-center relative diffusion estimates (arithmetic av) — $\mu\text{s}/\text{m}^3$.

toward the west and northwest. These patterns reflect topographic effects (Sec. 2.1.1) on the calculations. The hills to the north clockwise through south are considerably closer to LLL than the hills in the opposite directions. Beyond the eastern hills the terrain drops off sharply to about mean sea level.

To the west the terrain is relatively flat for about 20 km. As a result, the surface concentrations in the Central Valley would be much reduced, compared to concentrations

Fig. 2.3-5a Dry season close-in ground-level plume-center relative diffusion estimates (exceeded 5% of the time) — $\mu\text{S}/\text{m}^3$.

at the same distance toward the west or northwest. Hence, even though the frequency of winds toward the east is greater than toward the west, the surface concentrations at moderate distances will be less.

Although the western site boundary is the closest, the preponderance of winds toward the northeast during both the dry and wet seasons causes the relative concentrations to be higher at the northern and eastern site boundaries. As shown in Table 2.3-9, this

Fig. 2.3-5b Dry season distant ground-level plume-center relative diffusion estimates (exceeded 5% of the time) — $\mu\text{s}/\text{m}^3$.

is particularly obvious during the dry season when the relative lack of easterly winds causes the 5% probability x/Q at the western boundary to be 0. During the wet season, the x/Q 5% value at the western site boundary is $30 \mu\text{s}/\text{m}^3$, while at the more distant eastern boundary the value is $40 \mu\text{s}/\text{m}^3$. Compare Tables 2.3-9 and 2.3-10.

Fig. 2.3-6a Wet season close-in ground-level plume-center relative diffusion estimates (arithmetic av) — $\mu\text{S}/\text{m}^3$.

Values of χ/Q along the critical azimuth (CA)* at various distances to 50 km are presented in Table 2.3-10. The site boundary values of χ/Q along the critical azimuth for the arithmetic average and 5% probability level can be used to derive an equivalent stability category by solving for $\sigma_y\sigma_z$ and referring to σ vs distance graphs given by

*The critical azimuth is defined as that direction to the site boundary that shows the largest value of χ/Q .

Fig. 2.3-6b Wet season distant ground-level plume-center relative diffusion estimates (arithmetic av) — $\mu\text{S}/\text{m}^3$.

Slade.¹⁴ If this stability category were observed continuously during the dry or wet season or annually it would provide the same X/Q at the site boundary as the arithmetic average and 5% probability maps indicate. This comparison is given in Table 2.3-11. The body of the table shows the equivalent Pasquill stability category for a 5 m/s windspeed.

Fig. 2.3-7a Wet season close-in ground-level plume-center relative diffusion estimates (exceeded 5% of the time) — $\mu\text{s}/\text{m}^3$.

Fig. 2.3-7b Wet season distant ground-level plume-center relative diffusion estimates (exceeded 5% of the time) - $\mu\text{s}/\text{m}^3$.

Table 2.3-9 Relative dilution factors at site boundary for the critical azimuth and for the closest boundary. Units are $\mu\text{s}/\text{m}^3$.

	Arithmetic average			5% probability		
	Annual	Dry season	Wet season	Annual	Dry season	Wet season
Critical azimuth	NE	NE	E	NE	NE	E
	10	20	6	100	100	40
Closest site boundary	W	W	W	W	W	W
	3	1	4	7	0 ^a	30

^aBy definition this means less than 5% of winds in dry season blow to west.

Table 2.3-10 Values of χ/Q along the critical azimuth (CA) at the site boundary (SB) and other distances to 50 km. Units are $\mu\text{s}/\text{m}^3$.

		Distance (km)						
	CA	0.5	SB ^a	2	5	10	20	50
<u>Annual χ/Q</u>								
Arith. av	NE	20	10	5	2	0.5	0.03	0.01
5% probability	NE	100	100	30	10	3	0.2	0.07
<u>Dry Season χ/Q</u>								
Arith. av	NE	20	20	7	2	0.8	0.05	0.02
5% probability	NE	100	100	40	10	5	0.3	0.09
<u>Wet Season χ/Q</u>								
Arith. av	E	10	6	3	1	0.4	0.04	0.007
5% prcbability	E	90	40	20	6	2	0.7	0.06

^aCritical azimuth SB is north-eastern site boundary = 0.70 km, except for wet season where SB is eastern site boundary = 1.1 km.

Table 2.3-11 Equivalent Pasquill stability categories for 5 m/s windspeed.

Level	Annual	Dry season	Wet season
Arith. av	B-C	C	B-C
5%	E	E	D-E

2.4 Hydrology

2.4.1 Hydrologic Description

The reactor site is located on the high end of the Livermore Valley in the Central Coast Range. The valley is subjected to relatively heavy rains throughout the winter season at which time aquifers are replenished. During the summer months the valley is very dry and without rain. The Livermore Valley is a large bowl filled with water bearing gravels and sealed on all sides to underground escape.¹⁵ Water percolates the surface gravels easily and the gravel beds saturate rapidly during the rainy season. Some areas of the valley, however, are overlain with stiff clay, reducing water seepage to the underground. Free flow travel through the gravel beds can be as high as 9 m/d, the actual flow being dictated by pumping demands, evaporation, and replenishment.

The site hydrologic system consists of the fresh water piping systems and drainage systems. The Laboratory water supply is taken from the Hetch-Hetchy Aqueduct, a source of treated water for San Francisco. An alternate supply is a 12 in. water line tied into the Zone 7 water system which supplies the City of Livermore. Underground water is not a normal source for the reactor.

The drainage system consists of storm sewers and open ditches depending on the degree of development in the immediate area.

2.4.1.1 Hydrosphere - The main drainage streams for the Livermore Valley are Arroyo del Valle, Arroyo Las Positas, Arroyo Mocho, Arroyo Seco, Cottonwood Creek, Cayetano Creek, and Tassajara Creek as shown in Fig. 2.4-1. All these streams flow west except Cottonwood, Cayetano, and Tassajara Creeks which flow south. Cottonwood Creek, Cayetano Creek, and Arroyo Seco converge with Arroyo Las Positas which flows into Arroyo Mocho at the western end along with Tassajara Creek. Arroyo Las Positas and Arroyo Seco, which start east of the Laboratory site and flow westerly, have the strongest effect on the Laboratory surface hydrology.

(a) Arroyo Las Positas watershed is in the northeastern and eastern hills above the Valley floor. The only part of this basin that directly affects the Laboratory is in the hills directly to the east.

This basin east of the Laboratory covers about 13 km² and is 5.6 km long. Several smaller streams converge in the hills and on a flood plain east of the Laboratory. The channel disappears on the flood plain, then forms again along the eastern and northern boundary of the Laboratory.

Fig. 2.4-1 Drainage streams of the Livermore Valley.

(b) The hills southeast of the Laboratory form the basin for Arroyo Seco. This basin covers about 41 km^2 and is 19 km long. The channel here is well defined and cuts across Sandia Laboratory just to the south of the Laboratory. Arroyo Seco then cuts across the southwest corner of the Laboratory on its way to Arroyo Las Positas.

2.4.2 Floods

2.4.2.1 Flood History – Numerous floods have occurred in the western end of the Livermore Valley. One of the heaviest came in December 1955. Arroyo de la Laguna went over its banks causing agricultural damage. At the upper end of the valley heavy run off from Arroyo Del Valle cut into road beds beside the stream bed. Documentation by the Corps of Engineers fails to mention flooding along Arroyo Las Positas or Arroyo Seco. Table 2.4-1 lists recorded flood dates and the stream flow at Alameda Creek, which receives the flow from Arroyo de La Laguna.

Table 2.4-1 Flood dates and flow rates on Alameda Creek. Arroyo de la Laguna flows into Alameda Creek.

Date	Peak discharge m^3/s
November 1892	382
January 1895	419
March 1907	461
January 1909	354
March 1911	807
January 1914	289
February 1938	297
November 1950	428
January 1952	580
December 1955	657
April 1958	558

2.4.2.2 Flood Design Consideration – Since the series of floods in the 1950's stream channels have been widened to handle flood water at the west (lower) end of the valley. A reservoir with a capacity of 95 hm^3 of water was built in the Arroyo Del Valle approximately 6 km into the canyon. Of this amount 46 hm^3 is assigned to flood control.

Locally, storm waters are handled by storm sewers and open ditches. No local dams or dikes exist on or near the Laboratory site to prevent a general water runoff to the northwest where it enters Arroyo Las Positas. The laboratory site is high enough in the east end of the valley to negate general flooding concerns. The foundation of the reactor containment building and adjoining laboratory building range between 0.3 and 0.6 m above the surface level effectively precluding general flooding from sheet flow.

2.4.3 Probable Maximum Flood (PMF) on Streams and Rivers

The Lawrence Livermore Laboratory and hence the reactor site are not subject to flooding. The discussions of flooding and dam failures (Sec. 2.4.2 and 2.4.4) support the conclusion that the only possible mode of flooding at the reactor site is sheet flow.

2.4.3.1 Probable Maximum Precipitation (PMP) – Continuous precipitation records have been kept at Mt. Hamilton since 1880 and at Livermore since 1881 by the United States Weather Bureau. Rainfall ranged from 1.5 m at Mt. Hamilton in 1884 to 0.15 m at Livermore in 1923.⁶

- Average annual rainfall 0.366 m
- Maximum daily rainfall 0.089 m
- Maximum monthly rainfall 0.259 m
- Maximum yearly rainfall 0.572 m

The probable maximum precipitation (PMP) for the three major arroyos for orographic and restricted convergence storms is⁶:

- Arroyo Del Valle 0.610 m 72 h duration
- Arroyo Las Positas 0.478 m 12 h duration
- Arroyo Seco 0.470 m 12 h duration

2.4.3.2 Precipitation Losses – Typically, the antecedent precipitation will be relatively high because the PMP is during the wet season. For example, prior to the April 1958 storm, antecedent precipitation was 0.11 m for the month of March. Prior to the flood of December 1955, the antecedent precipitation was 33 m.

2.4.3.4 Probable Maximum Flood Flow – The flood water will be sheet flow of several inches depth flowing fairly fast from the south in a northerly direction along existing streets and channels. It will subside with the peak of the storm.

2.4.4 Potential Dam Failures (Seismically Induced)

The only dam in the Livermore Valley is the Del Valle Dam, an earth fill embankment rising 68 m above the stream bed of Arroyo Del Valle. Assume the following events:

- Del Valle Reservoir has 50 hm³ (typical summer load).
- Del Valle Dam destroyed.
- Arroyo de la Laguna blocked by landslide.

Under these circumstances, water entering the valley floor at 122 m above mean sea level (MSL) could collect over an area of approximately 65 km² (lower end at 98 m MSL) which is below the 122 m MSL contour line.

If other water sources are allowed, then the quantity of water dumping into the valley floor could climb to 140 m MSL before it began flowing north through the San Ramon Valley. The Laboratory is located between 150 and 180 m above MSL. No flooding would result either from the water onrush or from the maximum potential buildup.

2.4.5 Probable Maximum Surge and Sieche Flooding

The only large body of water capable of entering the Livermore Valley is the Del Valle Reservoir which is downhill from the Laboratory site and approximately 10 km to the southwest. The Laboratory site is high in the east end of the valley essentially between two arroyos. No surge flooding could result at the reactor site except the sheet flow from Arroyo Seco as described in Secs. 2.4.2 and 2.4.3.

2.4.6 Probable Maximum Tsunami Flooding

Tsunami type flooding is not a potentiality in the Livermore Valley.

2.4.7 Ice Flooding

There is no history of ice jam formation in the Livermore Valley.

2.4.8 Cooling Water Canals and Reservoirs

The LPTR uses neither cooling water canals nor reservoirs.

2.4.9 Channel Diversions

Not applicable. See Sec. 2.4.8.

2.4.10 Flooding Protection Requirements

The LPTR's location, at the upper end of an arid valley, precludes most kinds of flood hazard. The LPTR's foundation is high enough, 0.3 m, to thwart any conceivable sheet flow. See Sec. 2.4.2.2.

2.4.11 Low Water Considerations

Since the LPTR's ultimate heat sink is the atmosphere, low water poses no threat.

2.4.12 Environmental Acceptance of Effluents

The LPTR discharges no liquid effluents to the environment. Radioactive waste water is collected in tanks and disposed of by the radioactive waste facility.

2.4.13 Groundwater

2.4.13.1 Description and On-Site Use – The Reactor Complex utilizes low conductivity water (LCW) for use in experiment cooling systems, reactor primary water additions and general laboratory purposes. This constitutes a very small usage. A more significant water usage is the normal laboratory water system which supplies water to the reactor's cooling tower. During peak usage, water may be supplied to the cooling tower basin in larger quantities (e.g., $1.5 \text{ dm}^3/\text{s}$). This water is lost primarily by transpiration caused by the cooling tower fan. None of the water is supplied by local underground sources, however.

2.4.13.2 Sources – Although water consumed by the Laboratory is not taken from the Livermore Valley, an underground water table has supplied water to the site. The reactor is in the most eastern portion of the valley where the catchment area is underlain by only small amounts of ground water of inferior quality.

Cretaceous beds west of the Sunol fault prevent ground water from escaping underground to the west. Water enters the underground water table chiefly by percolation through gravels underlying arroyo stream channels and by seepage or percolation through Pliocene age gravel into the aquifers. Wells in the eastern Livermore Valley are typically about 30 m deep.

2.4.13.3 Accident Effects – The reactor primary water system is the only system having enough radioactive water to be of concern. Portions of the primary piping are buried under the reactor room floor and normally run at a pressure ranging between 0 and 0.2 MPa. Primary coolant losses would be emitted into soils consisting of¹⁶:

0-2 m	Very stiff damp brown sandy clay
2-4.6 m	Slight gravel brown sandy clay to sandy clay at 4.6 m
4.6-6.1 m	Stiff moist brown sandy clay
6.1-7.9 m	Very stiff brown sandy clay
7.9-10 m	Stiff brown sandy clay plus slight gravel

Water content ranges from 13-14% of dry weight at the surface to 12% at 10 m.

Gross water leakage from the primary coolant underground portion of piping would be significantly hindered from penetrating deeply because of the soil's compaction, makeup,

and water content, and because of the minimal pressure on the piping. The following calculations demonstrate clearly that leakage to underground soil is not a significant safety concern.

- Assume primary water emits from a failed pipe into a soil volume shaped like the upper half of an ellipsoid of revolution.
- Assume water table is 30 m below ground.
- Assume the volume of the half ellipsoid $V = \frac{1}{2} (4\pi a^2 b / 3)$ where a is maximum horizontal radius of wetted region and b is maximum vertical radius. Let b = 30 m and a = 15 m producing $V = 14\,000\text{ m}^3$.
- Assume soil will hold 13% water by weight or $250\text{ dm}^3/\text{m}^3$ (soil density = 1.92 Mg/m^3).
- Then approximately 3500 m^3 of water would be required to saturate the soil or displace existing water down to the water table.

Disregarding leaching effects and flow rates, it is evident that the ground could absorb almost one hundred times the total primary coolant before the fluid reached the water table. Small leakages (in excess of evaporative losses) could easily be detected and corrected before endangering water tables in the area. A far more conservative analysis, using a 10 m water table and taking into account ion exchange and radioactivity decay, was recently completed.¹⁷ This analysis again showed that no measurable quantity of radionuclides would reach the water table.

The secondary water system has significant quantities of water and has underground piping. It is not, however, contaminated with significant radioactivity. The principal concerns regarding the secondary system are the additives used to prevent corrosion and algae buildup. Sulphuric acid is added automatically to keep the pH in the range 6.8-7.1. Microbicides and antifoulants are added on a demand basis. Section 9.3.4.2 describes the water treatment system in more detail.

2.5 Geology and Seismology

Appendix A contains this Laboratory's report, UCRL-51592, on geology and seismology information.

The Category I, II, and III structures referred to are found in Secs. 3.2.1 and 3.2.2.

REFERENCES

1. General Plan, 1959-1990, City of Livermore, California (1964).
2. General Plan, Pleasanton Planning Area, City of Pleasanton, California (1968).
3. Safety Guide 3, U.S. Atomic Energy Commission, Washington, D.C. (1970).
4. H. B. Piper and F. A. Heddleson, Nuclear Safety 14 (6), 576 (1973).
5. Title 10, Code of Federal Regulations, Part 100 (1974).
6. Expected Meteorological Conditions for the Livermore Research Laboratory of the AEC, Scientific Services Division, U.S. Weather Bureau, Washington, D.C. (1952).
7. G. R. Rumney, Climatology and the World's Climate (MacMillan Co., New York, N.Y. 1968).
8. H. Gilliam, Weather of the San Francisco Bay Area (University of California Press, Berkeley, California, 1962).
9. T. B. Smith and V. A. Mirabella, Characteristics of California Tornadoes, Meteorology Research Inc., Altadena, California, Rept. MRI 72 FR-996 (1972).
10. T. V. Crawford, J. Concannon, B. Cremer, P. Heptner, G. Higgins, J. Long, and D. Montan, Air Pollution, Present and Future, Livermore-Amador Valley (Livermore Air Pollution Control Study Committee, Livermore, California, 1968).
11. Climatological Summary for Livermore (National Climatic Center, Asheville, North Carolina, date unknown).
12. J. W. Bolstad, G. E. Cummings, J. L. Eagan, and H. F. Finn, Safety Analysis Report for the Livermore Pool Type Reactor, Lawrence Livermore Laboratory, Rept. UCRL-50147 (1966).
13. D. H. Slade, Environmental Science Services Administration, U.S. Department of Commerce, Technical Note 2-ARL-1 (1965).
14. Meteorology and Atomic Energy, D. H. Slade, Ed. (U.S. Atomic Energy Commission Technical Information Center, Oak Ridge, Tennessee, 1963).
15. M. Smith, Ground Water in the Livermore Valley (1934).
16. R. L. McNell, Final Report, Soil Investigation, Proposed LPTR Laboratory and Reactor Building at Livermore, Woodward-Clyde and Associates, Oakland, Calif. (1956).
17. R. W. Walston, Lawrence Livermore Laboratory, Internal memorandum, November 28, 1973.

3.0 DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1 Conformance with AEC Design Criteria

The LPTR is a low-power research reactor designed to general criteria available during the 1950's. Each applicable criterion listed in Ref. 1 will be addressed in a general manner. Nonapplicable criteria will be listed, but without comment.

I Overall Requirements

1. Quality Standards and Records

Chapters 13.0 and 17.0 disclose the administration of vital records and procedures related to critical items.*

2. Design Bases for Protection Against Natural Phenomena

Discussions in Ch. 2.0 disclose that various forms of flooding, hurricanes, and tornadoes are not a reactor safety problem. The reactor was originally designed to earthquake standards of 2 m/s^2 (0.2 g) for this geographic location. New data indicate that the safe shutdown earthquake is now 5 m/s^2 horizontal. The reactor bridge crane and heat exchanger were found to require modification to withstand the stronger motion and not endanger the containment shell or primary coolant pressure boundary respectively. These modifications are being implemented. Loss of all services to the facility due to natural phenomena does not prevent a safe shutdown of the reactor.

3. Fire Protection

The basic reactor structure, containment building, and adjoining laboratory building have adequate fire prevention controls either in the form of fire alarms and extinguishers or in material control. Section 9.5 discusses fire protection.

4. Environmental and Missile Design Bases

Stored energy within the reactor primary coolant system is insufficient to produce missiles; thus, missile protection relates only to protection of critical reactor systems from exterior sources such as gas bottles or cooling tower fans and tornado-generated missiles. Refer to Secs. 3.5 and 5.2.1.7 for discussion.

*Critical items in general refer to critical components, systems, or structures.

5. Sharing of Structures, Systems, and Components

The control room is located within the adjoining laboratory building, but is separated from work spaces so as to be independent of them.

II Protection by Multiple Fission Product Barriers

10. Reactor Design

The LPTR reactor core is designed for nuclear safety and experimental versatility. Fuel movements and core loadings are very tightly administered by the use of procedures, checklists, and reactivity surveys (Sec. 4.3.4). The reactor, when operated within the design envelope, can be shut down safely without benefit of outside services as mentioned in Criterion 2.

11. Reactor Inherent Protection

The reactor has a negative temperature coefficient of reactivity whose magnitude depends on the level of poisoning. The reactor has a negative void coefficient for each core irradiation facility; except the center facility, the use of which is very tightly administered. Loss of coolant causes the reactor to shut down. Certain major accidents, such as flooding of a beam tube, can cause positive reactivity additions as discussed in Sec. 15.1. The reactor core size insures a stable flux profile (variation as a function of $\cos^2 x$) preventing nuclear power instabilities within the core.

12. Suppression of Reactor Power Oscillations (not applicable)

13. Instrumentation and Control

Nuclear instrumentation is provided to monitor the fission power over a range from minimal source level to well beyond normal full power. These instruments are sufficiently independent so as to prevent common mode failures, Sec. 7.2.1.11. Process variables are monitors for control and alarms. Radiation levels within and exterior to the containment structure are monitored, offering alarms and automatic safety functions.

14. Reactor Coolant Pressure Boundary

The primary coolant system maximum pressure occurs at the primary pump outlet and can reach 390 kPa. The minimum pressure rating of any component is 410 kPa. Sections 5.1 and 5.4 describe the RCPB system.

15. Reactor Coolant System Design

The reactor coolant system (Sec. 5.1) is provided with isolation valves and storage tank to accommodate routine as well as emergency actions.

16. Containment Design

The containment structure is designed to minimize the escape of radioactive isotopes in case of an accident while at the same time allowing routine entry and exit of men and equipment for normal and shutdown operations. The containment has a filter/scrubber system to capture particulate contaminants and certain of the radioactive gases. Section 6.2 discusses the containment system features. Chapter 15.0 discusses the consequences of an accident.

17. Electrical Power Systems

The Pacific Gas and Electric system normally furnishes power to the facility. Upon loss of normal power, an emergency generator picks up the load for safety-related equipment, nuclear instrumentation and controls. The reactor scrams upon any loss of power. Since the reactor requires no electrical power for safe shutdown, the backup power system serves to insure available power to various emergency-related equipment (scrubber/filter system, Decon system, etc.) and nuclear instrumentation, Ch. 8.0.

18. Inspection and Testing of Electrical Power Systems

The electrical power distribution system for the Laboratory supplies normal power to the reactor complex. The system is subject to surveillance by Laboratory technicians from elsewhere in the Laboratory. The emergency generator for the reactor is tested periodically in coordination with weekly reactor checks.

19. Control Room

Except in the case of loss of coolant, the reactor control room (Room 1325) is subject to background environmental radiation only. Section 15.2 demonstrates that radiation is anticipated to be below 5 Rem for the duration of a loss of coolant accident. Section 15.4.1.6 discusses radiation associated with a design basis accident (DBA). The DBA is the only situation where radiation levels in the control room would prevent lengthy access times.

III Protection and Reactivity Control Systems

20. Protection System Functions

The protection system will protect the fuel element cladding so long as trip points are set as shown in Sec. 16.2.2. Section 7.2.1 clearly shows that all key parameters related to protection of the fuel clad are monitored. Complete loss of power leads to shutdown and a safe configuration. Removal of decay heat by other than absorption in the pool is unnecessary.

21. Protection System Reliability and Testability

The reactor instrumentation is sufficiently reliable or is backed up such that parameters shown in Sec. 16.2.1 have a very low probability of being violated. The instrumentation is intended to reliably scram the reactor on failure or when intended as per Sec. 7.2.1. The components can be readily checked and tested during shutdown and to some extent at power; see Sec. 7.2.

22. Protection System Independence

The protection system is not completely independent. See Secs. 7.3 and 7.2.1.11.

23. Protection System Failure Modes

A failure mode analysis reveals that no single component failure can prevent a scram, Sec. 7.2.1.11.

24. Separation of Protection and Control Systems

The primary protective circuits (three-level safety chambers) are not used for control in any way. The linear level channel is used for control, although it may have scram and rundown circuits associated with its recorder movements.

25. Protection System Requirements for Reactivity Control Malfunctions

Accidental withdrawal of the control rods can occur in manual modes. Safety channels are set well below any limiting safety settings. Violation of the power level safety limit (Sec. 16.2) would not occur so long as safety system settings are preserved.

26. Reactivity Control System Redundancy and Capability

The reactivity control system consists of four boron carbide rods (control rods) and one stainless steel rod (regulating rod). This system is described in detail in Sec. 4.2.3. Section 16.3 gives reactivity limitations.

27. Combined Reactivity Control Systems Capability

Absorber rods (4 shims and 1 regulating rod) are the only normal mode of reactivity changes.

28. Reactivity Limits

Reactivity controlled by the rods is limited in magnitude and insertion rates are such that the core excess reactivity at any time is within that specified in Sec. 16.3.1 of the Technical Specifications. This assures that the core can be shut down under all conceivable conditions when the rods are inserted.

29. Protection Against Anticipated Operational Occurrences (not applicable)

IV Fluid Systems

30. Quality of Reactor Coolant Pressure Boundary

The piping system and reactor tank, Ch. 5.0 were built to codes applicable in 1956, Table 5.1-1.

31. Fracture Prevention of Reactor Coolant Pressure Boundary

Brittle fracture of reactor coolant pressure boundary is an extremely remote hazard because of the temperatures (52°C) and pressure (39.3 kPa) involved, Sec. 5.2.1.

32. Inspection of Reactor Coolant Boundary

Some portions of the primary coolant system are buried below ground and are inaccessible for inspection. The reactor tank and piping in the reactor equipment pit (which includes valving, heat exchangers, and pump) are readily accessible and subject to visual inspection.

33. Reactor Coolant Makeup

Demineralized water is available on the reactor top, reactor balcony, and in the reactor equipment pit. It can supply water in case of pipe breakage, to the extent of about $0.5 \text{ dm}^3/\text{s}$.

34. Residual Heat Removal

Heat losses to environmental surroundings constitute the only residual heat removal.

35. Emergency Core Cooling System (ECCS)

The LPTR Emergency core cooling system is designed to spray water down onto the core from nozzles located at the tank top. The system depends on laboratory supplied water and pressure. During a LOCA, fuel clad melting is not possible for the most conservative conditions.

36. Inspection of Emergency Core Cooling System

The ECCS is operationally tested regularly to verify operability.

37. Testing of Emergency Core Cooling System

The ECCS is functionally tested regularly to ascertain flow, leakage, and pressure status.

38. Containment Heat Removal (not applicable)

39. Inspection of Containment Heat Removal Systems (not applicable)

40. Testing of Containment Heat Removal System (not applicable)

41. Containment Atmosphere Clean-up

The containment atmosphere is cleaned of particulate activity via a filter system, Sec. 6.2.3. Argon-41 activity is the only hazardous gaseous activity regularly purged from the containment via the stack system, Sec. 11.3.

42. Inspection of Containment Atmosphere Clean-up Systems

The airborne particulate clean-up system is regularly checked for operability by regularly operating the system, noting air movements, and checking valve positions. See Sec. 6.2.3.

43. Testing of Containment Atmosphere Clean-up Systems

(See Criterion 42 above.)

44. Cooling Water

The primary cooling system mentioned in Criterion 30 is the only cooling water system to the reactor. The ultimate heat sink is described in Sec. 9.2.5. No alternate heat removal systems exist (except for heat losses to the environment). Backup power sources are not provided, and none are required for safe shutdown of the reactor.

45. Inspection of Cooling Water System

Criterion 32 covers this aspect.

46. Testing of Cooling Water System

The primary coolant system is open to the atmosphere; it requires no pressure testing, Sec. 5.3.1. The same holds for the secondary coolant system, Sec. 5.3.2.

V Reactor Containment

50. Containment Design Basis

The containment structure, Sec. 6.2, is a simple shell designed to confine any internal radioactivity that could be emitted from the reactor core or by an experiment. Openings to the structure are filtered, normally closed, connected to a closed system or are immediately sealable by inflatable gaskets and isolation valves. The structure is designed to withstand a small internal overpressure and no external overpressure. The structure is not designed to absorb significant mechanical energy and indeed the LPTR has no significant stored energy such as high-pressure or high-temperature systems.

51. Fracture Prevention of Containment Pressure Boundary (not applicable)

52. Capability for Containment Leakage Rate Testing

The LPTR uses a dynamic containment system as described in Sec. 6.2.3. No leakage testing is done.

53. Provisions for Containment Testing and Inspection

The operational components (doors, seals, filter/scrubber system, and isolation valves) are subject to operational checks where possible, and visual inspection on a regular basis. Reference Sec. 6.2.3.

54. Piping Systems Penetrating Containment

Service piping, conduits, and secondary coolant piping penetrate the containment boundary. None of these systems has isolation valves. Refer to Sec. 6.2.4.

55. Reactor Coolant Pressure Boundary Penetrating Containment (not applicable)

56. Primary Containment Isolation (not applicable)

57. Closed System Isolation Valves (not applicable)

VI Fuel and Reactivity Control

60. Control of Releases of Radioactive Materials to the Environment

Argon-41 gaseous activity is the only effluent released to the environment at the reactor. All other effluents are disposed of elsewhere in the laboratory, Ch. 11.0.

61. Fuel Storage and Handling and Radioactivity Control

New fuel elements are kept in the reactor tank either in the core or the fuel storage rack. Once in the core or rack, all criteria related to the reactor core apply. Spent fuel elements remain in the reactor tank until transferred to a spent fuel facility. Handling of radioactivity in the form of irradiated samples is done routinely.

62. Prevention of Criticality in Fuel Storage and Handling.

The configuration of the storage rack located within the reactor tank, Sec. 9.1, prevents criticality even when each position contains a new fuel element.

63. Monitoring Fuel and Waste Storage (Refer to Criterion 61)

64. Monitoring Radioactivity Releases

Gaseous and particulate airborne radioactivity are monitored by the duct and stack systems, Secs. 11.3 and 12.2, prior to release from the containment structure. Solid

and liquid radioactive wastes are retained within the facility until transferred to the Laboratory Processing Facilities. Refer to Sec. 9.4.3 and 11.2.

3.2 Classification of Structures, Components, and Systems

3.2.1 Seismic Classification

As there is no great need to maintain operation in the event of an earthquake, no Category II elements are designated. Only a design basis earthquake is considered and this is also the safe shutdown earthquake.

Table 3.2-1 lists those elements of the LPTR that are designated as Category I.

Table 3.2-1 Category I elements in the LPTR.

Class	Item	Remarks
Structures:	Reactor shield	Main support structure for reactor.
	Reactor core support	Supports grid plate.
	Heat exchanger support	Failure could cause loss of coolant.
	Control rod actuator support	Reactor can scram even in event of damage.
	Containment building	Safety related only to extent collapse would cause loss of containment.
	Containment air scrubber building	Safety related only to extent collapse would cause loss of containment.
Components:	Hold up tank	Failure would cause loss of coolant.
	Reactor pool tank	Failure would cause loss of coolant.
	Primary heat exchanger	Failure would cause loss of coolant.
	Containment building isolation valves	Failure would cause loss of containment.
Systems:	Containment air scrubber	Desirable in event fission products are released to containment building.
	Primary coolant piping	Failure would cause loss of coolant.

3.2.2 System Quality Group Classification

The quality group classification components of the reactor coolant system and emergency core cooling system are listed in Table 3.2-2. The piping diagram (Fig. 3.2-1) shows their arrangement.

Table 3.2-2 System quality group classification,^a

Component	Code classification
Primary heat exchanger	Group D
Hold-up tank	Group D
Reactor pool tank	Group D
Primary coolant piping	Group D
Pumps and valves	Group D

^aThe components with Group D code classification were designed, fabricated and installed when the reactor was erected in 1957. They met the requirements existing at that time.

Fig. 3.2-1 Primary and secondary coolant systems.

3.3 Wind and Tornado Design Criteria

3.3.1 Wind Criteria

The original design criterion² for the containment building was 960 Pa normal to the vertical surface. This is in good agreement with a current reference³ which gives, for a 50-year period of recurrence, a maximum wind speed of 36 m/s for the Livermore area.

The dynamic pressure "q" corresponding to this maximum wind speed is found from³:

$$q = 0.613 V^2 \text{ Pa}, \quad (1)$$

where V = wind speed, m/s. If we add 10% for gusts, $V = 39.3$ m/s and $q = 0.613 (39.3)^2 = 949$ Pa. This compares favorably with the 960 Pa used in the original design. For a 100-year recurrence, the maximum wind speed is 40 m/s. The maximum wind speed observed at the reactor site is about 27 m/s due to strong northerly wind gusts.

The pressure of 960 Pa was applied to the containment building only. Most of the category I structures are inside the containment and thus escape the wind loads.

The design wind force was determined from³:

$$F = C_D A q, \quad (2)$$

where C_D is the drag coefficient or shape factor for a smooth right circular cylinder and A is the projected area. In analysis the wind load was considered as a static force applied horizontally to the side of the shell at the centroid.

It is appropriate to mention the building that houses the containment air scrubber system. The building is a commercial prefabricated structure of sheet metal, Armco Series S, Type I. The building is designed to meet the Uniform Building Code requirements for this region, 718 Pa wind pressure. The wind pressure was computed as follows:

- 1) Weather Bureau records through 1951 of the fastest wind velocity at anemometer level were reduced to the building height by the one-seventh power formula.
- 2) The reduced velocities were multiplied by 1.3 to account for gusts. Impact pressure corresponding to gust velocities were then calculated.
- 3) Finally, to obtain the design pressure, a shape factor of 1.3 (the effect of the combined inward and outward pressures on exterior surfaces of ordinary rectangular buildings) was applied.⁴

Since the building met the above requirement, no additional analysis was performed.

3.3.2 Tornado Design Criteria

Although tornadoes are rare in California, a recent study⁵ has shown that they do occur at infrequent intervals. The following quotation comprises the conclusions of that report:

"A survey has been made as to the occurrence of tornadoes in California during the 22-y period 1950-1971 and in previous years when observational data are limited. During the 22-y period, an average of two tornadoes per year has been reported in the state, a value much less than the average annual tornado frequency for the tornado-belt states.

"Through a careful characterization of tornado reports during the 22-y period, it was concluded that the maximum wind speed, including both translation and rotation, of California tornadoes is estimated to be 90 m/s. The amount of pressure drop due to a tornado vortex inside a tornado of 90 m/s was estimated to be 10 kPa and the time rate of change of pressure caused by a fast-moving 90 m/s tornado vortex was computed to be up to 3 kPa/s.

"The probability of a tornado striking a portion of the state in which some possibility of a tornado exists was determined to be once in 292,000 y. For the Livermore area in particular, the probability as given above is considered to be on the conservative side based on terrain consideration."

In accordance with the above conclusions, the design criteria will be:

- a) Maximum wind speed, 90 m/s.
- b) Maximum pressure drop, 10 kPa.
- c) Maximum rate of pressure change, 3 kPa/s.

These design criteria apply to the two category I structures exposed to wind forces, the containment building and the containment air scrubber building.

The containment building pressure integrity is assumed to be ruined by tornado generated missiles, most likely by damage to the decontamination ducting outside the building. Therefore, missile impact loads were not considered in the later analyses (Sec. 3.8.2) on the containment shell.

The containment air scrubber building and its contents are assumed to be rendered inoperable by a tornado with 90 m/s wind speed. The loss of this system and the

possible accompanying breach of containment can be corrected by closing the system isolation valves. These valves are located just outside the containment building shell.

3.4 Water Level (Flood) Design

The Lawrence Livermore Laboratory and hence the reactor site are not subject to flooding (Sec. 2.4).

3.5 Missile Protection Criteria

The reactor and its associated piping and mechanical equipment, with a few exceptions, are located inside the containment building. The exceptions are the air conditioning systems (Fig. 3.5-1) and the containment air scrubber system which is located in an adjacent metal building (Fig. 3.5-2). The containment building was analyzed for its ability to maintain its integrity under impact from internal and external missiles.

The scrubber system is housed in a commercial prefabricated structure of sheet steel. The system is assumed to be rendered inoperable by wind-borne missiles from a tornado.

Credible internal missile sources are pressurized gas cylinders and the foam generator fan (largest piece of rotating machinery). The reactor is a pool type facility; it has no high-pressure or high-temperature sources to create high-velocity missiles. The maximum potential energy source in any liquid system is 830 kPa and 21°C in the laboratory low conductivity water supply. The maximum for a gas system is 700 kPa and 21°C in the compressed air supply. The largest piece of rotating machinery outside of the equipment pit is the foam generator fan driven by a 7.5 kW, 126 rad/s motor. The process machinery in the equipment pit is separated from the containment shell by 0.3 m of reinforced concrete and 9.1 m of air space. Here again, the largest source is a pump driven by a 15 kW, 183 rad/s electric motor. Missiles from the equipment pit constitute a negligible hazard due to their relatively small size and the presence of the 0.3 m shield.

External missile sources are tornado generated missiles and nearby rotating machinery. As can be seen from Fig. 3.5-3, there are few nearby structures or pieces of equipment. The nearest structure with rotating machinery is the cooling tower, 43 m from the containment building. Tornado-generated missiles can be expected from breakup of adjacent buildings, cooling towers, trees, or utility poles. Observed and analyzed debris in the April 2, 1957, Dallas tornado showed that none of the missiles

Fig. 3.5-1 Air conditioning, decontamination, and filter/scrubber systems.

exceeded 55.9 m/s for a wind speed of 76.9 m/s. Using this ratio $55.9/76.9 = 0.73$ on the design tornado of 90 m/s, the maximum missile speed is 65 m/s (Ref. 6).

The penetrating ability of missiles with velocities of 65 m/s is indicated in Fig. 3.5-4 as a function of missile size and density. Two different methods were used to calculate penetration.

a) Petry Formula⁷

$$T = K \frac{W}{A} \log_{10} \left(1 + \frac{v^2}{19,974} \right)$$

where

Fig. 3.5-2 Filter/scrubber building.

Fig. 3.5-3 Containment building environs.

T = shield thickness, m.

K = shield material constant, $7.65 \times 10^{-5} \text{ m}^3/\text{kg}$.

W = missile mass, kg.

A = missile minimum cross sectional area, m^2 .

V = missile velocity, 65 m/s.

The T determined by the above formula is the thickness that will safely stop a missile without penetration. Penetration actually occurs at a thickness $2/3$ of the " T " found in the formula.

Fig. 3.5-4 Thickness of steel penetrated as a function of missile size and density.

By substituting

$$W = AL\rho$$

where

L = missile length, m

ρ = missile density, kg/m^3

the formula reduces to

$$T = 4.13 \times 10^{-6} \rho L.$$

The two lines on Fig. 3.5-4 give the penetration for missiles with the densities of steel and of wood as a function of length.

b) Stanford Research Institute Formula⁸

Penetration occurs if

$$\frac{E}{D} \geq U (0.344 T^2 + 0.00806 HT)$$

where

T = shield thickness

H = shield width, inch.

E = missile kinetic energy, ft-lb.

D = missile diameter, inch.

U = ultimate tensile strength of shield material, 55 000 psi.

By substituting

$$E = \frac{1}{2} \frac{W}{g} V^2$$

$$W = AL\rho$$

$$A = \frac{\pi D^2}{4}$$

and letting $H = 8 D$, as recommended in Ref. 8, the formula reduces to:

$$LD \geq \frac{8.53}{\rho} (0.0215 + 0.0161 D).$$

Again using steel and wood as typical missiles the formula becomes:

$$(LD)_{\text{steel}} \geq 0.65 + 0.485 D$$

$$(LD)_{\text{wood}} \geq 7.9 + 5.92 D$$

where L and D are in inches. Converting to SI units,

$$(LD)_{\text{steel}} \geq 4.2 + 3.1 D$$

$$(LD)_{\text{wood}} \geq 51.0 + 38.2 D$$

where L and D are in centimeters. Penetration occurs when the product of LD equals or exceeds the right-hand side of the equation.

Note that the penetration by the two methods is not directly comparable since the SRI formula shows penetration proportional to the product of missile length only. As the SRI experiments used clamped shield plates with widths not over eight times the missile diameter, the application to the containment shell is questionable. A considerable variety of objects could penetrate according to either formula. Examples are steel plates, utility poles, or large planks.

A listing of typical missile sources appears in Table 3.5-1. The analyses of these missiles indicates that the only internal missile that could cause containment penetration is

Table 3.5-1 Typical missile sources.

Missile source	Remarks
Containment building liquid system	Stored energy is negligible (830 kPa and 21°C max).
Containment building gas system	Stored energy is negligible (700 kPa and 21°C max).
Containment building foam generator fan	Calculations indicate containment penetration.
Pressurized gas cylinder	Calculations indicate no containment penetration.
Rotating machinery in equipment pit	Separated from containment shell by 0.3 m reinforced concrete.
Cooling tower fan	Calculations indicate no containment penetration.
Tornado generated debris	Assumed to render containment building air scrubber system inoperative. Penetration of containment building shell depends on size, see Fig. 3.5-4 or SRI formula.

a foam generator fan blade. The fan blades are $760 \times 230 \times 2.3$ mm thick and turn at 126 rad/s. A 3.2-mm-thick steel shroud encloses the fan. If all the rotational kinetic energy of a fan blade were available for translation the blade would penetrate the shroud and containment shell. This is a very conservative assessment as the penetration could happen only if the fan blade struck end-on and did not crumple. Furthermore, the foam generator is used only as a last resort in event of a fire. It is tested on alternate years when the reactor is shut down. With this type of duty cycle the probability of failure at a critical time is remote.

The most vulnerable target for tornado-borne missiles is the air duct leading from the containment building to the decontaminating and air-conditioning machinery. This is a $0.5 \text{ m} \times 0.8 \text{ m}$ steel duct of 4.76 mm material that projects from the containment building about 3 m. An object, such as a trash bin, with a mass of 340 kg or more and traveling at 61 m/s would destroy the duct and break the containment.

The likelihood of a missile penetrating the concrete shield surrounding the reactor core is remote. The shield is 1.8 m thick at the core elevation and made of 21 MPa (3000 psi) reinforced concrete. It is also clad on the outside with 19 mm thick steel plate.

Using the Petry formula with,

$$K = 8.8 \times 10^{-4} \text{ m}^3/\text{kg} \text{ (for reinforced concrete)}$$

$$T = 1.83 \text{ m}$$

$$V = 65.3 \text{ m/s}$$

the formula requires that:

$$\rho L \geq 38,500 \text{ kg/m}^2$$

or, for a steel object

$$L \geq 4.9 \text{ m}$$

Therefore, the conclusion is that tornado generated missiles may penetrate the containment building but it is highly unlikely they could penetrate the reactor shield to damage the core.

3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

The effect of the blowdown jet, reactive forces, and pipe whip resulting from rupture of any LPTR piping is minimal.

There is no liquid system with a temperature higher than 40.5°C or pressure above 82 MPa. The compressed gas systems are at 69 MPa maximum and room temperature.

3.7 Seismic Design

This section describes the engineering methods used in conducting seismic analysis of the LPTR facility. All analyses conducted were based on a safe shutdown earthquake (SSE) for Category I structures. No discussions of operational basis earthquake (OBE) or other than Category I structures are included.

3.7.1 Seismic Input

This subsection discusses the input criteria for seismic analysis of the facility.

3.7.1.1 Design Response Spectra - The SSE used in the analysis of the LPTR has a peak horizontal acceleration of 5 m/s^2 (0.5 g) and a peak vertical acceleration of 3.3 m/s^2 at the base of the structure. It has a peak spectral acceleration of 21 m/s^2 for the 2% critical damping curve. The design response spectra for the SSE are shown in Fig. 3.7-1.

These spectra have been developed for the Lawrence Livermore Laboratory by John A. Blume and Associates, Engineers, and are presented in the report entitled Investigation

Fig. 3.7-1 Design response spectra for the SSE.

of Faulting at the Lawrence Livermore Laboratory.⁹ The work was done to develop design criteria for the new plutonium facility located 427 m from the LPTR site.

A discussion of earthquake duration, distance and depths between seismic disturbances and the site, and existing available earthquake records are included in Appendix A.

3.7.1.2 Design Response Spectra

Derivation — The design Response Spectra was used for all analyses. No synthetic or actual normalized earthquake records for time-history analysis were generated. This approach yields conservative response estimates.

3.7.1.3 Critical Damping Values — Damping values of 2% of critical damping were used in the analysis of the Category I structures. The same damping values were used for all materials and modes of vibration. The 2% value provides a conservative estimate of the actual damping in the structure when stresses approach the yield point.

3.7.1.4 Basis for Site Dependent Analysis — Site dependent analysis was not used to develop the shape of the design spectra. The design spectra are based on a standardized configuration approved by the AEC for nuclear power plants. The spectral shape is based on a report prepared for Division of Reactor Licensing entitled "Recommendations for Shape of Earthquake Response Spectra."¹⁰

A site dependent analysis of the LLL site was previously conducted.¹¹ The results of the study indicated that the spectra generated by a site response analysis is in reasonable agreement with the SSE spectra developed by Blume.

3.7.1.5 Soil-Supported Category I Structures — All Category I structures considered are located within the main containment building which is founded on soil. A soil profile at the site of a proposed plutonium facility 427 m from the LPTR is the only deep soil data available at the LLL site. The soil profile provided by Shannon and Wilson, Inc.¹², soil mechanics and foundation engineers, is shown in Fig. 3.7-2.

Fig. 3.7-2 Generalized soil properties at Livermore site.

3.7.1.6 Soil-Structure Interaction — The Category I structures are founded on massive reinforced concrete foundations, resting on stiff clay. An estimate of the displacement due to rocking of the reactor shield was conducted, as discussed in Sec. 3.7.2.4. These displacements were found to be within acceptable limits and therefore all seismic analyses were conducted on the basis of fixed base models.

3.7.2 Seismic System Analysis

This subsection describes the seismic analysis performed for Category I structures.

3.7.2.1 Seismic Analysis Methods — Seismic analyses were conducted on the Category I structures shown in Fig. 3.7-3, these include:

- Reactor shield.
- Containment building.
- Bridge crane.
- Heat exchanger support.
- Control rod actuator support.

The response spectra method, which uses the natural period, mode shapes, and damping ratios for the structure, was used to conduct all seismic analyses.

Fig. 3.7-3 LPTR reactor building, E-W section facing north.

Yield and ultimate stresses of the construction materials were used for comparison with calculated stresses. Displacement criteria were based on physical interference between adjacent components.

Dynamic loads for the reactor shield and containment building were obtained with the response spectrum method applied to the multidegree-of-freedom models. The equivalent static displacement for the crane, heat exchanger, and control rod actuator support, were obtained by treating these structures as single-degree-of-freedom systems and applying the response spectrum method. The type of analysis conducted on each Category I structure is summarized in Table 3.7.-1.

Table 3.7-2 lists the computer programs used to conduct the analysis of the structures.

The dynamic analysis by the response spectrum method is well documented in Ref. 13. In the dynamic analysis the root mean square procedure was used for combining the modal response of the reactor shield and containment building as shown below:

$$V_{\max} = \sqrt{V_1^2 + V_2^2 + \dots + V_n^2}$$

Table 3.7-1 Methods used for seismic analysis of Category I structures.

Category I structure	Type of analysis
Reactor shield	MA ^a
Containment building	MA
Crane	ESD ^b
Heat exchanger	ESD
Control rod actuator support	ESD

^aModal analysis with response spectra.

^bEquivalent static displacement.

Table 3.7-2 Computer programs used by Lawrence Livermore Laboratory for seismic analysis.

Title	Program capability
GHOSH	Dynamic elastic stress analysis of axisymmetric structures subjected to nonaxisymmetric loads. ^{13,14}
TABS	Elastic, static, and earthquake analysis of three-dimensional frame and shear wall buildings. ^{15,16}
FRAME 2	Elastic static analysis of two-dimensional frame structures. ¹⁷

where: $V_1, V_2, \dots, V_n$ = modal response quantities (i.e. displacements, accelerations, stresses)

$V_{\max}$ = maximum value of quantity.

A description of the equivalent static displacement method used for analysis of the crane, heat exchanger, and control rod actuator support follows.

The lateral stiffness of the structure was determined by calculating the displacement resulting from a unit lateral load:

$$K = \frac{P}{\Delta}$$

where:

K = lateral stiffness.

P = unit load.

Δ = lateral displacement due to unit load.

The fundamental period of vibration of the structure was then calculated by considering the structure as a single-degree-of-freedom system:

$$T = 2 \pi \sqrt{\frac{M}{K}}$$

where:

T = fundamental period of vibration.

M = mass of system.

The spectral acceleration was obtained from the spectral response curve, Fig. 3.7-1, for the calculated period.

The maximum lateral displacement was calculated from the spectral acceleration as:

$$\Delta_{\max} = \frac{M S_a}{K}$$

where:

$\Delta_{\max}$ = maximum lateral displacement.

S_a = spectral acceleration as a function of damping ratio and period (see Fig. 3.7-1).

Member forces were then determined by conducting a static analysis of the structure subjected to the applied vertical loads and the calculated lateral displacement.

More details of the seismic methods used can be found in Ref. 18.

The mathematical models used for the analysis are described below:

3.7.2.1.1 Reactor Shield — A detailed description of the reactor shield is in Sec. 3.8.1. The reactor shield was analyzed about its weak axis as an axisymmetric structure. The structure was modeled with a finite element mesh composed of solid elements for standard and heavy concrete and shell elements for the aluminum pool tank. The shield was considered rigidly connected to its foundation for the analysis. The model is shown in Fig. 3.7-4. Material properties used for the linear elastic analyses are presented in Table 3.7-3.

The model was subjected to dead loads, horizontal and vertical components of earthquake, and a temperature gradient through the thickness of the shield.

Ten modes of vibration were used to determine the response of the structure to the horizontal earthquake component and five modes were used for the vertical response.

Table 3.7-3 Material properties used for reactor shield analysis.

Material	Modulus of elasticity, GPa	Poisson's ratio	Unit weight, kN/m ³	Mass density, Mg/m ³	Coefficient of thermal expansion, $\mu\text{m}/\text{m}\cdot\text{K}$	Thermal conductivity, W/m $\cdot$ K	Specific heat, J/kg $\cdot$ K
Standard concrete	22.8	0.17	24	2.40	9.0	2.76	837
Heavy concrete	34.5	0.20	35	3.60	9.0	2.76	837
Aluminum	73.2	0.33	27	2.77	—	—	—

3.7.2.1.2 Containment Building —

A detailed description of the containment building is in Sec. 3.8.2. The containment building was modeled as an axisymmetric structure.

A finite element mesh consisting of shell elements was used to model the steel structure. The model used is shown in Fig. 3.7-5. Solid elements were connected to the shell at five locations to

Fig. 3.7-4 Computer model of LPTR shield and aluminum tank.

Fig. 3.7-5 Computer model of LPTR containment building.

simulate the effect of stiffeners in the actual structure. The mass density of the steel was increased to include the effect of insulation weight. The building was considered rigidly connected to its foundation. Shell thickness were varied to model the different thickness of the shell and roof. Lap joints were included by increasing the thickness in the region of the joint. The compression ring located at the roof connection was included as a thicker shell. This has the effect of locally stiffening the shell-roof connection.

A linear elastic analysis was conducted. The loading conditions included dead load, 14 kPa internal pressure, and horizontal and vertical components of earthquake.

Ten modes of vibration were used to determine both horizontal and vertical response to earthquake excitation.

3.7.2.1.3 Crane - A detailed description of the crane is in Sec. 3.8.4.5. The first phase of the crane analysis consisted of three-dimensional analyses of the support structure with the hook load located at various positions. The direction of earthquake input was varied to determine the worst combination of hook position and earthquake direction. For these preliminary studies the crane support was modeled as a one story frame with its mass lumped at various positions at the story level. The supporting truss was modeled as a beam with equivalent moment of inertia.

The computer code TABS (Refs. 16 and 17) was used to conduct these studies. The worst case situation determined was with the hook load located on a long span truss near the one quarter span point. The earthquake input direction was selected to be in the plane of the long span truss. This combination of loading and earthquake direction represents the worst situation from a structural view point. A second loading position with the crane at mid span directly over the pool tank was selected to represent the worst condition from a nuclear accident point of view. Results of these studies indicated that a long span frame with these two loading conditions would adequately represent the actual structural behavior.

Only masses that would be lifted to the top of the structure were considered to effect the dynamic response of the structure. Masses moved around the operating floor and lifted only a few centimeters were neglected. The maximum expected crane hook load was then determined by finding the crane capacity and determining actual loading conditions that occurred at the LPTR facility. The crane has a capacity of 133 kN, but only a maximum load of 31 kN would be raised to the top of the structure. Table 3.7-4 shows loads and frequency of occurrence based on past usage.

For this seismic analysis a maximum expected crane load of 31 kN was selected.

Table 3.7-4 Frequency of occurrence for different crane loads.

Loading condition	Frequency y^{-1}
Typical load a few centimeters off ground	52
Maximum load at maximum height	3

A detailed linear elastic analysis of the long span frame was then conducted for the dead load plus the 31 kN maximum expected live load. The crane support was modeled as the plane frame shown in Fig. 3.7-6. Properties of A-7 structural steel were used in the analysis: $E = 207 \text{ GPa}$, Yield = 228 MPa, and Ultimate = 414 MPa. Examination of

the member connection details suggested semirigid internal joints and pinned column connections at the base.

3.7.2.1.4 Heat Exchanger Support — A detailed description of the heat exchanger support is in Sec. 3.8.4.2. The support structure was modeled as a pin ended frame with rigid joints. The stiffening effect of the connecting piping and the dead load of the structure were neglected in the analysis. The three-dimensional structure was

Fig. 3.7-6 Plane model of the LPTR crane structure used in analyzing the response to the SSE under the two worst case loading conditions.

Fig. 3.7-8 Control rod actuator support structure.

Fig. 3.7-7 LPTR heat exchanger support structure and plane frame model used to analyze structure's response to the SSE.

treated as a plane frame loaded as shown in Fig. 3.7-7. The total heat exchanger load of 45 kN was distributed as shown on each of two frames.

3.7.2.1.5 Control Rod Actuator

Support — A detailed description of the control rod actuator support is in Sec. 3.8.4.1. The control rod actuator support structure was considered as a lumped mass cantilevered from the reactor shield. A conservative estimate of 8.9 kN was made for the equipment and housing of the structures. The weight of the cantilever beams was neglected.

The structure was modeled as a single degree of freedom system and evaluated by the spectral response method. It was assumed that the SSE spectra was not altered by the shield and could be used as a direct input for analysis of the control rod actuator support.

Figure 3.7-8 is a sketch of the actual structure. The response of the structure to horizontal and vertical earthquake components was computed.

3.7.2.2 Natural Frequencies and Response Loads – A summary of natural frequencies and modal response loads and displacements for each of the Category I structures is shown in Sec. 3.8. Results for the reactor shield are in Sec. 3.8.1, the control rod actuator support in Sec. 3.8.4.1, the heat exchanger support in Sec. 3.8.4.2, the crane in Sec. 3.8.4.5, and the containment building are in Sec. 3.8.2.

3.7.2.3 Procedure Used to Lump Masses – The finite element method was used for analysis of the reactor shield and the containment building. Masses were concentrated at each nodal point in the mathematical models to properly represent the inertia forces in the direction of the element degrees of freedom. These masses refer to both translational and rotational inertia of the elements.

The crane, heat exchanger support, and control rod actuator support structures were considered single-degree-of-freedom systems and their total mass lumped at the appropriate locations. Since these structures all have large mass concentrations this approach models the inertia properties very well.

3.7.2.4 Rocking and Translational Response Summary – Dynamic analysis of the reactor shield and containment building were conducted on fixed base models.

Including the underlying soil in the reactor shield model has the effect of reducing the rigid connection at the base to a more compliant connection. The effect of the soil on the structural response of the reactor shield would be to increase the fundamental period of vibration and increase the absolute displacement at the top of the structure.^{19,20}

The analysis used to evaluate the effects of rocking and translation was to first estimate maximum displacements based on the period and spectral acceleration of the fixed base model. The effect of the soil-structure interaction on the fundamental period of vibration and spectral acceleration were then examined.

3.7.2.4.1 Method of Analysis for Rocking and Translational Response of Reactor Shield – The structure was assumed to be rigid and the effect of the soil was modeled as a translational and torsional spring. The values of the spring constants were determined from results derived for elastic half-spaces.²¹

The equation used for the translational spring constant was:

$$K_x = 2(1 + \nu) G \beta_x \sqrt{BL}$$

where:

G = shear modulus of soil.

ν = Poisson's ratio of soil.

B = effective foundation width.

L = effective foundation length.

β_x = parameter determined as a function of L/B ratio.

The equation used for the torsional spring constant was

$$K_\phi = \frac{G}{1-\nu} \beta_\phi B L^2$$

where:

B = effective foundation width.

L = effective foundation length in plane of rocking.

β_ϕ = parameter determined as a function of L/B ratio.

Values of β_x and β_ϕ for various L/B ratios and shown in Fig. 3.7-9.

Soil properties used to determine the spring constant are $G = 240$ MPa, obtainable from the site soil properties, and an assumed Poisson's ratio of $\nu = 0.4$.

The effect of earthquake loading was modeled by placing a lateral load at the center of gravity of the shield. The magnitude of the lateral load was determined as follows:

$a = 5 \text{ m/s}^2$ = peak acceleration from SSE spectra for calculated period of fixed base model of $T = 0.02 \text{ s}$.

$$F = Ma = 0.5 Mg = 0.5 W$$

where

g = acceleration of gravity.

M = mass of shield.

W = weight of shield.

Fig. 3.7-9 Coefficients β_x and β_ϕ for rectangular footing.

A static analysis of the model shown in Fig. 3.7-10 results in expressions for the maximum rocking and maximum translational displacement of

$$\Delta_{\max \text{ rocking}} = \frac{(0.5 W)(H)(h + t)}{K_\phi}$$

$$\Delta_{\max \text{ translational}} = \frac{0.5 W}{K_x}$$

Fig. 3.7-10 Model of reactor shield for estimating rocking and translational displacement.

period of vibration of the reactor shield with rocking and translation were made by applying Dunkerley's formula^{19,20}

$$T = \sqrt{T_{FB}^2 + T_t^2 + T_R^2}$$

where

T = fundamental period of shield with soil-structure interaction.

T_{FB} = fundamental period of fixed base model.

T_t = fundamental period due to ground translation alone.

T_R = fundamental period due to ground rocking alone.

The periods of ground translating and rocking can be expressed as¹⁹

$$T_t = 2\pi \sqrt{\frac{WH}{gK_x}}$$

Table 3.7-5 Estimates of maximum lateral displacement of reactor shield due to rocking and translation.

Earthquake direction	Effective foundation dimensions for rocking ^a		Effective foundation dimensions for translation ^b		β_ϕ	K_ϕ (GJ/rad)	β_x	K_x GN/m	Δ_{max} rocking (mm)	Δ_{max} translation (mm)
	B(m)	L(m)	B(m)	L(m)						
North-south	9	6	14	9	0.45	61	1.0	8.8	1.12	2.2
East-west	6	9	9	14	0.55	112	1.0	8.8	0.63	0.2

^aActual concrete pad dimensions.

^b1.5 times actual concrete pad dimensions. These dimensions were increased due to the confinement provided by adjacent foundations of other structures within the containment building.

$$T_R = 2\pi \sqrt{\frac{W}{g} \cdot \frac{H}{82.8 K_x}}$$

where

$\frac{W}{g}$ = mass of reactor shield per unit length,

H = height of reactor shield.

The calculated period with ground interaction was $T = 0.045$ s. From examination of the SSE spectra no dynamic amplification is caused by this increase in fundamental period, and the assumption of peak ground acceleration acting on the reactor shield for the translational and rocking displacements is valid.

3.7.2.5 Methods used to Couple Soil with Seismic-System Structures – A finite element analysis of the coupled soil-structure system was not conducted since estimates of rocking displacements were within acceptable limits.

3.7.2.6 Development of Floor Response Spectra – Floor response spectra were not required.

3.7.2.7 Differential Seismic Movement of Interconnected Components – No Category I structures are interconnected between floors at the facility.

3.7.2.8 Effects of Variations on Floor Response Spectra – Floor response spectra were not calculated.

3.7.2.9 Use of Constant Vertical Load Factors – Vertical response of the reactor shield and the containment building were obtained by a dynamic analysis as described in Sec. 3.7.2.1. The response spectrum used for vertical excitation was 2/3 of the SSE spectrum. Resulting stresses were combined directly assuming a simultaneous occurrence of the vertical and horizontal earthquake motions.

Vertical analysis of the control rod actuator support structure was conducted with the response spectrum method. It was assumed that the reactor shield was sufficiently rigid in the vertical direction not to change the shape of the response spectrum.

Since the crane and heat exchanger support were very stiff in the vertical direction and were designed for vertical loading with a factor of safety of 2, they were not analyzed for the vertical seismic excitation.

3.7.2.10 Methods Used to Account for Torsional Effects – Torsional modes of vibration were included in the finite element analysis of the reactor shield and containment building. Torsional effects on these structures are negligible due to symmetry.

Torsion of the crane support structure was analyzed with the three-dimensional computer program TABS^{16,17} and found to be negligible.

Torsional vibrations were neglected for the heat exchanger and control rod actuator support structures since these structures are symmetrical about their torsional axis.

3.7.2.11 Comparison of Responses – Since the time-history method of analysis was not used, results could not be compared with the response spectra approach.

3.7.2.12 Methods for Seismic Analysis of Dams – This topic is not applicable to the LPTR facility.

3.7.2.13 Methods to Determine Category I Structure Overturning Moments – Dynamic stresses were computed directly for the reactor shield and containment building. Stresses at critical locations were examined. Since stresses were calculated directly overturning moments were not needed.

Overturning moments were calculated for the control rod actuator support, heat exchanger support, and the crane structure by applying the response spectrum method to a single-degree-of-freedom system.

3.7.2.14 Analysis Procedure for Damping – All Category I structures were assumed to have 2% damping in all materials and all modes of vibration.

3.7.3 Seismic Subsystem Analysis

A seismic analysis of the Category I piping was conducted. Each section of pipe between supports was modeled as a uniformly loaded beam. The corresponding acceleration for the fundamental mode was evaluated by the spectral response method assuming 2% damping. Resulting stresses were combined assuming a simultaneous occurrence of vertical and horizontal components of seismic motion. The stresses were compared to the allowable values listed in the Nuclear Power Piping Code.²² The buried portion of the Category I piping, the containment building foundation, the soil beneath it and the equipment pit were assumed to move as a unit. Actual displacements between any of these items were considered negligible.

3.7.4 Seismic Instrumentation Program

This topic is not appropriate to the LPTR facility.

3.7.5 Seismic Design Control

This topic is not appropriate to the LPTR facility.

3.8 Design of Category I Structures

3.8.1 Concrete Support Structure

This section provides a description, original design considerations, and seismic analysis of the concrete support structure.

The reactor shield is a massive reinforced concrete structure whose primary function is the biological shielding of the reactor core. The shield supports the reactor pool tank, the control rod actuators, and most of the experimental irradiation facilities. The reactor pool tank in turn supports the reactor core and thermal columns. See Figs. 3.8-1 through 3.8-5.

A pair of stepped openings at the base penetrate the reactor shield concrete from the east and west directions. They form the cavities for the two thermal columns. The openings are 2.1 m square and step down to a 1.7 m high by 1.4 m wide opening.

On the east side of the monolithic concrete shield is a cave (igloo) formed of reinforced concrete blocks. These blocks are keyed to one another and to the reactor shield. The cave's external dimensions are 3.35 m high by 4.57 m wide by 2.64 m deep with 1.22 m thick walls and roof. The cave blocks are further tied to the monolithic reactor shield by a pair of 0.24 m channels. These channels prevent relative motion between the cave and shield in the event of an earthquake.

The ends of both thermal columns are closed with 1-m-thick concrete doors.

The shield is constructed of high density concrete (3.6 Mg/m^3 , magnetite aggregate) to the 4.3 m elevation and conventional concrete above that.

A thermal shield of lead 1.8 m high by 63 mm thick is attached to the outside diameter of the reactor pool tank. Its horizontal centerline corresponds to that of the reactor core. This shield limits the thermal stress and temperature rise in the concrete due to gamma ray heating.

The reactor pool tank serves as the inner wall of the reactor shield. The tank shell carries the radial stresses due to the hydrostatic head in the tank. The assumption is that the reactor shield is subjected to only dead loads due to its weight and seismic loads.

Fig. 3.8-1 North-south sectional view through the centerline of the reactor looking west.

The reactor shield foundation is a 6.1 m X 9.1 m by 1.37-m-thick reinforced concrete pad. The foundation is keyed to accept the igloo cave blocks and has 90 No. 6 bars connecting the shield to the foundation. The foundation covers

Fig. 3.8-2 East-west sectional view through the centerline of the reactor looking north.

enough area to distribute the weight of the pool tank, its contents, the reactor shield and igloo blocks without exceeding the soil bearing pressure.

Table 3.8-1 lists data on the reactor shield design.

Fig. 3.8-3 Plan view of experimental irradiation facilities.

The reactor shield was analyzed about its weak axis as an axisymmetric structure. The structure was modeled with a finite element mesh composed of solid elements for standard and heavy concrete and shell elements for the aluminum pool tank. The shield was considered rigidly connected to its foundation for the analysis. The model is shown in Fig. 3.7-4. Material properties used for the linear elastic analyses are presented in Table 3.7-3.

The model was subjected to dead loads, horizontal and vertical components of earthquake, and a temperature gradient through the thickness of the shield. Seismic analysis methods used for the shield are discussed in Sec. 3.7.2.1.

Fig. 3.8-4 Reactor shield looking northeast.

3.8.1.1 Stress and Displacements Due to Earthquake – Ten modes of vibration were used to determine the response of the structure to the horizontal earthquake component and five modes were used for the vertical response. Table 3.8-2 shows

Fig. 3.8-5 Reactor shield looking south.

the first three periods of vibration for both horizontal and vertical components of earthquake. Results of the dead load and seismic analysis are plotted in Figs. 3.8-6 through 3.8-9. Maximum seismic displacements of 0.19 mm lateral and 0.127 mm axial were calculated at the top of the shield.

It was assumed that the steel carries all the tension load and the concrete carries none. The steel was then checked to see if its area was adequate to carry the loads.

Regions of large axial stress occur at the connection to the foundation and at the standard concrete to heavy concrete interface. (See Fig. 3.8-10.) There is an overlap of reinforcement in these areas. Evaluation of steel stresses throughout the shield

Table 3.8-1 Reactor shield design parameters.

Applicable code	Uniform Building Code, 1952
Minimum compressive strength:	
Heavy concrete (28 d)	17 MPa
Standard concrete (28 d)	21 MPa
Dead load	5.3 MN
Concrete density:	
Magnetite aggregate (to 4.3 m)	3.6 Mg/m ³
Conventional aggregate (above 4.3 m)	2.4 Mg/m ³
Reinforcing rod	ASTM A 15-50 intermediate grade, hard, and ASTM A 305-50
Lead thermal shield	64 mm thick, ASTM B29
Allowable foundation bearing pressure:	
Dead and live load at a minimum of 0.9 m below ground surface	144 MPa
Total load including wind and seismic	192 MPa
Seismic criteria:	
Original design	1 m/s ² (0.1 g)
Later criteria	See Secs. 3.7.1 and 3.7.2

Table 3.8-2 Periods of vibration of the reactor shield.

Mode	Period of vibration, ms
Horizontal	
1	20
2	9.1
3	4.8
Vertical	
1	7.3
2	3.8
3	3.3

indicates they are always below ultimate values. Some yielding of the vertical steel is expected to be caused by the design basis earthquake. Steel stresses throughout the reactor shield are shown in Fig. 3.8-11.

Maximum shear stresses of 344 kPa were computed in the concrete. (See Fig. 3.8-12.) The allowable shear stress is 414 kPa for concrete having a compressive strength of 21 mPa .

3.8.1.2 Stresses Due to Temperature

Gradient — To get an estimate of the

stresses caused by thermal gradients through the shield during operation, a steady state thermoelastic analysis of the shield was conducted with the same model used for the seismic analysis. The model was subjected to the boundary conditions shown in Fig. 3.8-13. This plot also shows the resulting temperature distribution.

Fig. 3.8-6 Deformed shape of LPTR shield and aluminum tank, using rms approximation, resulting from horizontal component of SSE. Displacement amplification, $\times 4000$. (Dots indicate unstressed position.)

Fig. 3.8-7 Deformed shape of LPTR shield and aluminum tank, using rms approximation, resulting from vertical component of SSE. Displacement amplification, $\times 4000$.

Figure 3.8-14 shows the displaced shape of the shield magnified 1000 times. A maximum lateral displacement of 0.15 mm and a maximum axial displacement of 0.5 mm were calculated. The tangential and axial stress contours due to the thermal gradient are shown in Figs. 3.8-15 and 3.8-16. These stresses were calculated assuming the concrete can carry tension and does not crack. High tangential and radial stresses occur at the concrete interface and along the tapered section.

Fig. 3.8-8 Lateral displacement at outer surface of LPTR shield, horizontal and vertical SSE.

Fig. 3.8-9 Vertical displacement at outer surface of LPTR shield, horizontal and vertical SSE.

These calculated stresses are believed to be an upper limit on operational stresses. Any local cracking of the concrete in this area will reduce stresses and allow free thermal expansion. Cracks are known to exist in this region of the shield.

3.8.1.3 Shield Tie to Foundation — The axial stress distribution at the reactor shield foundation due to dead load, vertical earthquake, and horizontal earthquake are shown in Fig. 3.8-17. The dead load and vertical earthquake stress distributions are axisymmetric while the distribution due to the horizontal component of earthquakes varies as the cosine of the angle between the earthquake direction and the position of interest.

Figure 3.8-18 shows the maximum compressive stress distribution at the foundation. This assumes that both earthquake components combine with the dead load to give the largest compressive values. A maximum compressive stress of 1.4 MPa was calculated. This is well within the crushing strength of the heavy concrete, 17.2 MPa at 28 days.

Fig. 3.8-10 Maximum normal stress at each elevation of LPTR shield due to combined dead load and horizontal and vertical components of SSE (rms approximation).

The earthquake stresses and dead load stresses were then combined in such a way as to produce the maximum tension stress distribution at the base. The structure is held down at the base by 90 No. 6 bars distributed over the foundation. These bars were evenly spaced, 22 at a 1.1 m radius and 68 at 2.4 m radius. To simplify the calculation of bar stresses the axial stress distribution was approximated as varying linearly from 138 kPa to 620 kPa as shown in Fig. 3.8-19.

The maximum bar stress can be obtained by considering an axisymmetric tension stress distribution. Results show a tension stress of 380 MPa for the inner circle of bars and 450 MPa for the outer bars. For hard grade deformed bars of billet steel the yield stress is 345 MPa and the ultimate stress is 552 MPa.

Bond stresses were then checked to see if the bars will pull out. The actual bond stress was calculated by²³

$$u = \frac{F}{\phi \Sigma_0 L}$$

where

F = tension force in steel.

Σ_0 = sum of perimeters of steel bars.

L = embedment length of steel.

ϕ = capacity reduction factor = 0.75 (assumes imperfect workmanship).

The allowable bond stress was computed from

$$u_{\text{allowable}} = 1.33 \left(\frac{9.5 \sqrt{f'_c}}{D} \right)$$

where

f'_c = compressive strength of concrete.

D = bar diameter = 19 mm.

Properties of A15-50 reinforcing steel

- $\sigma_{\text{yield}} = 350 \text{ MPa}$

- $\sigma_{\text{ultimate}} = 550 \text{ MPa}$

Fig. 3.8-11 Reactor shield reinforcement details and calculated steel stresses.

This results in a maximum bond stress of 6.2 MPa and an allowable of 6.4 MPa. Therefore, the bars will hold.

Since there is adequate bond, some yielding of the bars will take place if both earthquake components combine in this worst case situation.

The transfer of base shear was then examined. The maximum shear stress in the concrete at the base is 234 kPa from Fig. 3.8-12. The base shear force was then computed. It was assumed that this shear force is distributed equally over all 90 No. 6 bars. This results in a shear stress of 203 MPa in each bar. This value is slightly greater than the shear yield stress ($\tau_{\text{yield}} = 345 \text{ MPa}/\sqrt{3} = 200 \text{ MPa}$). Thus some yielding of the bars is expected. There are two other means of transferring the shear load at the base. These include mechanical interference due to the key and friction between the foundation and shield. Since these were not considered in the analysis the above results are conservative.

Fig. 3.8-12 Maximum shear stress at each elevation of LPTR shield due to combined dead load and horizontal and vertical components of SSE (rms approximation).

3.8.2 Steel Containment System

This section provides a description, original design considerations, and seismic analysis of the steel containment system.

3.8.2.1 Containment Building Shell — The reactor is housed in a controlled-leakage steel building designed to assure effective containment of radioactive effluents under all conditions. The building is a cylindrical shell of welded steel plates 6.3 mm thick (reinforced with stiffener angles) with a domed, self-supporting roof of 8 mm steel plates. The base of this structure is welded to large anchor bars buried in a massive reinforced-concrete ring wall. The building is 24 m in diameter, 16 m high, with a working floor area of nearly 418 m². Volume is about 6258 m³. Section and plan views of the building are shown in Figs. 3.8-20 and 3.8-21 and the structural elements of the shell are shown on Fig. 3.8-22.

Fig. 3.8-13 Computed temperature distribution inside reactor shield.

Fig. 3.8-14 Deformation of the LPTR shield by thermal stress. Displacement amplification, $\times 1000$.

Fig. 3.8-15 Tangential thermal stresses in the LPTR shield.

A stack located at the middle of the containment building roof is 2.4 m in diameter, 1.5 m high, fabricated from 6.3-mm-thick plate and has a domed roof. It is designed to withstand 14 kPa.

The containment building air lock is a 2.4-m-diam-by-3-m-long horizontal steel cylinder with gasketed doors at each end. The exterior of the air lock is insulated as is the containment building steel shell. Doors in the air lock have a clear opening of 1.14 $\times$ 2.1 m. The doors have 4.7 mm thick cover plates on both sides over angle stiffeners.

A reinforced concrete saddle rigidly connected to the shell foundation of the containment building supports the air lock. The saddle, located 2.6 m radially outward from

Fig. 3.8-16 Axial thermal stress in the LPTR shield.

Fig. 3.8-17 Axial stress distribution at base of LPTR shield due to dead load and SSE components.

the centerline of the shell, is 0.23 m thick and has 0.46 m wide spread footing. Length of the saddle in the direction transverse to the air lock axis is 2.4 m. Attaching the saddle to the shell foundation serves to prevent any differential settling between the air lock and the containment building.

Two escape doors provide emergency exit from the containment building. These doors are similar in construction to those of the air lock except that they are 0.76 m wide by 1.8 m high.

A truck door 3 X 4.2 m provides access to the containment building by large equipment. Its rubber gasket is inflatable to seal it tightly against pressure buildups within the

Fig. 3.8-18 Maximum compressive stress distribution at LPTR base, i.e., sum of horizontal, vertical, and dead loads.

Fig. 3.8-19 Maximum tension stress distribution at LPTR shield base, i.e., sum of horizontal plus vertical minus dead load.

building. This door and its frame are adequately reinforced by channels, angles and stiffener plates. As usually prescribed by vessel codes, the area cut out of the shell is compensated by reinforcement.

The containment shell is penetrated by three rectangular ducts on the west side, see Fig. 3.8-23. The openings form the supply and return ducting for the containment air conditioning and the return duct for the emergency decontamination system. Both systems share the same supply duct. All the ducting external to the containment building shell is of sufficient gauge to withstand 14 kPa internal pressure.

The return duct is rectangular in cross section .76 × 1.2 m and is fabricated from 6.3-mm-thick carbon steel plate. It has 4 × 4 × 6.3 mm stiffener angles continuously welded around the periphery of the duct on 0.51 m centers. The stiffener angle is welded into a complete rigid frame. This same construction applies to the 1.1 × 0.51 m supply duct. The decontamination duct is also rectangular in cross section 0.51 × 0.66 m and is fabricated from 4.2 mm carbon steel plate.

Fig. 3.8-20 Reactor containment building elevation.

Fig. 3.8-21 Plan view of reactor containment building.

Fig. 3.8-22 Structural features of the LPTR containment building.

3.8.2.2 Foundation — The reactor shield foundation consists of a reinforced concrete pad $6 \times 9 \times 1.4$ m. This pad is large enough to distribute the weight of the entire pool tank, its contents, and the reactor shield, such that the actual soil pressure is below the 144 kPa allowable foundation soil pressure. In the center of the containment building there is a 0.15 m deep depression to accommodate the pool tank. No anchor bolts are required for the pool tank since it is adequately stabilized by the shielding concrete.

Fig. 3.8-23 Containment building ducting.

The inlet and outlet for the pool tank run through this foundation. At the time of erection, grout was forced under the tank bottom thereby making a permanent level base for the pool tank. All joints between the floor slab and foundation pad have a copper water stop to prevent leakage into the subsoil.

The pool tank outlet line runs beneath the floor slab and enters the 13.6 m^3 hold up tank. The tank is 1.8 m in diameter and its centerline is 2.7 m below the top of the floor slab. The tank is supported by two reinforced concrete saddles 0.3 m wide with a spread footing $0.96 \times 2.1 \text{ m}$.

A 46 m^3 storage tank is also buried within the confines of the containment building foundation. This tank is 3 m in diameter and the centerline of the tank is 2.7 m below the top of the floor slab. Two concrete saddles support the tank. A $0.01 \times 0.30 \text{ m}$ bearing plate wrapped around the bottom of the tank for more than $\pi/3$ rad on each side of the vertical centerline strengthens the tank where it rests on the saddles. The concrete saddle, made of reinforced concrete, is 0.3 m wide and is integral with a spread footing $1.2 \times 3.4 \times 0.46 \text{ m}$. This foundation is designed to support the fully loaded tank as well as anchor it to eliminate the possibility of flotation.

3.8.2.3 Specifications, Codes and Materials – The containment building steel shell design was not completely covered by any single code. The original design calculations make reference to the following:

- The American Petroleum Institute Recommended Rules for the Design and Construction of Large Welded Low-Pressure Storage Tanks (Tentative Standard) API Standard 620, 1956. (API)
- The American Institute of Steel Construction Specification for the Design, Fabrication and Erection of Structural Steel for Buildings.
- The American Water Works Association Standard Specifications for Steel Tanks, AWWA D 100-55, 1956. (AWWA)
- The API-ASME Code for Unfired Pressure Vessels for Petroleum Liquids and Gases, 1951.
- The American Welding Society Standard Rules for Field Welding of Steel Storage Tanks, D 5.1-55, 1955. (AWS)
- The American Society for Testing Materials Book of Standards, Part One, Ferrous Metals, 1955. (ASTM)
- Uniform Building Code, 1952 Edition.

The foundation design calculations make reference to the following codes:

- Uniform Building Code, 1952 Edition.
- American Concrete Institute Manual of Concrete Practice.
- ACI Building Code Requirements for Reinforced Concrete, ACI 318-51 Materials.

All steel plates used in the shell and domed roof are fabricated from ASTM-A283 Grade C steel which has a specified minimum tensile strength of 380 MPa. All

structural steel members are fabricated from ASTM-A7 steel which has a specified minimum tensile strength of 400 MPa.

Concrete reinforcing bars are deformed bars of hard grade billet steel complying with A. S. T. M. Standard Specification A-15, with deformations in accordance with A. S. T. M. Specification No. A-305. Welded steel wire fabric complies with A. S. T. M. Standard Specification No. A-185.

3.8.2.4 Load Combinations – The analysis of the containment building takes into account a number of loading conditions. The various combinations considered are displayed in Table 3.8-3

Table 3.8-3 Load combinations considered in stress analysis of the containment building.

Load	Case number								
	Steel shell					Foundation			Slabs and walls
	1	2	3	4	5	6	7	8	9
Dead weight	X	X	X	X	X	X	X	X	X
Live Load									X
Wind	X	X				X			
Internal pressure		X			X	X			
External pressure			X				X		
Tornado				X				X	
Seismic					X				

3.8.2.5 Design and Analysis Procedure – The design and construction of the containment building was in accordance with the Uniform Building Code, 1955 Edition. According to the Uniform Building Code, the reactor site lies in seismic zone 3; hence, the building was designed as recommended for this zone in the Code. The containment building has since been reanalyzed by more advanced methods to determine its behavior during a design basis earthquake, see Sec. 3.7.2.1.

The containment building as a whole is designed for a horizontal wind pressure acting in any direction of 1 kPa. In calculating the wind load, allowance is made for the shape of individual elements by reducing the pressures by a shape factor.

The steel shell of the containment building is designed for an internal pressure of 14 kPa. When the building is sealed, this pressure may be developed by variations in barometric pressure and temperature. The pressure rise was calculated on the following basis: The outside ambient temperature was assumed to be 4°C when the building

is sealed. This temperature rises to 43°C during the next 12 h. The 39°C temperature rise is the maximum expected during any day at the site. Tests performed at the LPTR indicate the building air temperature range is about 75% of the ambient range under sealed conditions. The building air range is then 29°C. The resultant pressure increase would be 10.3 kPa. Assuming a concurrent barometric pressure drop of 3.4 kPa, the total internal pressure would be 14 kPa.

It should be noted that the loss of coolant accident for the LPTR does not lead to a pressure rise in the containment. The open pool tank design and low power level (3 MW) preclude the attainment of appreciable stored energy.

To limit the external pressure on the building shell there are two 6 in. vacuum relief valves installed. They open at 0.43 kPa differential pressure and are each capable of 0.5 m³/s at a differential pressure of 1.4 kPa. This combined flow rate of 1.0 m³/s corresponds to the capacity of the building exhaust fan at standard temperature and pressure.

The foundation was designed on the basis of an investigation of soil conditions at the site. As a result of this investigation, the soils were found to be highly preconsolidated. Since the preconsolidation pressure was greater than the pressure to be added, the resulting settlements were small. To insure the settlement would be small, the following conditions were imposed.

1. The reactor proper was to be supported on a mat type foundation that extended at least 0.9 m below the present or final ground surface, whichever was lower. The maximum dead-plus-live load soil pressure was not to exceed 140 kPa and the maximum soil pressure due to all loads (including wind or seismic) not to exceed 190 kPa.
2. Particular care was to be taken to design the walls of the equipment pit as substantial retaining walls. If these walls should yield, the soil would settle and the reactor would tend to tip.
3. If filling was required under foundations due to careless excavating beyond the designated depths, it was to be done with concrete. No earth filling was allowed under footings.
4. The backfill in any excavation or the fill in any area was to be compacted with pneumatic tampers in 0.15 m layers to 95% of maximum density.

All concrete, except the equipment pit cover slabs, has a minimum compressive strength of 21 MPa at 28 days. Concrete in the equipment pit slabs has a minimum compressive strength of 27 MPa at 28 days. Concrete is in accordance with "Building Code Requirements for Reinforced Concrete" as set forth by American Concrete Institute.²³

The containment building and equipment pit floors are made of reinforced concrete designed to support a uniformly distributed floor load of 95.8 kPa. This design loading applies also to the removable basement cover slabs and the various pit cover plates.

Other design considerations investigated since the initial design and construction are the Wind and Tornado Design Criteria, Sec. 3.3, the Missile Protection Criteria, Sec. 3.5 and the Seismic Design, Sec. 3.7.

The design bases are summarized in Table 3.8-4. The following sections summarize the calculation of the various loading combinations.

Table 3.8-4 Containment structure design.

Wind pressure	958 Pa
Wind speed	39.3 m/s
Internal pressure	13.8 kPa
External pressure	1.4 kPa
Foundation slab live load	96 kPa
Maximum soil pressure	
a) Dead plus live load	144 kPa
b) Dead plus live load plus wind or seismic	190 kPa
Tornado criteria	
a) Wind speed	89 m/s
b) Pressure drop	10.3 kPa
c) Rate of pressure change	2.8 kPa/s
d) Missile speed	65.3 m/s
Seismic criteria	See Sec. 3.7

3.8.2.5.1 Case 1 - The stresses

in the steel shell due to the load were analyzed according to the rules of API-620. The dead load compressive stress was determined from the shell weight of 1.2 MN and the cross sectional area of 0.486 m².

The moment caused by the 0.06 kPa wind pressure was calculated with the shell projected area of 365 m², the cylinder shape factor of 0.6, and the distance from the base to the area centroid. With the flexure formula ($S = \frac{MC}{I}$) the resulting tensile and compressive stresses at the base were found by considering the shell a cantilever beam. The assumption that the flexure formula applies may be questioned because of the low length-to-

depth ratio (about 0.6). The bending stresses are small, however; even if they are increased by a factor of 7 the total stresses are within the allowable values.

Table 3.8-5 lists the results.

3.8.2.5.2 Case 2 - The Case 2 loading is the same as that of Case 1 with the addition of an internal pressure of 14 kPa. The shell is analyzed as a simple cylinder. The resulting hoop and axial shell stresses are determined from API-620, 3.60.9 and 3.10.3.

The roof stresses are determined by considering the roof as a spherical segment of the tank and applying API-620, 3.10.2 e(1). The roof to shell connection is analyzed per

Table 3.8-5 Case 1 results; stresses in the steel shell due to dead weight and wind.

Load	Stress, MPa
Dead weight	-2.38
Wind	± 0.55
Total (maximum)	-2.93
Maximum allowable compressive stress per API-620, 3.05.3 a	-6.4

API-620, 3.12.3, rules for compression ring area. The plenum chamber on the containment building roof, the truck door, the air lock, the escape manddoors and other shell openings are designed and analyzed to the rules of API-620, 3.16.2. The original design calculation by Chicago Bridge and Iron Company,² analyzed the doors and openings for the doors which have to withstand the 14 kPa design pressure. The calculations are brief and contain few details, but do show

that consideration was given to the reinforcement of the openings and to the resistance of the doors to flexure.

The doors and door-frames were extensively modified in 1960 to provide inflatable seals and quicker operation. Reinforcement around some openings was cut away and partially replaced by the new door frames. No design calculations for these changes were available so a check was made on the strength of the doors and the adequacy of reinforcement of the openings for the following:

3 × 4.3 m	Truck door
1.1 × 2.14 m	Airlock doors (2)
.76 × 1.9 m	Escape manddoors (2)

All the doors and all the openings as modified are of satisfactory strength. In most cases the stress to be expected at the full 14 kPa loading is less than 70 MPa.

Table 3.8-6 presents the results of the Case 2 calculations.

Table 3.8-6 Case 2 results; stresses in the steel shell due to dead weight, wind, and 14 kPa internal pressure.

Load	Stresses		Roof Axial, MPa
	Shell Hoop, MPa	Shell Axial, MPa	
Dead weight		-2.3	-0.89
Wind		± 0.55	-0.82
Internal pressure	26.5	13.2	17.0
Total	26.5	11.4 ^a	15.3 ^a

^aMaximum allowable tensile stress for ASTM A283C with 70% weld efficiency per API-620, 3.05.2a is 73.4 MPa.

3.8.2.5.3 Case 3 – This case covers the situation where the only outside loading on the building is external pressure. The solution for the shell critical buckling pressure is obtained using the Windenburg formula from the Appendix of the API-ASME Code. The Windenburg formula is used because it allows solutions for very large ratios of shell diameter to shell thickness and very small ratios of stiffener spacing to shell diameter. The ASME charts for external pressure do not extend to our design parameters.

$$P_c = 2.6 E \left(\frac{t}{D} \right)^{2.5} \left[\frac{L}{D} - 0.45 \left(\frac{t}{D} \right)^{-0.5} \right]^{-1}$$

P_c = buckling pressure.

E = modulus of elasticity, 120 GPa.

t = shell thickness, 6.3 mm.

D = shell diameter, 24.4 m.

L = stiffener spacing, 2.6 m.

Substituting and solving for P_c gives a shell buckling pressure of 5.7 kPa.

The vacuum relief valves on the containment shell are set to open at 431 Pa. They will admit 1.04 m³/s when the building pressure is 1.4 kPa below atmospheric pressure. The 1.4 kPa and 1.04 m³/s flow rate are significant only if the building exhaust damper stays open and the fan (capacity 1.04 m³/s) continue to run during a building seal. Regardless of the probability of the malfunctions, the 1.4 kPa will constitute the maximum allowable external pressure. This gives a safety factor of 4 when compared with the shell buckling pressure. The shell circumferential stiffeners are designed per the API-ASME code for an external pressure of 1.4 kPa.

The roof theoretical buckling pressure was calculated using a formula based on the work of von Karman and Tsien.²⁴

$$P = \frac{2Et^2}{R^2 \sqrt{3(1 - \mu^2)}}$$

P = buckling pressure.

E = modulus of elasticity, 120 GPa.

t = head thickness, 8 mm.

R = spherical radius, 19.5 m.

μ = Poisson's ratio, 0.3.

Table 3.8-7 Case 3 results; pressures and safety factors for buckling of the containment shell under external pressure.

Design external pressure	1.4 kPa
Shell buckling pressure	5.7 kPa
Roof buckling pressure	120 kPa
Shell safety factor	4
Roof safety factor	85

Substituting and solving for P,

$$P = 121 \text{ kPa.}$$

For an external design pressure of 1.4 kPa, the safety factor for buckling is 85. Table 3.8-7 presents these results.

3.8.2.5.4 Case 4 — The analysis for tornado loads uses the criteria listed in Sec. 3.3.2. The 90 m/s wind speed

was converted into a dynamic pressure load on the shell by use of Eq. (1) from Sec. 3.3.1. An azimuth pressure profile was developed by use of wind pressure coefficients from the ASCE paper No. 3769, "Wind Forces on Structures," Ref. 3. Two modes of loading were investigated: (I) the tornado's 10.3 kPa barometric pressure drop is added algebraically to the wind pressure and (II) just the wind pressure. See Figs. 3.8-24 and 3.8-25 for radial pressure profile.

The stresses were determined using a computer code NAOS (nonaxisymmetric analysis of shells). The code uses the finite element method to calculate the shell stresses due

Fig. 3.8-24 Azimuth pressure profile from tornado winds, shell loaded with 10.3 kPa internal pressure.

Fig. 3.8-25 Azimuth pressure profile from tornado winds alone, without internal pressure.

to nonaxisymmetric pressure profile. The model does not consider roof loading. These loads were calculated separately and added.

The roof wind loads are approximated using data from ASCE No. 3269. The data are for wind normal to an arched roof supported above ground. Pressure coefficients are listed for three segments of the arch, the center half and the leeward and windward quarters. It is assumed that a spherical roof such as the LPTR's would constitute a less severe case. The roof pressure coefficients are listed in Table 3.8-8.

Table 3.8-8 Roof pressure coefficients.

Angle, rad	Pressure coefficient	Equivalent internal pressure, kPa
$0-\pi/4$	-0.9	4.4
$\pi/4-3\pi/4$	-0.9	4.4
$3\pi/4-\pi$	-0.5	2.5

To allow for the wind suction on the roof an internal pressure of 4.1 kPa was assumed to act on the entire roof.

The mean hoop stresses as a function of azimuth are shown on Figs. 3.8-26 and 3.8-27.

The mean axial shell stresses are shown in Fig. 3.8-28. They include the effects on the roof of the internal pressure and the dead weight.

Fig. 3.8-26 Hoop stresses vs azimuth from tornado winds, shell loaded with 10.3 kPa internal pressure.

Fig. 3.8-27 Hoop stress vs azimuth from tornado winds alone, without internal pressure.

Fig. 3.8-28 Mean axial stresses from tornado winds, with and without 10.3 kPa internal pressure.

The stresses for mode I, 10.3 kPa ΔP and wind pressure, are shown on Figs. 3.8-19 and 3.8-26. The stresses over the entire azimuth are tensile. Since they are less than the API-620 allowable 73.4 MPa, the containment shell is satisfactory for this mode of loading.

The stresses for mode II, wind pressure only, are shown in Figs. 3.8-27 and 3.8-28. It can be seen that the hoop and axial stresses are both compressive in the 0 to $\pi/6$ azimuth interval. Elsewhere they are tensile except that the axial stress is compressive in the $2\pi/3$ to $8\pi/9$ interval. The reason for this compressive region on the leeward side of the shell is uncertain. Nevertheless, the worst case is on the windward side where a $\pi/3$ segment is in biaxial compression. It is possible that local buckling could occur in this area. However, since most of the remaining shell wall is in tension, collapse is improbable.

The roof stresses for mode I are slightly greater than those for case 2 where the internal pressure was 13.8 kPa. Under mode I loading there is a 10.3 kPa internal pressure due to the tornado plus a 4.1 kPa internal pressure due to the 89.4 m/s tornado suction. This gives a combined pressure of 14.5 kPa. However the resultant tensile stress of 27.8 MPa is still far below the allowable 73.4 MPa.

3.8.2.5.5 Case 5 — The containment building was modeled as an elastic, axisymmetric, shell structure subjected to nonaxisymmetric static and dynamic loading (computer code GHOSH). A finite element mesh consisting of shell elements was used to model the steel structure. Solid elements were connected to the shell at five locations to simulate the effect of stiffeners in the actual structure. The mass density of

the steel was increased to include the effect of insulation weight. The building was considered rigidly connected to its foundation. Shell thickness was varied to model the different thickness of the shell and roof. Lap joints were included by increasing the thickness in the region of the joint. The compression ring located at the roof connection was included as a thicker shell. This has the effect of locally stiffening the shell-roof connection.

A linear elastic analysis was conducted on the model of the containment building shown in Fig. 3.7-5. The model was subjected to dead load, 14 kPa internal pressure which could result from atmospheric conditions, and horizontal and vertical components of earthquake.

Ten modes of vibration were used to determine both horizontal and vertical response to earthquake excitation. The first three periods of horizontal and vertical motion are presented in Table 3.8-9.

Table 3.8-9 Vibration periods of the containment shell.

Mode	Period vibration, ms
Horizontal 1	55
2	42
3	41
Vertical 1	65
2	40
3	32

Figure 3.8-29 shows the displaced shape of the structure amplified 100 times when subjected to the 14 kPa internal pressure.

Figures 3.8-30 and 3.8-31 show lateral and axial displacements due to the pressure loading. The maximum lateral displacement is 2.3 mm and the maximum axial displacement is 19 mm.

Figures 3.8-32 and 3.8-33 show the displaced structural shape for the horizontal and vertical components of earthquake. These shapes had been amplified by 320 and 600 respectively.

horizontal and vertical components of earthquake. These shapes had been amplified by 320 and 600 respectively.

Figures 3.8-34 and 3.8-35 show the lateral and axial displacements combined for both horizontal and vertical components of earthquake. A maximum lateral displacement of 3.4 mm and a maximum axial displacement of 3.9 mm were calculated.

The stresses in the shell and roof as calculated per the computer code for the 13.8 kPa internal pressure and dead load are listed in Table 3.8-10. The elevations chosen are approximately at the midpoints between stiffeners or lap joints. Also listed for comparison are the stresses due to the horizontal and vertical components of the earthquake. The seismic stresses generally are 10 to 20% of the 14 kPa internal pressure stresses. Superposing the seismic and pressure stresses still fails to approach the maximum allowable tensile stress of 73.4 MPa (API-620, 305.2 a).

Fig. 3.8-29 Deformation of the LPTR containment building, 14 kPa internal pressure. Displacement amplification, $\times 100$. (Dots indicate unstressed position.)

3.8.2.5.6 Case 6 – The uplift on the containment building shell due to the 14 kPa internal pressure is 84 kN/m. The wind drag on the roof causes an additional uplift of 4.89 kN/m. The wind pressure on the vertical wall of the shell adds a peak uplift on one side of 3.5 kN/m. Opposing this total uplift of 92.5 kN/m is the dead weight of the shell, the shell concrete footings, and the soil between the top of the footings and the bottom of the building floor slab.

The containment building shell footing, a massive annular ring 76.5 m in circumference, is composed of an upper reinforced concrete ring 0.61 $\times$ 0.51 m wide keyed to a spread footing 2.3 $\times$ 1.2 m. In the spread footing on 0.91 m centers are 84 anchor bars, 35 mm square by 1.2 m long. These anchor bars are set in the spread footing and project into the upper

concrete ring. Prior to pouring the upper ring, the first shell course was welded to these anchor bars, pipe sleeves inserted, and reinforcing steel hooks and the antileak ring welded to the buried shell course.

Table 3.8-10 Shell and roof stresses.

Elevation m	Hoop stress (MPa)			Axial stress (MPa)		
	Static loads ^a	Earthquake		Static loads ^a	Earthquake	
		Horizontal	Vertical		Horizontal	Vertical
Shell						
1.14	26.7	0.17	0	13.8	3.46	1.14
3.30	26.7	0.34	0	13.7	2.93	1.14
5.44	26.6	0.45	0	13.6	2.52	1.03
7.60	26	0.62	0	13.4	2.00	1.00
9.78	26.8	0.69	0	13.0	1.66	1.03
Roof						
12.1	51.4	1.03	3.11	37.5	3.56	2.68
13.6	43.9	3.73	2.87	41.0	3.63	3.63
15.2	37.5	2.68	3.29	21.9	1.56	2.00
15.8	19.5	3.63	2.59	1.00	9.58	3.29

^aDead load plus 14 kPa internal pressure.

Fig. 3.8-30 Lateral component of displacement of LPTR containment building under 14 kPa internal pressure.

Fig. 3.8-31 Axial component of displacement of LPTR containment building under 14 kPa internal pressure.

Weight of the soil above the spread footing is 5.55 kN/m based on 1.4 Mg/m^3 density. Weight of the containment building foundation based upon 2.4 Mg/m^3 concrete is 72.96 kN/m. Weight of the steel shell and insulation is 15.1 kN/m based upon a 118 Mg shell. The total dead weight is 93.6 kN/m, 1.1 kN/m more than the total uplift.

The anchor bars that transfer the net uplift to the footings have a cross sectional area of 12.2 cm^2 . Based on their 0.91 m, the net uplift per bar is calculated as follows:

Total uplift	92.6 kN/m
Shell dead weight	<u>15.1 kN/m</u>
Net uplift	77.5 kN/m
Net uplift per bar	70.8 kN
Tensile stress per bar	58 MPa

Fig. 3.8-32 Deformed shape of LPTR containment building, using rms approximation, resulting from horizontal component of SSE. Displacement amplification, X 320.

Fig. 3.8-33 Deformed shape of LPTR containment building, using rms approximation, resulting from vertical component of SSE. Displacement amplification, X 600.

Maximum allowable tensile stress	138 MPa
Maximum allowable bond stress	931 kPa
Actual bond stress	414 kPa
Embedment length required	0.55 m
Embedment length provided	1.22 m

The above values for dead weight and anchorage indicate the shell is adequate for Cafe 6 loading.

3.8.2.5.7 Case 7 ~ Case 7 treats the effect on the foundation of the combination of dead weight and external pressure loading. The loading in Case 7 causes the highest soil pressure and can be compared with the maximum allowable soil pressure of 140 kPa. From Case 6, the weight of the shell and foundation is 93.6 kN/m. External pressure of 1.4 kPa on the building causes a downward force of 663 kN or 8.68 kN/m of circumference. The total unit force of 102 kN/m when divided by the footing width of 2.3 m gives a pressure of 44.8 kPa. A soil surcharge of 7.20 kPa is added to give a total soil pressure of 51.9 kPa. This represents a little over one third of the maximum allowable soil pressure.

Fig. 3.8-34 Lateral component of displacement of LPTR containment building from combined horizontal and vertical components of SSE (rms approximation).

Fig. 3.8-35 Axial component of displacement of LPTR containment building from combined horizontal and vertical components of SSE (rms approximation).

3.8.2.5.8 Case 8 – Case 8 treats the effect on the foundation of the combination of dead weight and tornado wind loading. The loads for this case are identical to those of Case 4 where the steel shell was analyzed. Figure 3.8-28 indicates a maximum axial stress of 18.9 MPa. Multiplying this stress by the shell cross sectional area of $6.35 \times 10^{-3} \text{ m}^2/\text{m}$ circumference determines the total uplift per metre:

Total uplift	120 kN/m
Net uplift per anchor bar (0.914 m spacing)	110 kN
Tensile stress per bar ($1.2 \times 10^{-3} \text{ m}^2$)	90.3 MPa
Maximum allowable tensile stress	138 MPa
Maximum allowable bond stress	931 kPa
Actual bond stress	655 kPa
Embedment length required	0.85 m
Embedment length provided	1.2 m

As in Case 6, the anchorage is sufficient to sustain the uplift loads.

3.8.2.5.9 Case 9 Case 9 combines dead weight and live loading and calculates the effects on slabs and walls. The floor slab is designed on the basis of maximum tensile strength "S" in the concrete as computed by Pickett's formula²⁵:

$$S = \frac{3.36 P}{d^2} \left(1 - \frac{\sqrt{a/l}}{0.925 + 0.22 a/l} \right)$$

where

P = load positioned in corner = 95.8 kPa

d = slab thickness = m

a = radius of load = m

l = radius of relative stiffness

$$= \sqrt[4]{\frac{D}{K_B}} = m$$

K_B = modulus of subgrade reaction = 47.1 kN/m³

D = flexural rigidity

$$= \frac{Ed^3}{12(1 - \mu^2)}$$

E = modulus of elasticity of concrete

= 27.5 GPA

μ = Poisson's ratio = 0.15

The modulus of subgrade reaction was assumed based on a subgrade of gravel-sand-clay. A slab thickness of 0.15 m was selected with protected corners (20% of the load transferred).

The maximum tensile stress of 1.76 MPa (calculated by the above formula) occurs when the load radius is 0.53 m. The modulus of rupture of 21 MPa concrete is 3.3 MPa (P.C.A.) resulting in a safety factor of 1.9. The slab is reinforced with No. 3 bars on 0.30 m centers, each way.

Particular care was taken to provide the equipment pit with substantial retaining walls because if these walls should yield, soil beneath the reactor shield foundation could settle causing the reactor to tip. From 1957 to date there has been no noticeable settling. The east and west retaining walls of the equipment pit were designed as counterfort walls in order to withstand the originally specified seismic forces and horizontal soil and surcharge pressures due to a 95.8 kPa floor loading. All counterfort beams, walls, and the floor of the basement are reinforced with steel.

Removable slabs cover the equipment pit. The weight of the largest slab is 56 kN. Each slab has four symmetrically placed pick-up hooks. The maximum load per pick-up assembly is 14 kN; however, the type of pick-up used is capable of lifting an

ultimate load of 170 kN. The factor of safety is 12.0. All slabs are 0.30 m thick, 26 MPa reinforced concrete, and are designed to support the specified floor loading.

3.8.2.6 Design Evaluation — Examination of the results for the various load combinations discloses that the stresses are below the maximum allowable for all cases. In the cases where external pressure is present, there is a safety factor of at least 4 on the buckling pressure. For the cases involving the concrete foundation the footing size, anchor bar stresses, and embedment length are sufficient for the loads applied. Also, the maximum soil bearing pressure is well below the maximum allowable.

As discussed in Sec. 3.5, tornado generated missiles are capable of penetrating the containment shell. If this occurs, the containment building loading would go from the Mode I loading of Case 4 to Mode II. In other words, the 10 kPa internal pressure in Mode I would be lost. The Mode II loading with a 90 m/s wind speed is the worst case. The tensile stresses associated with the Mode II loading are minor as can be seen in Figs. 3.8-27 and 3.8-28. However, the presence of compressive stresses on the windward side is of some concern. As stated in Case 4 it is possible that local buckling could occur in that area. Nevertheless, collapse of the shell and roof is improbable because of the presence of the crane structure inside the building. The separation between the roof and the crane structure is 80 mm at the top of the monorail truss. If the shell wall buckled the roof would soon contact the crane structure, tending to limit further movement.

3.8.3 Concrete and Structural Steel Internal Structures of Steel or Concrete Containment

This section provides descriptions of the internal structures as constructed. Refer to Fig. 3.8-21 for location of these structures within the containment building.

3.8.3.1 Equipment Pit — The equipment pit is located on the west side of the containment building as shown in Fig. 3.8-21. It is 10 m long, 4.0 m wide and 4.9 m deep with a mezzanine floor at the north end 2.1 m above the equipment pit floor.

The equipment pit houses the pumps, heat exchanger and demineralizer of the primary loop, waste tanks, air purge filter and blower, pneumatic tube blower and control cabinet, and exposed process piping. The equipment pit is constructed entirely of reinforced concrete, and is covered with five .30 m thick removable concrete slabs set flush with the operating floor. There is one cover section on the east-west centerline of the building which is not removable. This cover section carries the load of the west thermal column door when it is retracted from the reactor shield. The removable slabs are capable of carrying a uniformly distributed load of 100 kPa. These slabs weigh approximately 55.6 kN and are removable by the building crane for access to

the equipment. Access to the basement is through a hatch at the north end of the equipment pit.

3.8.3.2 Containment Building Floor — The building floor is a 0.15 m thick concrete slab, with a 0.1 m curb around the entire circumference. An antileak ring is welded to the steel ring wall below the concrete surface, and copper waterstop is used wherever there is a break in the concrete construction. To facilitate decontamination, the floor is surface hardened and covered with five coats of a strippable plastic paint. The floor has several drains and a number of covered pits, including the large equipment pit surrounded by narrow drainage gutters. Other than this, the floor is quite free of breaks or obstructions. In order to minimize floor clutter, fourteen 4-in. conduits were buried in the concrete floor, running radially from the reactor shield faces to the building wall and capped at floor level for use with experimental equipment or for utilities extensions.

Expansion joints are provided in the floor to prevent the floor slab from buckling or cracking and to isolate any vibrations that might be transmitted to the steel building shell. All construction joints in the concrete floor slab and between concrete floor and walls contain a copper water stop, soldered to form an unbroken periphery to minimize leakage into the subsoil. These joints are keyed and have a minimum thickness of 0.2 m.

3.8.3.3 Floor Storage Pit — A floor storage pit, able to accommodate a large number of fuel elements and a variety of other radioactive items, is situated close to the reactor in the area serviced by the crane (Fig. 3.8-21). Thirty 5-in. aluminum pipes are sunk to a depth of 3.4 m. These are capped on the bottom and can be filled with water. Similarly, half a dozen larger-diameter carbon-steel pipes provide storage space for other radioactive items which require temporary or extended storage. Originally designed for spent-fuel-element storage, this pit now provides storage of the core only during periods when removal of the elements from the tank is required for intank maintenance or modifications.

3.8.3.4 Beam Plug Storage — A shielded storage facility is located on the southeast of the containment building (Fig. 3.8-21). It is $4.3 \times 7.6 \times 2.4$ m high and contains 20 horizontal storage tubes of various diameters from 0.12 m to 0.44 m. All the tubes are 3.05 m long. The structure stands on a concrete footing and has concrete walls on all four sides of 0.305 m thickness. This concrete "box" is filled with compacted earth. The pipes are covered with at least 1.06 m of earth. The earth at the top is topped with 0.05 m of mastic sealer. The tubes are closed with magnetite concrete plug 0.13 m thick in 6.3-mm steel shells.

3.8.4 Other Category I Structures

This section provides a description, original design information, and seismic analysis results of other Category I structures.

3.8.4.1 Control Rod Actuator Support — The control rod actuator support is part of a larger structure (the reactor top structure) which serves the additional purpose of covering the top of the reactor pool, see Figs. 3.8-36 and 3.8-37. The control rod actuators, discussed in Sec. 4.2.3 are mounted on the actuator support plate. This plate is supported on an angle frame so that its top is 0.97 m above the top of the reactor shield. The entire frame is 0.69 m by 0.86 m by 2.26 m high and is made from $80 \times 80 \times 6$ mm steel angles or better. The frame is of welded construction.

Fig. 3.8-36 Control rod actuator support structure, top view with cover plates removed.

Fig. 3.8-37 Control rod actuator support structure, sectional elevation.

The actuator support plate is a 2024-T4 aluminum alloy plate 648 mm by 825 mm with a minimum finished thickness of 41 mm. The plate has thirty-five 22-mm drilled holes corresponding in location to the 35 fuel element holes in the fuel grid plate. These holes are large enough to permit free movement of the actuator rack through

them. There are four 9.5 mm holes in a square pattern centered on each of the rack clearance holes. This pattern matches the bolt holes in the actuator mounting flange. The position of the actuator support plate can be adjusted vertically and horizontally to obtain the proper relation to the fuel grid plate. Lateral adjustment was made possible by making the actuator support plate smaller than the inside dimensions of the frame of the housing. An adjustment of 13 mm in either direction can be made. The final position is held by eight oval point set screws in tapped holes in the corner post angles of the actuator housing. These set screws bear directly on the sides of the actuator support plate. The corners of the actuator support plate are held down firmly against the plate support frame angles by 12.7-mm-diam bolts threaded into the actuator plate and secured by nuts on the underside of the frame. An actuator with control rod attached weighs approximately 356 N. There are five control rods resulting in a nominal loading of 1.8 kN on the actuator support plate. For a 41 mm thick plate the resulting stresses are negligible; the stresses on the angle support structure are also negligible.

Supporting the actuator structure are two cantilever beams (5 WF 18.5) attached to the top of the reactor shield, Fig. 3.7-8. The use of cantilever beams permits the largest free area at the top of the reactor pool for manipulating the handling tools and moving the fuel elements from the core to the storage racks. These cantilevers are designed for a live load 4.8 kPa on the pool tank cover plates in addition to the dead load of the cover plates, actuator support structure, actuators and actuator support plate. The beams are sized to minimize deflections under full load. The attachment of the cantilevers to the reactor shield is accomplished by the use of eight hooked anchor bolts, 19-mm-diam by 508 mm long, embedded in the concrete.

Analyzing the support beam as a statically determinate cantilever beam, the maximum bending stress is 45 MPa and the deflection of the beam at its extreme end is 2 μm for the full live load of 4.8 kPa. This deflection would move the neutron absorber section of the control rod a negligible 10 μm . The maximum anchor bolt tensile stress would be 46.9 MPa and the concrete under the fixed end of the beam would be stressed to 3.4 MPa compression under conditions of full load. Actuator support design data is presented in Table 3.8-11.

See Sec. 3.7.1 and 3.7.2 for seismic analysis procedure. The top structure was considered as a lumped mass cantilevered from the reactor shield. A conservative estimate of 910 kg was made for the equipment and housing of the structure. The actual mass is about 680 kg. The mass of the cantilever beams was neglected.

The top structure was modeled as a single-degree-of-freedom system and evaluated by the spectral response method. It was assumed that the SSE spectrum was not altered by the shield and can be used as a direct input for analysis of the top structure.

Table 3.8-11 Actuator support design.

Applicable code	None
Structural steel	ASTM A 7
Dead load	5.3 kN
Live load on pool cover plates	4.8 kPa
Cantilever stress	45 MPa
Cantilever safety factor (based on yield strength)	4.6
Cantilever deflection	Negligible
Anchor bolt stress	46.9 MPa
Anchor bolt safety factor (based on yield strength)	4.4
Compression in concrete	3.4 MPa
Allowable compressive strength of concrete	20.7 MPa

Table 3.8-12 Results of top structure calculations.

Earthquake loading	Cantilever beam stiffness, MN/m	Fundamental period, ms	Maximum displacement, mm
North-south	61.9	24	0.071
East-west	1430	5	0.0025
Vertical	12.4	54	0.48

Figures 3.8-36 and 3.8-37 show the actual structure and Fig. 3.7-8 shows the model used for this analysis. The response of the top structure to horizontal and vertical earthquake components was computed. Results of the analysis are shown in Table 3.8-12.

Analysis of the anchor bolt stresses indicates a shear stress of 14.6 MPa and tensile stress of 62.5 MPa. These values are the combined static and seismic stress value in the "worst case." They were compared with AISC allowable for combined bolt stresses and found to be acceptable. No problems are expected at the support and the seismic-caused displacements are negligible.

3.8.4.2 Heat Exchanger Support —

The heat exchanger structural support rests on four 0.23 × 0.19 m concrete piers. The piers are cast into the equipment pit floor with a 19-mm-diam anchor bolt in each pier. The four vertical corner members are 0.1 × 0.1 × 0.009-m angles. They have 0.15 × 0.15 × 0.02-m-

thick base plates welded at the bottom and 9-mm-thick gusset plates welded at the top. The horizontal members that enclose the top are eight 11.5 channels. The horizontal members are joined to the frame with nine 19-mm-diam bolts at each end.

The heat exchanger rests on the frame at four points, the points of application of load being the center of each horizontal member. The connection is made by a 19-mm-diam bolt at each point. The complete frame, measuring 1.1 × 1.1 × 1.6 m, is shown in Fig. 3.7-7.

Heat exchanger support design

Applicable code	AISC
Structural steel	ASTM A-7
Heat exchanger weight: Full	44.5 kN
Empty	26.8 kN
Seismic criteria	Sec. 3.7.1 and 3.7.2.1

The support structure was modeled as a pin-ended frame with rigid internal joints. See Fig. 3.7-7. The stiffening effect of the connecting piping and the dead load of the support were neglected in the analysis. The total heat exchanger load of 4.4 kN was distributed to each frame. The seismic analysis procedure is discussed in Sec. 3.7.2.1.

The structure lateral stiffness was determined by calculating the displacement resulting from a unit lateral load

$$K = \frac{P}{\Delta} = \frac{1 \text{ kip}}{0.287 \text{ in.}} = 613 \text{ kN/m.}$$

Other calculated values are shown below:

$$\text{Mass} = M = \frac{W}{g} = 2.3 \text{ Mg.}$$

$$\text{Period} = T = 2\pi \sqrt{\frac{M}{K}} = 0.38 \text{ s.}$$

$$\text{Spectral acceleration} = S_a = 20.1 \text{ m/s}^2 \quad \text{for } T = 0.38 \text{ s} \\ \xi = 2\% \text{ damping.}$$

$$\text{Maximum lateral displacement} = \Delta_{\max} = \frac{MS_a}{K} = 75 \text{ mm.}$$

Member forces were then determined by conducting a static analysis of the frame subjected to the heat exchanger dead load and the lateral displacement due to the earthquake. Results of the analysis are shown in Fig. 3.8-38.

Fig. 3.8-38 Analysis of fully loaded heat exchanger support (earthquake loading).

Anchor bolt stresses exceed allowable design values and extreme fiber stresses in both beam and columns exceed ultimate values for A-7 structural steel. The deficiencies in the structure can be corrected by adding cross bracing to the legs to increase the lateral stiffness of the support.

An analysis was conducted with cross bracing to determine an adequate member size to reduce stresses to allowable values. The cross bracing was rigidly connected to the support legs and assumed to resist tension forces only (no compression).

Eight 130 × 13 mm steel bars welded to the support legs as shown in Fig. 3.8-39 are adequate to brace the structure for the earthquake loading. Results of this analysis

are shown in Fig. 3.8-40. For this modification all member and bolt stresses are below allowable design values for A-7 structural steel.

Calculated values are:

$$K = \frac{P}{\Delta} = 16.1 \text{ MN/m}$$

$$T = 2\pi \sqrt{\frac{M}{K}} = 70 \text{ ms}$$

$$\text{Spectral acceleration} = S_a = 11.8 \text{ m/s}^2$$

$$\text{for } T = 0.07 \text{ s}$$

$$\xi = 2\% \text{ damping}$$

$$\text{maximum lateral displacement} = \Delta_{\max} = \frac{MS_a}{K} = 1.6 \text{ mm.}$$

Fig. 3.8-39 LPTR heat exchanger support structure crossbracing scheme to improve resistance to lateral earthquake loads.

3.8.4.3 Reactor Core Support Structure

The core support structure consists of an aluminum frame and the grid plate. The grid plate, Fig. 3.8-41, is

0.127-m-thick 1100 aluminum alloy plate used to support fuel and reflector elements in their proper lattice positions. A maximum array of up to 5 × 7 elements is allowed by 35 holes 62 mm in diameter.

The support frame is fabricated from four 3 × 3 × 0.375 in. 5052 aluminum alloy angles, one at each corner of the rectangular grid plates. The angles are welded to a 19-mm-thick bearing plate which in turn is welded to the bottom of the tank. The grid plate, after being accurately leveled and set to height in the field, was tack welded and pinned to the support angles. Four 6.3-mm-thick 5052 aluminum enclosure plates are welded

Fig. 3.8-40 Analysis of fully loaded heat exchanger support structure with crossbracing (earthquake loading).

to the four corner angles. The completed support is then in the form of a rigid box. It measures $0.38 \times 0.71 \times 1$ m high. The support was designed with emphasis on rigidity rather than keeping within allowable stress values. Therefore, it is very much over-designed. The nominal load on the grid plate is approximately 2.2 kN. This results in a compressive stress of 170 kPa in the 6.3-mm-thick box sides. Obviously, there is no danger of failure with the present loading and configuration.

A seismic analysis of the grid plate support was conducted. The structure was modeled as a single-degree-of-freedom system. The spectral response method was used to calculate the maximum expected seismic displacement as follows:

$$M = \text{grid plate mass} = \frac{2.2 \text{ kN}}{9.8 \text{ m/s}^2} = 226.9 \text{ kg.}$$

$$K = \text{support lateral stiffness} = \frac{3EI}{L^3} = 79.3 \text{ MN/m.}$$

$$f = \frac{1}{2\pi} \sqrt{\frac{K}{M}} = 94 \text{ Hz.}$$

$$T = \frac{1}{f} = 10 \text{ ms.}$$

$$S_a = 5 \text{ m/s}^2 \text{ for } T = 10 \text{ ms}$$

$\xi = 2\% \text{ damping.}$

$$\Delta = \frac{MS_a}{K} = 0.015 \text{ mm.}$$

Since the seismic displacement was so small no problems were expected for the grid plate support. No further analysis was conducted.

All dimensions in inches.
One in. = 25.4 mm

Fig. 3.8-41 LPTR core support structure.

3.8.4.4 Containment Air Scrubber Building — The air scrubber building houses a system designed for pressure venting of the containment building atmosphere. The system is only required in the event that there is a release of fission products to the containment and a subsequent increase in containment building pressure. The system is more fully described in Sec. 6.2.3.

The air scrubber building is a commercial, prefabricated structure set on a 0.15-m-thick reinforced concrete slab. The soil under the slab is compacted to 90% of maximum density. The building provides protection from wind and weather and is designed to Uniform Building Code standards. At the time of its installation there were no tornado or missile protection requirements. It is assumed that the building would not withstand either tornadoes or wind generated missiles. Failure of the building due to the above causes could result in failure of the air scrubber system. However, neither of these events has been shown to cause any accident internal to the containment building that could result in release of fission products. The loss of the air scrubber system is therefore not critical. It would be highly improbable that a tornado or 50-year wind would coincide with a release of fission products to the containment.

Containment air scrubber building design

Applicable code	Uniform Building Code
Wind load	720 Pa
Slab thickness	0.15 m
Concrete compressive strength	17 kPa at 28 d
Building	Steel, Armco Series S, Type 1
Size	3.7 × 5.3 × 2.7 m

3.8.4.5 Bridge Crane Support Structure — A 133-kN traveling crane services the main area of the building, the reactor pool tank, and experimental facilities. The crane has an auxiliary 8.9-kN hoist with the same floor coverage, 15.5 × 10.4 m. Both hoists have a maximum hook height of 11 m, clearing the highest pool-top structure by about 1.2 m. An 18 kN monorail hoist is mounted above the main crane, with a 9.8 m run perpendicular to the main crane rail, and a hook height of 14 m. The crane trolley and all hoists have variable-speed motors, with a low-speed range and magnetic brakes. As an added precaution, special provisions have been made that prevent the long handling tools used in the reactor pool tank from touching the crane's electrical conductors.

The first phase of the crane analysis consisted of three-dimensional analyses of the support structure with the hook load located at various spacial positions. The direction of earthquake input was varied to determine the worst combination of hook position and earthquake direction. For these preliminary studies the crane support was modeled as

a one story frame with its mass lumped at various positions at the top of the frame. The supporting truss was modeled as a beam with a moment of inertia equivalent to that of the truss.

The worst case situation determined was with the hook load located on a long span truss near the one quarter span point. The earthquake input direction was selected to be in the plane of the long span truss. This combination of loading and earthquake direction represents the worst situation from a structural view point. A second loading position with the crane at mid span directly over the pool tank was selected to represent the worst condition from a nuclear accident point of view. Results of these studies indicated that a long span frame with these two loading conditions would adequately represent the actual structural behavior. See Fig. 3.8-42.

Only masses that would be lifted to the top of the structure were considered to effect the dynamic response of the structure. Masses moved around the operating floor and lifted only a few centimetres were neglected. The maximum expected crane hook load

Fig. 3.8-42 LPTR bridge crane structure, showing the location of worst case live loads.

was then determined by finding the crane capacity and determining actual loading conditions that occurred at the LPTR facility. The crane has a capacity of 133.4 kN (30 kips), but only a maximum load of 31 kN (7 kips) would be raised to the top of the structure. Table 3.7-4 shows loads and frequency of occurrence based on past usage.

For this seismic analysis a maximum expected crane load of 31 kN was selected.

A detailed linear elastic analysis of the long span frame was then conducted for the dead load plus the 31.1 kN (7 kip) maximum expected live load. The crane support was modeled as a plane frame. The model used for this analysis is shown in Fig. 3.7-6. Properties of A-7 structural steel were used in the analysis: $E = 210$ GPa, ultimate $= 414$ MPa. Examination of the member connection details suggested semirigid internal joints and pinned column connections at the base.

An estimate of the lateral stiffness K , of the frame was determined by imposing a unit force $P = 4.5$ kN (1 kip) and determining the resulting lateral displacement Δ . The stiffness was then determined by:

$$K = \frac{P}{\Delta} = 2.6 \text{ MN/m}$$

The fundamental period of vibration T , of the structure was calculated by considering the frame as a single-degree-of-freedom system with its mass, M , lumped at the upper level and having lateral stiffness K .

$$T = 2\pi \sqrt{\frac{M}{K}}$$

Spectral accelerations were obtained as a function of period for 2% damping. The response spectrum method was then used to determine the maximum lateral displacement. Results of these calculations for both load cases are shown in Table 3.8-13. A maximum lateral displacement of 86 mm was calculated.

Table 3.8-13 Summary of crane calculations for both loading cases.

Loading condition	Dead load plus max expected load, W kN	Mass, $M = \frac{W}{g}$	Period, T s	Spectral acceleration, S_a m/s ²	Max lateral displacement, $\Delta_{\max} = \frac{MS_a}{K}$ mm
Case 1	$87.2^a + 31 = 118$	12.0	0.43	18.6	86
Case 2	$87.2^a + 15.6 = 103$	10.5	0.40	19.6	79

^aFrame weight = 65.2 kN, crane hoist and trolley weight = 22.0 kN. Total dead load = 87.2 kN.

Member forces were then determined by conducting a static analysis of the plane frame subjected to the maximum lateral displacement $\Delta_{\max}$ and the loading conditions stated previously.

In conclusion lateral displacements of the structure are acceptable. There now exists a 140 mm clearance between the crane support structure and the containment building walls.

In Case 1 (structural) examination of the member forces indicates the anchor bolts at the pinned base plate connection are overstressed, extreme fiber stresses in the columns exceed ultimate values, and some secondary compression struts are overstressed. AISC allowable stresses in combined tension and shear for A-307 bolts were used to evaluate the anchor bolts.

In Case 2 (nuclear) the anchor bolts exceed the allowable shear stresses and some secondary struts are overstressed. Ultimate stresses are not exceeded under this loading condition.

In view of the above results, certain modifications were recommended. Cover plates were designed for the column and brace that would reduce the stresses in these members during the earthquake loading. Figure 3.8-43 shows the details of the modification needed. An analysis similar to that conducted on the original structure was conducted on the modified structure. The following properties were used for the modified frame:

$$K = \frac{P}{\Delta} = 6.3 \text{ MN/m.}$$

$$T = 2\pi \sqrt{\frac{M}{K}} = 0.27 \text{ s.}$$

$$\text{Spectral acceleration} = S_a = 19.6 \text{ m/s}^2 \quad \text{for } T = 0.27 \text{ s} \\ \xi = 2\% \text{ damping}$$

$$\text{Maximum lateral displacement} = \Delta_{\max} = 38 \text{ mm}$$

The worst case structural loading condition was used in the analysis. For the cover plate design shown, all member and anchor bolt stresses were below acceptable values.

3.8.4.6 Description and Seismic Analysis of Category I Piping – The Category I piping consists of the reactor primary coolant system and the secondary coolant system that is connected to the heat exchanger. This latter piping is only of concern in the equipment pit and for the sole reason that it could possibly damage the heat exchanger support structure during an earthquake.

Fig. 3.8-43 Suggested modifications to the LPTR crane support structure to improve resistance to lateral earthquake loads.

The secondary coolant piping is 8 in. Sch. 40 carbon steel pipe per ASTM A53, GR B. An isometric drawing of the equipment pit piping system appears in Fig. 3.8-44. The inlet run to the heat exchanger is supported by three brackets attached to the wall and one tall stanchion. The pipe is attached to the brackets with U-bolts that effectively prevent motion in any direction. The support at the post prevents motion only in the down direction. The bellows adjacent to the heat exchanger reduced the effects of piping-heat-exchanger interaction. The bellows were originally installed to provide

3.8-50

All dimensions in metres

Fig. 3.8-44 Secondary coolant piping in the equipment pit.

an offset required because of misaligned piping. They were not required for thermal expansion effects. The outlet run from the heat exchanger has a single support. It is identical to the stanchion on the inlet run.

The primary coolant piping is 6 in. and 8 in. Sch. 40 aluminum pipe per ASTM B241, 6063-T6. The 8 in. pipe is buried from the reactor pool tank to the equipment pit. The 6 in. run is only in the equipment pit. An isometric drawing of the equipment pit primary coolant piping is shown in Fig. 3.8-45. The pipe run in the equipment pit is supported by one rigid rod hanger and a number of resting supports which permit sliding only in the axial direction. The supports are on stanchions which vary in height from 0.46 m to 2.44 m.

3.8.4.6.1 Secondary Coolant Piping – The inlet leg to the heat exchanger is rigidly attached to the equipment pit wall at three points for the first half of its run. The lengths between supports are small and the pipe is very stiff (large moment of inertia). This portion of the run will move with the wall during an earthquake and will be subjected to the maximum horizontal ground acceleration of 5 m/s^2 . The remaining portion of the inlet run is modeled as a cantilevered beam for horizontal seismic vibrations in the east-west direction. No credit is taken for the 'ellows connection at the heat exchanger nor the support at the stanchion. The analysis on the inlet leg was made just for the east-west direction because it represents the worst case. Seismic analysis in the vertical direction would not ignore the stanchion support and, hence, stresses in this direction would be less. Seismic analysis in the north-south direction would likewise give lower stresses since the unsupported leg is shorter.

The models used to represent the secondary piping as shown in Fig. 3.8-46. The analysis for the natural frequency follows the procedure outlined in Ref. 26.

$$f_n = \frac{\alpha}{L^2} \sqrt{\frac{EI}{\frac{w}{4} + P}}$$

where

f_n = natural frequency of pipe (first mode), Hz.

α = coefficient to account for end conditions and mode,

= $.0121 \sqrt{\frac{N}{\text{m} \cdot \text{Pa}}}$ for cantilever.

E = modulus of elasticity,

= 210 GPA.

Fig. 3.8-45 Primary coolant piping in the equipment pit.

2

Secondary coolant piping models

Fig. 3.8-46 Models of secondary coolant piping.

$$I = \text{moment of inertia of pipe,} \\ = 3010 \text{ cm}^4 \text{ (8 in. Sch. 40).}$$

$$w = \text{weight per ft of pipe (including contents and insulation)}$$

$$P = 733 \text{ N/m end load (valve, bellows and pipe weight)} \\ = 1.8 \text{ kN}$$

$$L = \text{length of pipe} \\ = 1.73 \text{ m}$$

$$f_n = \frac{0.0121}{1.73^2} \sqrt{\frac{207 \times 10^9 \times 30 \times 10^{-6}}{\frac{733}{4} + \frac{1780}{1.73}}} \\ = 20.7 \text{ Hz.}$$

The period in seconds is:

$$T_n = \frac{1}{f_n} \\ = \frac{1}{20.7} = 0.05 \text{ s.}$$

The spectral acceleration corresponding to this period can be obtained from Fig. 3.7-1 for 2% damping.

$$S_a = 5 \text{ m/s}^2$$

This acceleration is equal to that of the supporting structure, the equipment pit. No dynamic amplification is expected.

The above method can also be used on the exit leg from the heat exchanger. The results for the east-west direction, the worst case, are:

$$f_n = 5.45 \text{ Hz}$$

$$T_n = 0.18 \text{ s}$$

$$S_a = 18.1 \text{ m/s}^2 \text{ (Fig. 3.7-1).}$$

The maximum moment for this case, a uniformly loaded cantilever beam with concentrated end load, is:

$$M_{\text{horiz}} = \left(PL + \frac{1}{2} w L^2 \right) S_a$$

$$= (1780 \times 3.96 \times 0.5 \times 733 \times 3.96^2) 18.1$$

$$= 23.7 \text{ kN}\cdot\text{m}$$

The vertical spectral acceleration for the exit leg was obtained by considering the pipe as a cantilever beam with an eccentric load at the heat exchanger end. The support at the stanchion and bellows is neglected. Again using Ref. 26:

$$f_n = \frac{0.0121}{L_b^2} \sqrt{\frac{\alpha}{\beta}}$$

and

$$\alpha = \frac{E_b I_b}{\frac{1}{4} w_a \left(\frac{L_a}{L_b} \right) + \frac{1}{2} w_b + \frac{P}{L_b}}$$

$$\beta = 1 + \frac{I_b}{I_a} \left(\frac{L_a}{L_b} \right)^3 + 3 \frac{E I_b}{G J_a} \left(\frac{L_a}{L_b} \right)$$

where

$$L_b = \text{eccentric leg} = 0.61 \text{ m}$$

$$L_a = \text{cantilever leg} = 3.96 \text{ m}$$

$$I_a = 72.5 \text{ in.}^4 = 3010 \text{ cm}^4$$

$$I_b \sim 72.5 \text{ in.}^4 \text{ (considered same as pipe)} = 3010 \text{ cm}^4$$

$$w_a = 733.2 \text{ N/m}$$

$$w_b = 733.2 \text{ N/m}$$

$$P = (\text{valve and bellows weight}) 1334 \text{ N}$$

$$E = 210 \text{ GPA}$$

$$G = \text{torsional shear modulus} \\ = 75.8 \text{ GPA}$$

$$J = \text{polar moment of inertia} \\ = 6020 \text{ cm}^4$$

Substituting in above equations yields:

$$\alpha = 8.47 \times 10^6$$

$$\beta = 284$$

$$f_n = 5.6 \text{ Hz}$$

$$T_n = 0.18 \text{ s}$$

$$S_a = 12.0 \text{ m/s}^2 \text{ (from Fig. 3.7-1 where vertical acceleration is } 2/3 \text{ horizontal acceleration)}$$

The moment equation for the vertical direction is the same as for the horizontal direction. In addition, there is the moment due to mass only or:

$$M_{\text{vert}} = \left(\frac{M}{S_a} \right)_{\text{hor}} (1 \pm S_a)_{\text{vert}}$$

$$= (\text{maximum}) 28.560 \text{ kJ.}$$

The Nuclear Power Piping Code, Ref. 22, states that the primary stress intensity limit is:

$$B_1 \frac{PD}{2t} + B_2 \frac{D}{2t} M_i \leq 1.5 S_m$$

where

B_1, B_2 = primary stress indices (Table D-201, p. 150, Ref. 2)

$$B_1 = 0.5$$

$$B_2 = 1.0$$

P = design pressure = 690 kPa.

D_o = pipe o.d. = 219 mm.

t = pipe wall thickness = 8.18 mm.

$$I = 3010 \text{ cm}^2.$$

$$M_i = \sqrt{M_{\text{horiz}}^2 + M_{\text{vert}}^2} = 5.34 \text{ MJ.}$$

S_m = allowable stress intensity value,

= 138 MPa at 121°C (used value for A-106, GR B which has same ultimate tensile stress as A53, GR B).

From left side of above equation:

$$0.5 \left(\frac{689.5 \times 219}{2 \times 8.18} \right) + 1.0 \left(\frac{8.18 \times 10^{-3}}{2 \times 30 \times 10^{-6}} \right) 5.34$$

$$= 4.62 \text{ MPa} + 134.6 \text{ MPa}$$

$$= 139 \text{ MPa}$$

This value is less than:

$$1.5 S_m = 206 \text{ MPa}$$

Therefore, seismic loads are acceptable even when no credit is taken for the stanchion support.

Fig. 3.8-47 SAP4 computer code model of the secondary coolant piping.

The secondary coolant piping was modeled as shown in Fig. 3.8-47 using the pipe elements in the SAP4 computer code. The pipe is supported at two points as shown. The center support is assumed to be flexible and the pipe is assumed to have a fixed end at node 14. Additional lumped masses were employed to account for the added weight of the valves and orifices.

The piping was subjected to the LLL SSE spectra applied simultaneously in the east-west, north-south, and vertical directions. The vertical excitation used was $2/3$ of the horizontal. The effects of deadweight loading and internal pressure on the stresses were also included. Table 3.8-14 lists the two lowest fundamental frequencies.

From the computer analysis it was found that the maximum stresses occurred at the equipment pit wall. The maximum stress resultants are shown and labeled in

Table 3.8-14 Resonant frequencies as calculated by piping formula and computer.

Frequency No.	Piping system			
	Secondary		Primary	
	Hand calculation, Hz	SAP4, Hz	Hand calculation, Hz	SAP4, Hz
1	5.45	5.44	7.8	6.94
2	20.7	18.9	8.8	10.0

Table 3.8-15 Maximum stress in primary and secondary piping by hand and computer calculations.

	SAP4, MPa	Hand calculation, MPa	1.5 S _m MPa
Secondary	93	139	210
Primary	34	20	100

Fig. 3.8-48. The maximum principal and shear stresses caused by these stress resultants do not exceed 93 MPa and 47 MPa respectively. These stresses are less than 1.5 S_m where S_m is the allowable stress intensity value (see Table 3.8-15).

3.8.4.6.2 Primary Coolant

Piping – The pipe run from the suction side of the primary coolant pump to the equipment pit wall was modeled as shown in Fig. 3.8-49. From Ref. 26 the approximate natural frequencies for these configurations can be determined by:

$$f_n = \frac{\alpha}{L^2} \frac{EI}{w_{eff}}$$

where

L = total length m

E = 70 GPA

I = 1170 cm⁴ (6 in. Sch. 40)

α = factor depending on ratio of pipe legs (Fig. 9.13 of Ref. 26)

w_{eff} = effective unit force (kg/m). This must be adjusted to account for the force of the valve and pipe lengths at the bend in Fig. 3.8-45. In Ref. 5, the concept of effective force is an approach by which

Fig. 3.8-48 Maximum stress resultants in the secondary coolant piping at the equipment pit wall.

distributed force is treated as an equivalent concentrated force. Reversing the process, an effective unit force can be found for the above piping configuration.

Primary coolant piping models

Fig. 3.8-49 Model of primary coolant piping from suction side of the pump to the equipment pit wall.

$$w_{\text{effective}} = \frac{16 W}{5 L} + w_{\text{pipe}}$$

where

W = force applied at bend

= 1.13 kN

L = 6.5 m

$w_{\text{pipe}} = 277.3 \text{ N/m}$

which results in

$$w_{\text{eff}} = 831.8 \text{ N/m.}$$

The analysis for east-west motion, which is the worst case in the horizontal direction, gives:

$$f_n = \frac{5.76}{6.5^2} \sqrt{\frac{68.9 \times 10^9 \times 11.7 \times 10^{-6}}{831.8}}$$

$$= 7.8 \text{ Hz.}$$

$$T_n = 0.13 \text{ s}$$

$$S_a = 18 \text{ m/s}^2 \text{ (Fig. 3.7-1)}$$

The maximum moment is calculated by assuming this configuration can be modeled as a uniformly loaded beam with a concentrated load added at the bend, and built-in ends. The maximum moment is at the wall and is:

$$M_{\text{horiz}} = S_a \left[\frac{w L^2}{12} + \frac{W}{L} (bL^2 - b^3 - 2ab^2) \right]$$

where

a = length from wall to concentrated load = 5.06 m

$b = L - a = 1.43 \text{ m}$

$w = 277.3 \text{ N/m}$

$$M_{\text{horiz}} = 18 (973 + 987)$$

$$= 3528 \text{ J.}$$

The calculations for vertical spectral acceleration are obtained in a manner similar to that above. Using Fig. 3.8-49

$L = 6.15 \text{ m}$

$\alpha = 5.83$

$f_n = 8.8 \text{ Hz}$

$$T_n = 0.11 \text{ s}$$

$$S_a = 12 \text{ m/s}^2$$

The maximum vertical moment at the wall is determined from the preceding equation where:

$$S_a = 10 + 12 = 22 \text{ m/s}^2$$

$$w = 277.3 \text{ N/m}$$

$$L = 6.15 \text{ m}$$

$$a = 5.06 \text{ m}$$

$$b = 1.09 \text{ m}$$

$$W = 1.13 \text{ kN}$$

$$M_{\text{vert}} = 2.2 (876 + 841)$$

$$= 3776 \text{ J}$$

The primary stress intensity can now be calculated per Ref. 23.

$$B_1 = 0.5$$

$$B_2 = 1.0$$

$$P = 410 \text{ kPa}$$

$$D_o = 84 \text{ mm}$$

$$t = 7.11 \text{ mm}$$

$$I = 1170 \text{ cm}^4$$

$$M_i = \sqrt{M_{\text{horiz}}^2 + M_{\text{vert}}^2} = 744 \text{ kJ}$$

$$B_1 \frac{PD_o}{2t} + B_2 \frac{D_o}{2t} M_i$$

$$= 1.22 \text{ MPa} + 18.5 \text{ MPa}$$

$$= 19.8 \text{ MPa}$$

This value must be equal to or less than $1.5 S_m$. There are no values of S_m given for aluminum pipe in Ref. 6. However, S_m can be taken as 1/3 the ultimate tensile strength.

$$S_m = \frac{1}{3} S_{\text{ult}}$$

$$= \frac{1}{3} (207 \text{ MPa})$$

$$= 68.9 \text{ MPa}$$

and

$$1.5 S_m = 103 \text{ MPa}$$

The preceding analysis shows the pipe run under consideration is acceptable.

By inspection, the remaining pipe runs can be seen to have shorter lengths between supports and hence smaller periods and spectral accelerations. The primary stress intensities are far below the allowable value and the primary coolant system is not expected to suffer seismic damage.

The primary coolant piping was modeled as shown in Fig. 3.8-50. Additional lumped masses were used to account for the added weight of the valves and orifices.

Fig. 3.8-50 SAP4 computer code model of the primary coolant piping.

Force kN	Torque kN·m
$T_x = 3.07$	$T_{xx} = 1.11$
$V_y = 7.24$	$M_{yy} = 0.43$
$V_z = 1.00$	$M_{zz} = 3.90$

Fig. 3.8-51 Maximum stress resultants in the primary coolant piping at node 11, Fig. 3.8-50.

The piping was analyzed using a procedure similar to that employed in the analysis of the secondary piping described above. The first two natural frequencies for the piping are given in Table 3.8-14.

The maximum stress on the primary piping was found to occur at nodal point 11. A free body diagram with the stress resultants shown and labeled is given on Fig. 3.8-51. From this diagram the maximum principal and shear stresses which may occur are conservatively estimated to be 33.8 MPa and 19.3 MPa, respectively. Since $S_m = 69$ MPa, no problem should arise relating to the failure of this piping.

3.9 Mechanical Systems and Components

3.9.1 Dynamic Systems Analysis and Testing

Dynamic analysis has been conducted only for seismic loading on Category I items, see Sec. 3.8. The magnitude of other dynamic effects as well as their occurrence is small with the LPTR design.

3.9.1.1 Piping Systems — The coolant systems have no quick closing isolation valves to induce piping vibrations. All valves are manually operated with the exception of V-97, a temperature controlled flow valve on the secondary coolant system, Fig. 3.2-1. This valve does cause vibration of the piping downstream from it upon sudden opening and closing. Figure 3.8-45 shows the installation. The vibration is the result of the change in direction of the coolant stream at the downstream elbow. The bellows just beyond the valve V-66 allows this leg to vibrate because of its nonrigid support character. The rapid opening and closing of V-97 occurs exclusively upon reactor start-up and disappears after two or three cycles. It is not considered a safety problem because of the low total number of occurrences and the low stresses involved. Pump starts and stops cause negligible vibration of any of the piping systems.

3.9.1.2 Reactor Internals — Flow induced vibration of reactor internals is negligible. The coolant velocities range from a high of 0.82 m/s through the fuel elements to essentially zero for the rest of the reactor pool tank volume.

3.9.1.3 Heat Exchanger — The present heat exchanger was placed in service in early June 1967. In mid-December 1968, a baffle failed. This flow directive baffle pivoted so as to block the primary flow discharge port. The baffle was originally fillet welded to a cross baffle along one 2.7-mm-thick $\times$ 380-mm-wide edge. The other 380-mm-wide edge rests against the top tube sheet. The baffle was essentially a cantilever with flow normal to its surface (380 mm wide $\times$ 660 mm long). The failure apparently was the result of poor design and fatigue caused by flutter of the baffle.

The baffle was returned to its original position and supported by adding five 12.7 mm diameter rods installed perpendicular to the baffle surface. No further problems have been experienced since the modification.

3.9.1.4 LOCA — The loss of coolant accident imposes negligible dynamic loads on the reactor or its systems. The reactor coolant system is open to the atmosphere and the maximum flow rate from a pipe break would be a function of the static head. A double-ended pipe break at the equipment pit wall would result in an initial flow rate of 0.24 m³/s and a velocity of 7.0 m/s. This would result from a break of either of the 8 in. Sch. 40 primary coolant lines that enter and leave the pit. A break downstream of the primary coolant pump would have a lower flow rate because of the flow restrictions in the line from the equipment pit wall to pump suction. The maximum force exerted by the discharging fluid is 3.2 kN. The piping and supports will withstand this loading.

3.9.2 ASME Code Classes 2 and 3 Components

At the time the LPTR was built, the ASME Code Classes 2 and 3 were not in existence. The piping generally met the requirements of the Code for Pressure Piping, ASA B 31.1 - 1955. The tanks were specified to be built in accordance with Sec. VIII, ASME Code, 1952 Edition, but the Foster-Wheeler drawings stated that a Code Stamp was not required. The heat exchanger is designed to Sec. VIII and Code Cases 1270 N and 1273 N and carried a Code Stamp.

Design temperatures and pressures are given in Table 5.2-1.

Refer to Ch. 5 for information on reactor coolant system components.

The components produced and installed under the foregoing requirements correspond most nearly to the present "Class 3" described in Subsection ND of Sec. III, ASME Code.

3.9.3 Components not Covered by ASME Code

These include the fuel elements, the control rods, and the control rod drives. The control rods are a gravity drop system and are not adversely affected by seismic loads. Their insertability is assured by the system design, see Sec. 4.2.3.

The mechanical design of the fuel elements is covered in Sec. 4.2.1.

3.10 Seismic Design of Category I Instrumentation and Electrical Equipment

3.10.1 Seismic Design Criteria

Since the control rods fall into the core by the force of gravity, and since no energy sources are required for the safe shutdown of the core, seismic concerns relate primarily to major components staying in place. Conservative calculations (Sec. 15.2) show that core destruction is not possible during a loss of coolant accident; therefore, the failure of ECCS system (Sec. 6.3), does not amplify the accident. Thus, the design basis for the control room and the containment building was the Uniform Building Code 1955 edition.

Major components, such as the emergency generator and filter scrubber system, are firmly held in place on concrete foundations by straps or bolts.

Complete electrical failure leaves the building in a sealed condition and power is not required for the filter/scrubber system to operate.

The observation made on components and structures in the San Fernando Earthquake in Southern California in 1970 were applied to many areas at LLL to insure that simple movement of components did not destroy process piping or wires to the component.

3.10.2 Seismic Analyses, Testing Procedures and Restraint Measures

Not applicable.

3.11 Environmental Design of Mechanical and Electrical Equipment

Mechanical and Electrical Equipment in the containment structure is subject to different environmental conditions depending on whether it is under normal, test, or accident conditions. For the LPTR, normal and test conditions are identical to the normal

a ambient environment. Only accident conditions present a hostile environment. The hostile environment can be identified as either fire or flooding.

3.11.1 Fire

Mechanical and electrical equipment can be damaged by fire. Electrical power lines are not put in the same conduits as signal lines; control circuits may be in the same conduits as low level power. The National Electrical Code is the primary standard of practice for installed wiring within the containment. Power lines are protected by circuit breakers to ensure that they are turned off during an overload.

3.11.2 Flooding

The only area of flooding that exists at the LPTR is the large equipment pit where the primary and secondary coolant systems have many components. Important switches, filters and blowers are located above the flood level that would be created should the primary coolant system drain into the pit. Flooding from sources outside the containment is considered impossible because of the reactor's location high in the eastern portion of the valley, Sec. 2.4.

The equipment pit is monitored by a high level sump alarm located in the 0.17 m³ sump.

REFERENCES

1. 10 Code of Federal Regulations Part 50, Appendix A, March 16, 1971.
2. Design Analysis Manual, Livermore Pool Type Reactor (Foster-Wheeler Corporation, New York, N. Y., 1958).
3. Final Report - Wind Forces on Structures, Task Committee on Wind Forces, Committee on Loads and Structures, Structural Division, American Society of Civil Engineers. Published in Trans. ASCE, Paper No. 3269 (1961).
4. L. E. Brownell and E. H. Young, Process Equipment Design (Wiley and Sons, Inc., New York, N. Y., 1959) pp. 157-61.
5. T. B. Smith and V. A. Mirabella, Characteristics of California Tornados, Lawrence Livermore Laboratory, Rept. UCRL-13533 (1972).
6. P. L. Doan, Nucl. Safety 11 (4), 296 (1970).
7. "Section V, Personnel and Equipment Shields," in Mechanical Engineering Safety Manual, Lawrence Livermore Laboratory, Manual M-012, Rev. 1 (1970).
8. C. V. Moore, "The Design of Barricades for Hazardous Pressure Systems," in Nuclear Engineering and Design (North Holland Publishing Company, Amsterdam, 1967) pp. 81-89.
9. John A. Blume and Associates, Investigation of Faulting at the Lawrence Livermore Laboratory, Lawrence Livermore Laboratory, Rept. UCRL-13568 (1972).
10. Recommendations for Shape of Earthquake Response Spectra, prepared by John A. Blume and Associates for the Directorate of Licensing, USAEC Wash-1254 UC-11 (1973).
11. D. L. Bernreuter and F. J. Tokarz, Design Basis Earthquake for the Lawrence Livermore Laboratory Site, Lawrence Livermore Laboratory, Rept. UCRL-51193 (1972).
12. R. J. Dietrich, Shannon and Wilson, Inc., Burlingame, California, private communication November 24, 1971.
13. S. Ghosh and E. Wilson, Dynamic Stress Analysis of Axisymmetric Structures Under Arbitrary Loading, University of California, Earthquake Engineering Research Center, Rept. EERC 72-1 (1972).
14. L. W. Christensen, Lawrence Livermore Laboratory, Internal Document ENW 72-13 (1972). Readers outside the Laboratory who desire further information on LLL internal documents should address their inquiries to the Technical Information Department, Lawrence Livermore Laboratory, Livermore, California 94550.
15. E. L. Wilson and H. H. Dovey, Static and Earthquake Analysis of Three-Dimensional Frame and Shear Wall Buildings, University of California, Earthquake Engineering Research Center, Berkeley, Rept. EERC 72-13 (1972).
16. M. Stephensen, Lawrence Livermore Laboratory, Internal Document EEN 72-168 (1972).

17. W. P. Doherty and E. L. Wilson, Analysis of Two-Dimensional Frame Structures, University of California, Berkeley, Computer Programming Series Notes (1967).
18. R. C. Murray and F. J. Tokarz, Seismic Analysis of the Livermore Pool Type Reactor, Lawrence Livermore Laboratory, Rept. UCRL-51507 (1973).
19. L. Jacobsen, J. Appl. Mech., 5, 1 (1938).
20. L. Jacobsen, Trans. ASCE, 104, 402 (1939).
21. Seismic Design for Nuclear Power Plants, R. J. Hansen, Ed. (M.I.T. Press, Cambridge, 1970) p. 261.
22. Nuclear Power Piping, USA Standard B 31.7 (American Society of Mechanical Engineers, 1969).
23. Building Code Requirements for Reinforced Concrete (ACI 318-63) (American Concrete Institute, Detroit, Michigan, 1963).
24. T. von Karman and H. Tsien, J. Aeronaut. Sci., 6, 633 (1939).
25. "Part 1, Recommended Practice Design of Concrete Pavements," in ACI Manual of Concrete Practice (American Concrete Institute, Detroit, Michigan, 1968).
26. Design of Piping Systems, M. W. Kellogg Company, New York, N. Y. (Wiley and Sons, New York, 1965) pp. 257-267.

CHAPTER 4.0 REACTOR

4.1 Summary Description

The LPTR is a light water moderated and cooled reactor with sufficient simplicity to be flexible and easy to maintain. It utilizes an open tank as its primary vessel.

The coolant system, consisting of a single loop and heat exchanger, is operated typically in either of two modes, forced convection at a constant flow rate with coolant flowing downward through the core or natural convection at a reduced power level.

The reactor core consists of an array of MTR plate type fuel elements selectively placed in the grid plate to maximize the neutron flux in the core center. The core is reflected on top and bottom with water, on two sides with beryllium and on two sides with lead and graphite.

Nuclear instruments located below the core monitor neutron flux from source levels up thru full power levels. They are divided into safety instrumentation and control instrumentation. Sufficient safety instrumentation is provided to insure adequate protection against fuel damage. These instruments are described in detail in Ch. 7.0.

For reactors of this type such items as radiation damage to the reactor vessel walls, xenon instabilities, auxiliary coolant systems, cold loop start-ups, and primary coolant pressure controls are not problems.

A summary description of the reactor is provided in Table 4.1-1. The overall design of the reactor including tank, tank penetrations, fuel elements, core structure and parameters, reactivity controls, and coolant control, is described in detail in the following sections. Irradiation facilities and flux values are described in Sec. 10.2 and 10.3.

Table 4.1-1. Livermore pool type research reactor summary description as of February 1974.

<u>General</u>	
Reactor type	Pool type, fully enriched (93%) uranium, light water moderated and cooled, beryllium reflected.
Nominal reactor power	3 MW thermal
Peak thermal flux	$7.5 \times 10^{13} \text{ n/cm}^2 \cdot \text{s}$
Purpose	Research in neutron physics, isotope production, radiography, activation analysis.

Table 4.1-1 (Continued)

Owner	United States Atomic Energy Commission.
Operator	University of California
Designer and builder	Foster Wheeler Corp.
Present status	3 MW operation February 1967 Major repairs November 1970

Reactor physics

Neutrons: energy Thermal, 50 μ s
lifetime

Core parameters $\eta = 2.07$, $p\epsilon \approx 1.0$
 $f = 0.75$

$k_{\infty} = 1.537$, $k_{eff} = 1.012$
 $L^2 = 3.42 \text{ cm}^2$, $\tau = 56.5 \text{ cm}^2$

Neutron flux Thermal av $1.8 \times 10^{13} \text{ n/cm}^2 \cdot \text{s}$
Thermal max $7.5 \times 10^{13} \text{ n/cm}^2 \cdot \text{s}$
Fast av $6.7 \times 10^{12} \text{ n/cm}^2 \cdot \text{s}$
Fast max $2.8 \times 10^{13} \text{ n/cm}^2 \cdot \text{s}$

Reactivity balance Max built in (cold clean) 6% to compensate for:

Temperature 0.2%
Xe build up 3.24%
Xe override 1.3%
Sm 0.71%
Burnup 0.56%

Core

Shape and dimensions Parallelepiped $0.38 \times 0.53 \text{ m}$, 0.61 m high

No. of channels and sub-assemblies Grid plate with 5×7 position. Normal operating core of 20 standard fuel elements, 5 special fuel elements, 9 beryllium reflector elements, and the central position occupied by a round dummy element.

Lattice Rectangular
Pitch $77.09 \times 81 \text{ mm}$.

Critical mass 3.1 kg ^{235}U (standard core).

Core loading at rated power 3.7 kg ^{235}U at 3 MW.

Average specific power in fuel 819 kW/kg ^{235}U .

Av power density in core fuel 25 MW/m^3 nominal.

Fuel burnup Core average 15%

Fuel loading and unloading Performed only after shut-down. Manual, with conventional swimming pool handling tools. Lead coffin for spent element.

Table 4.1-1 (Continued)

Irradiated fuel storage	1 m above top of core in the tank is a circumferential rack with room for 40 elements. Floor storage pit for 30 elements for longer cooling period.
Moderator	Light water, 26.5 m ³ in the tank.
Blanket gas	None, open to atmosphere

Fuel element

Form and composition	MTR-type, curved plates; meat dimensions 0.5 × 63.5 × 597 mm, plate overall 1.5 × 76.2 × 610 mm, enrichment 93%, alloyed with 85% aluminum.
Cladding	0.5 mm aluminum, cold rolled.
Subassemblies	18 plates forming element 76 × 76 × 876 mm.

Core heat transfer

Heat transfer area	Per standard element	1.39 m ²
	Per control element	0.7 m ²
	Total	31.4 m ²
Heat flux at 3 MW	Av 100 kW/m ²	
	Max 315 kW/m ²	
Fuel element temperatures	Max fuel 100°C nominal (67 dm ³ /s)	
	Max cladding 110°C (54 dm ³ /s)	
	Film temperature drop 18°C nominal	
Heat transfer coefficient	5.49 kW/m ² ·K	
Coolant flow area and velocity	Per standard fuel element	35.3 cm ²
	Per control rod element	17.7 cm ²
	Total	53.0 cm ²
	Av velocity	0.8 m/s
Coolant mass flow rate	67 kg/s	
Coolant pressures and temperatures, nominal	Atmospheric	
	Inlet 29°C Outlet 40°C	
Hot channel factors	Overall 1.62 (position C-3).	
Coolant outlet temperatures, max	73.3°C at 53 dm ³ /s.	
Shut-down heat removal	Natural convection	

Control

Regulating and shim-safety rods	4 shim-safety rods, B ₄ C in SS, active length 623 mm, total 728 mm. Oval cross section 57 × 22.2 mm. 1 regulating rod, stainless steel 728 mm total and active length, cross section same as shim-safety rods. Total worth of shim-safety rods 13.2% Δk/k Worth of regulating rod 0.6% Δk/k Speed of shim-safety rods 1.3 mm/s = 3.09% Δk/k/min max. Speed of regulating rod 5.1 mm/s = 0.45% Δk/k/min max.
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Table 4.1-1 (Continued)

Scram time and mechanism	750 ms slow, 650 ms fast Magnetic release, gravity fall
Sensitivity of automatic control	Approx. 1% of full power.
Temperature coefficient	$11 \times 10^{-3} \% \Delta k/k/^{\circ}\text{C}$ (clean)
Burnable poison	None
Other control, safety and shut-down provisions	None

Reactor vessel and overall dimensions

Form, material and dimensions	Cylindrical aluminum tank, 2.0 m diam., 8.1 m high, 9.53 mm wall thickness.
Thermal shields	50 mm lead exterior circumference of tank from 0.15 to 2 m.
Working, design and test pressure	Atmospheric
Reactor shielding	Octagonal base, approx. 5.8 m diam., east side with 2.1 m elongation, 7.75 m above floor level.

Reflector and shielding

Reflector	Beryllium pieces in fuel element form $75 \times 75 \times 630$ mm.
Radiation levels	Design: 0.1 mrem/h at working floor level; 1.0 mrem/h near beam tubes, over equipment pit and fuel element storage pit; 10 mrem/h at balcony and top of pool.
Shielding	Sides: 0.15 m lead (in front of thermal columns only), 0.76 m light water, 9.5 mm tank wall, 63 mm lead, 1.83 m magnetic concrete Bottom: ordinary concrete Top: 6.1 m light water

4.2 Mechanical Design

4.2.1 Fuel

4.2.1.1. Design — The LPTR uses a plate type fuel element very similar to that of the Materials Test Reactor. The mechanical design bases for these elements are: reliable containment of fission products, good structural stability, good corrosion resistance, and negligible radiation damage. The fact that MTR plate type fuel elements have been in use since 1952 with many megawatt years of successful operating experience would indicate that the design objectives have been satisfied. The basic plate-type fuel element has been operated at coolant conditions and power densities far more severe than those of the LPTR. Table 4.2-1 describes the physical

Table 4.2-1. Fuel-element description and operating conditions in high-power-density research and test reactors.

Characteristic or condition	Reactor				
	NBSR ^a	ORR ^b	MTR Mark XI ^c	GTR ^d	LPTR mark IV
Core inlet water temperature °C	38	49	39	46	30
Water velocity in fuel-element channels m/s	3.7	9	10	3.0	0.82
Core pressure drop kPa	83	172	280	23	7.6
Nominal water channel thickness mm	2.95	2.64	2.98	2.98	2.98
Fuel plate thickness mm	1.3	1.3	1.3	1.5	1.5
Fuel meat thickness mm	0.5	0.5	0.5	0.5	0.5
U in fuel alloy, wt%	18	18	20	18	20
Width of fuel plates between side plates mm	61.4	63.8	63.6	66.6	63.6
Maximum heat flux MW/m ²	0.85, equilibrium	2.4	2.7	0.66	0.32
Hot spot surface temperature °C	115	115	145	97	100
Average ²³⁵ U burnup %	40	35-40	25-30	4.8	25

^aNBSR - National Bureau of Standards Reactor.

^bORR - Oak Ridge Reactor

^cMTR - Materials Testing Reactor.

^dGTR - Ground Test Reactor.

characteristics and operating conditions of the fuel elements used in the LPTR and several other research and test reactors. The LPTR Mark IV fuel elements are shown in Fig. 4.2-1. Each fuel element, except the special elements, contains 18 fuel plates with a nominal fuel loading of 190 g of 93% enriched ²³⁵U. The special fuel elements have a flux wire tube integral with them and are fabricated with a slot down the center to accommodate a control rod. This requires an element with nine less fuel plates and, consequently, it has half the fuel weight of the standard element.

All fuel elements are equipped with cast 356 aluminum alloy end fittings. The end fittings secure the elements in position in the grid plate. An alignment pin in the grid

Note: All dimensions in inches. 1 inch = 25.4 mm.

Fig. 4.2-1. Standard and special fuel elements, longitudinal and cross sections.

plate. An alignment pin in the grid plate mates with the blind hole in the end fitting to ensure proper orientation of the fuel element in the core.

The fuel plates are of sandwich construction. The meat or core section is an alloy of aluminum and uranium. The structure consists of primary compounds of UAl_3 surrounded by a matrix of eutectic between aluminum and UAl_3 . The core is centered in a picture frame of 1100 aluminum alloy and is clad on both sides with sheets of the same material. The fuel plates are hot rolled to metallurgically bond the core and clad to ensure good thermal conductivity and fission product retention. The fuel plates are currently roll swaged into a pair of 6061-T6 aluminum alloy side plates. The joint between each fuel plate and side plate is required to have a strength of not less than 35 kN/m of joint. Samples are pull tested to verify this requirement. The present LPTR fuel element specification essentially incorporates the American National Standards Institute specification, "Quality Control for Plate Uranium-Aluminum Fuel Elements, ANSI-N 8.1."

4.2.1.2 Dimensions — A cross-sectional view of a fuel element with dimensional specifications is given in Fig. 4.2-1. The coolant gap between fuel plates is 2.97 mm (0.117 in.). The side plate thickness is 4.77 mm (0.188 in.), while the fuel-bearing plates are 1.52 mm (0.060 in.) thick. The center-to-center dimension in the grid for positioning the fuel element is 77.09 mm (3.035 in.), while the over-all assembly width is 76.1 mm (2.996 in.). This difference allows for a coolant gap between elements in this direction of 1 mm (0.039 in.). The spacing in the other direction is 2.79 mm (0.110 in.), or approximately the same as the coolant gap between adjacent fuel plates in the assembly.

4.2.1.3 Irradiation Experience — MTR type fuel elements have been in use since 1952. Known failures have been the result of (1) coolant blockage, (2) an inadvertent design change that caused lateral pressure difference to buckle the fuel plates, and (3) inadequate quality control during fabrication. Item 1 was corrected by more stringent administrative controls. Item 2 was corrected in 1954 by a redesign and item 3 by tighter quality control.

Measurements made at MTR indicate there are only slight dimensional changes accompanied by small decreases in density in the elements after long-term reactor exposure. With respect to mechanical and physical properties, it has been noted that the yield strength and hardness of an 18% uranium-aluminum alloy was doubled after reactor exposure and there was a marked decrease in ductility. Reductions of 20% to 30% in thermal conductivity have been recorded.^{1,2} However, under the operating conditions encountered in the LPTR the changes described above have not resulted in any fuel element failures or loss of operating efficiency.

As for fission gas retention, measurements were made on fuel plates by collecting the radioactive krypton (10 yr half life) as a function of annealing temperature. At temperatures up to 550°C only 0.2% of the krypton present was released and most of this in the first 2 h of a 20 h anneal. Complete melting was required to remove all krypton present.³

4.2.2 Reactor Pool Tank Internals

The reactor pool tank internals consist of the irradiation facilities, the reactor core, the grid plate and its support, and the control rods. The irradiation facilities will be covered here, the others are described in Secs. 4.2.1, 4.2.3, and 3.8.1. Certain aspects of the tank internals relating to coolant boundary are also presented in Sec. 5.4.

4.2.2.1 Beam Port Description — A plan view of the irradiation facilities is shown in Fig. 4.2-2. Six horizontal beam ports are provided, five of these with diameters between 100 to 150 mm at the core face, and one large beam port having a

Fig. 4.2-2. Plan view of irradiation facilities.

diameter over 300 mm. A 6-in. beam port B-1 appears in Fig. 4.2-3. Thermal columns cover two core faces, requiring all beam ports to be positioned within the space provided by the two remaining core faces. Consequently, the beam port center lines vary from between 1.2 and 1.5 m above the tank bottom. Table 4.2-2 presents beam port design parameters.

4.2.2.1.1 Twelve Inch Beam Port — The 12 in. beam tube, B-6, is located on the north face of the reactor. See Fig. 4.2-4. It consists of a permanently placed stepped sleeve that penetrates the concrete shield and the reactor tank, and a removable stepped liner that has a window facing the reactor core. The beam port sleeve that passes through the shield is made of aluminum alloy 1100-0.

At the interior end, the liner is constructed of an aluminum alloy 1100-F cylinder with a 320-mm i.d. A 9.5-mm plate of 1100-0 aluminum alloy seals off the core end of the cylinder from water in the reactor tank. Under the full hydrostatic head of water in

Table 4.2-2. Beam port design.

Beam port	In tank liner	End closure	Design pressure		Hydrostatic test pressure	
			kPa	(psi)	kPa	(psi)
B-1	6 in. Sch 80 pipe, 6063-T6 aluminum	6.3 mm thick 1100-0 aluminum	70	(10)	200	(30)
B-2	6 in. Sch 80 pipe, 6063-T6 aluminum	6.3 mm thick 1100-0 aluminum	70	(10)	200	(30)
B-3	5 in. Sch 80 pipe, 6063-T6 aluminum	6.3 mm thick 1100-0 aluminum	70	(10)	200	(30)
B-4	5 in. Sch 80 pipe, 6063-T6 aluminum	6.3 mm thick 1100-0 aluminum	70	(10)	200	(30)
B-5	6 in. Sch 80 pipe, 6063-T6 aluminum	6.3 mm thick 1100-0 aluminum	70	(10)	200	(30)
B-6	338 mm o.d. X 8.64 mm wall tube, 1100 F aluminum	9.5 mm thick 1100-0 aluminum	70	(10)	200	(30)
Instrument thimbles	4 in. Sch 40 pipe 6063-T6 aluminum	3.2 mm thick 5052-0 aluminum	77	(11)	Unknown	

the tank this plate will be stressed to a maximum of 25.4 MPa, less than 30% of the ultimate strength of the material.

A clearance of 1.5 mm is maintained between the liner end plate and the face of the fuel elements when the end closure plate is welded.

The shielding plug for the 12 in. beam port weighs approximately 680 kg. The plug shell is constructed entirely of Type 304 stainless steel for its corrosion resistant properties. The interior end plate is backed up with a 6.3 mm boral plate to minimize activation of the dense shielding concrete in the plug shell.

The nosepiece for the 12 in. beam port is made of 9.5 mm aluminum alloy 1100-F plate rolled to a 300 mm o.d. cylinder. The end plate is a 9.5 mm aluminum alloy 5052-F plate field welded into place when the final position of the beam tube and plug has been determined. The nosepiece is approximately 0.6 m long.

Design of the nosepiece is to withstand an internal hydrostatic test pressure of 206 kPa. Under this condition of loading the maximum stress at the junction in the cylindrical section of the nosepiece is 32.9 MPa and the maximum stress in the flat end plate is

Fig. 4.2-3. Irradiation facilities, south side, before pouring shield concrete.

Fig. 4.2-4. Twelve inch beam port (B-6).

29.5 MPa. This results in minor yielding during the initial load cycle but presents no danger. The normal nosepiece coolant pressure of 140 kPa causes stresses below the yield strength.

Corrosion products between the heavy concrete biological shield and the beam port sleeve (aluminum 1100) have locked the 12 in. plug in place. Section 5.2.3.2 discusses the corrosion in more detail. Since the 12 in. beam plug is no longer removable, its sudden removal is no longer considered in maximum water loss rates from the pool.

The nosepiece is fabricated of 9.5 mm aluminum alloy 1100-F plate rolled to 300 mm o.d. cylinder. The end plate is a 9.5 mm aluminum alloy 5052-F plate field welded into place when the final position of the beam tube and plug is determined. The nosepiece is approximately 0.6 m long.

4.2.2.1.2 Six Inch Beam Port — The 6 in. beam ports B-1, B-2, and B-5 are identical in construction with only minor variations in overall length due to the configuration of the reactor core and variations in elevation caused by the physical limitations of space available at the core face. Each of the 6 in. beam ports consists of a permanently placed sleeve that penetrates the concrete shield and reactor tank wall extending to a face of the reactor core. This differs from the 12 in. beam port which has a removable liner extending to the core face. The sleeve is built in four sections:

- 1) a carbon steel vestibule box
- 2) a Type 304 stainless steel cylindrical exterior section
- 3) a Type 304 stainless steel shutter slot
- 4) an aluminum alloy cylindrical interior section.

The vestibule box, exterior sleeve section, and shutter slot are welded together to form a subassembly unit. The aluminum alloy cylindrical section is bolted to the interior end plate of the shutter slot to form the completed beam tube.

The exterior cylindrical section of the sleeve fabricated from stainless steel has an inside diameter of 0.217 m. The shutter slot is also stainless steel since the plug rests on the bottom plate of the shot and is inaccessible for maintenance. The interior cylindrical section of the beam tube is fabricated from aluminum alloy 6063-T6, 6 in. Sch. 80 pipe. The length allows a clearance of 1.52 mm (0.06 in.) between the end plate and fuel elements in the reactor core. The end plate is 6.3 mm aluminum alloy 1100-F. This portion of the beam tube is made of aluminum alloy for its favorable low neutron absorption cross section. The end plate is TIG welded to the pipe with a full penetration groove weld. The joint is hydrostatically tested to 206 kPa (three times maximum design pressure) and dye penetrant tested.

A flanged nozzle in the reactor pool tank wall permits the in-tank portion of the beam tube to penetrate the tank. The in-tank sleeve is connected to the reactor pool tank nozzle in the following manner:

A 9.5 mm thick by 19 mm wide annular ring of 5052 aluminum was slipped over the in-tank sleeve. The entire sleeve was then aligned with respect to the reactor core. At this time the annular ring was fillet welded to the in-tank sleeve and the pool tank nozzle flange. Next the outer end of beam port sleeve was welded to the reactor shield face plate. This rigidly located the beam tube. After a hydrostatic test and dye penetrant check of the welds, the shield concrete was poured into the space between the shield face plate and pool tank outside diameter.

The beam port sleeve is thus firmly supported over its total length except at the end, where about 0.6 m is cantilevered into the pool tank.

The foregoing description essentially completes what could be considered the beam port's reactor pool tank internals. The remaining component, the shutter, is not in contact with the reactor coolant but will be described for the sake of completeness.

The shutter is a steel encased shield which can be lowered into the beam port when the facility is not being used. It affords a shielding length of approximately 0.4 m. The width of the shutter is equal to or slightly greater than the liner diameter. Use of the shutter reduces gamma flux when the port is not in use. The shutter is inclined at an angle of 12° to the vertical and is approximately 2.7 m long. It hangs from a jackscrew shaft supported by a bearing on the upper plate of the shutter casing. This casing is permanently placed in the reactor concrete shield extending from balcony to the beam port where it is field welded to the shutter slot. The shutter moves within the shutter casing and is guided by roller wheels in grooves in the casing.

The upper section of the shutter shell is fabricated of carbon steel plate and filled with dense concrete. The lower section of the shutter shell is a Type 304 stainless steel box filled with lead. The shutter weighs approximately 1.2 Mg. A reversible gear motor moves the shutter from full closed to full open position in 59 s. Limit switches prevent the shutter from hitting either the top or the bottom of the shutter slot.

4.2.2.1.3 Four Inch Beam Ports — The 4 in. beam ports B-3 and B-4 both have the same construction features as the 6 in. beam ports, the difference being solely in diameters of the liner and width of the shutter. Diameter of the beam tube liner at the exterior end is 166 mm. Materials for the liner are the same as for the 6 in. beam tube. The end plate is 6.3 mm aluminum alloy 1100-F plate welded and tested in the same manner as the 6 in. liner.

Shutter construction is a scaled-down version of that used for the 150 mm beam tube. Motor drive and control system are the same as for the 150 mm beam port shutter.

4.2.2.1.4 Instrument Thimbles — The instrument thimbles, while not irradiation facilities, are very similar to the beam ports. Consequently, they will be described in this section. See Figs. 4.2-2 and 4.2-5 for their relative locations. The five thimbles are all of the same design and construction. They are fabricated from 4 in. Sch. 40, 6063-T6 aluminum pipe and are 2.67 m long. The in-tank end closure is a 3.18 mm thick plate of 5052-0 aluminum welded over the end of the pipe. The exterior ends of the south side thimbles are flanged and bolted to a carbon steel vestibule box. The north side thimbles are modified to alleviate deformation caused by corrosion problems discussed in Sec. 5.2.3.2.

The in-tank portion of the thimble pipe starts 50 mm back from the core vertical centerline. In elevation, the thimble centerlines vary from 114.3 mm to 419 mm below the bottom of the core grid plate. The thimbles penetrate the 5.35 mm plates

Fig. 4.2-5. Instrument thimbles, south side, before pouring shield concrete.

on the north and south sides of the grid plate support structure to enter beneath the core, see Sec. 3.8.1.4. A flanged nozzle in the reactor pool tank wall permits the in-tank portion to enter the tank. The remainder of the thimble pipe passes through the shield concrete and terminates in the vestibule at the shield face.

4.2.2.2 Thermal Columns — Arrangements of the thermal columns are shown in Figs. 4.2-2 and 4.2-6. The two thermal columns are essentially identical in overall dimensions but the east thermal column is penetrated by a pneumatic rabbit tube R-2, the through tube, and two irradiation tubes S-2 and S-3. In addition, the west thermal column has a separate, removable lead gamma shield whereas the east thermal column has the lead shield integral with the inner enclosure.

The thermal columns are of the "canned" type consisting of an inner and an outer enclosure. The outer thermal column enclosure is welded to the reactor pool tank at its inner end and to the inner thermal column enclosure at the outer end. A 6.3 mm annular gap open to the tank is provided around the inner thermal column enclosure for water cooling by natural circulation.

The side walls, floor and roof of the outer enclosures (which are part of the pool tank) are 19 mm thick 5052-F aluminum plate. The corresponding portions of the inner enclosure (which is considered part of the pool tank internals) are fabricated from the same material and thickness for the east thermal column. The west thermal column uses 6061-T6 aluminum. The two enclosures are joined to one another with 25 mm diam plug welds made from the inner enclosure side. There are 60 such welds in the east thermal column (ETC) and 20 in the west thermal column (WTC). Since the outer enclosure is rigidly supported by the concrete shield, the plug welds prevent any movement of the inner enclosure.

The plug welds also resist the hydrostatic loads on the top, bottom and sides of the inner enclosure. Using the bottom as the worst case (highest load) one plug weld is almost sufficient to take the entire load. The bottom was analyzed as a rectangular plate with built-in edges. There was a uniform load over the entire surface with a concentrated load (plug weld) at the center opposing the uniform load. There is zero deflection at that point. The stress analysis indicates some yielding at the plug weld so additional welds are provided to preclude any stresses beyond the yield point.

A 6.3 mm slot weld at the outer end joins the two enclosures together. The slot weld is assumed to support the entire 14.6 Mg hydrostatic load on the front face of the inner enclosure. The shear and tensile stresses resulting from this load are respectively 4.6 MPa and 17 MPa. This gives safety factors greater than 10 based on typical properties of the weldment (170 MPa psi ultimate tensile strength and 83 MPa shear strength).

The front face of the ETC has a 150-mm thick lead shield bonded to it, see Figs. 4.2-7 and 4.2-8. The shield is employed to reduce the core gamma flux in the thermal column graphite for experimental reasons. The front face is a 31 mm thick plate of 1100-F aluminum. It has been milled to provide horizontal ribs 3.1 mm thick by 25 mm long. The lead fills the spaces between the ribs to form a 158 mm thick composite plate.

Fig. 4.2-6. East-west sectional view through the centerline of reactor looking north.

The heat generated in the lead is transferred to the reactor pool water at the front (core) side and to a 50 mm air gap at the back (graphite) side. Thermocouples located near the calculated adiabatic surface in the lead indicate a maximum lead temperature of about 193°C. This temperature is 135°C below the lead melting point ($T_{mp} = 327^{\circ}\text{C}$). The aluminum surface temperature in contact with the pool water is less than 93°C at an area opposite the maximum lead temperature.

It should be noted that the above thermal conditions exist in an area approximately 100 cm^2 centered 75 mm below the geometric core centerline. The lead temperatures in the volume beyond the square are 93°C or so. Operation to date with the above temperatures has caused no problem.

Fig. 4.2-7. Overall thermal column arrangement.

The front face of the WTC is a 19 mm thick plate of 1100-F aluminum. The lead gamma shield in this case is separate from the inner enclosure, see Figs. 4.2-9 and 4.2-10. It is made up of a box with 6.4 mm thick 1100-F aluminum front, back and top. The sides and bottom are 19 mm thick. The inside is filled with lead 150 mm thick. This lead shield is essentially identical in external appearance to that in the ETC, but with a 6.36 mm plate at the back of the lead. In the original construction, both thermal column lead gamma shields were the same as the ETC. In 1971, after 13 years of operation, the WTC was changed to its present configuration. The change was prompted by a slow rise in temperature in the WTC lead to 284°C caused by slow deterioration of the bond between the lead and the ribbed aluminum face. This increased the thermal resistance and temperature to the point where replacement was necessary.

The new lead shield has a large number of hook shaped rods welded to the inside of its front and back faces. These hooks tend to maintain the aluminum faces in contact with the lead. There are five thermocouples cast into the lead at the anticipated maximum

Fig. 4.2-8. Top view of lead gamma shield on west thermal column (WTC).

temperature region. Also a vent line is attached to the shield top. The vent prevents gas pressure build-up and allows the shield to be placed under a vacuum if desired. When the lead shield is placed in position in front of the inner enclosure it is cooled on all sides by the reactor pool water. The maximum measured temperature in the lead is now 137°C.

Fig. 4.2-9. Side view of lead gamma shield on west thermal column.

Fig. 4.2-10. West thermal column gamma shield.

The replacement lead shield rests on part of the support used for the original WTC inner enclosure. It bears directly on a pedestal fabricated from $76 \times 38 \times 9.5$ mm channel of 5052-F aluminum. The safety factor against yielding of this structure is 36. Obviously there is adequate vertical support. The horizontal support for the lead shield is different from the original design. Whereas before the lead was integral with the inner enclosure it is now free standing and requires horizontal restraint. This restraint has been provided by welding slotted rails to the front of the new inner enclosure. Corresponding rails on the lead gamma shield fit into the slots to hold the shield in place. The slotted rail weldments can withstand an acceleration of 4.6 m/s^2 (0.47 g) in the north-south direction and 12.3 m/s^2 (1.25 g) in the east-west direction without yielding. The design basis earthquake response spectra for the lead shield indicates an acceleration of 3.9 m/s^2 . For conservatism the analysis for the north-south direction assumed no friction at the bottom of the 1170 kg (2600 lb) lead shield where it rests on the pedestal. The slotted rail weldment is the most highly stressed component during an earthquake and, as can be seen, it is adequate.

4.2.2.3 Irradiation Ports (S tubes) - The four irradiation ports, S-1, S-2, S-3 and S-4 are shown in Figs. 4.2-2 and 4.2-6. The irradiation tubes are used for irradiating small samples that can be inserted and removed while the reactor is at power with a minimum amount of radiation exposure to the operator. The S-1 and S-4 tubes are for exposing samples in a region of high fast neutron flux and the S-2 and S-3 tubes are for exposing samples in a region of predominantly thermal neutron flux. Irradiation tubes are accessible from the balcony of the reactor 3.4 m above the operating floor. The S-1 tube is located on the southeast face of the reactor shield, penetrating the shield and extending to the south face of the reactor core. The S-2 and S-3 tubes are located on the east face of the reactor above the ETC cave, penetrating the shield and extending into sleeves in the thermal column just behind the lead gamma shield. The S-4 tube is located on the northwest face of the reactor shield, penetrating the shield and extending to the north face of the reactor core.

Each irradiation tube assembly consists of a carbon steel vestibule box set in the face of the reactor shield on the balcony level, a 5-in. Sch. 40 aluminum alloy 6063-T6 pipe sleeve set in the reactor concrete shield and a 4-in. Sch 40 aluminum alloy 6063-T6 pipe for the irradiation tube.

The sleeve is flanged at the exterior end for attachment to the vestibule box and is welded to the reactor pool tank at the interior end. A seal plate is provided which is welded to the sleeve and the tank to prevent leakage of water into the shield.

The irradiation tube is closed on the interior end with a 6.3 mm thick aluminum alloy 5052-F plate and open at the exterior end for insertion of samples. The exterior end has a flange of aluminum alloy 6061-T6 in the vestibule box. The annular gap between

the sleeve and tube is filled with epoxy. Formerly this gap was open to the pool water but leakage developed at a vent line fitting, allowing water to enter the concrete.

Each of the irradiation tubes is bent on a constant radius of curvature of ~3 m to prevent direct streaming of radiation from the reactor core. This curvature provides 1.4 m of water and 1.5 m of dense concrete for shielding between the vestibule and the reactor core. This is approximately equal to the shielding afforded by the biological shield at the operating floor level. To provide shielding for stray radiation from the water in the reactor tank, the irradiation tube will normally be filled with water. This affords approximately 1.2 m of water shielding.

Figure 4.2-11, taken during construction, shows irradiation tubes S-2 and S-3. The vent lines shown have been plugged and filled with epoxy to prevent leakage of water into the concrete.

4.2.2.4 Through Tube — The through tube runs tangential to the reactor core and extends from the north to the south face of the reactor shield. It passes through the east thermal column just behind the 150 mm lead gamma shield. Arrangement of the through tube is shown in Fig. 4.2-2. This facility permits irradiation of samples in a high-intermediate neutron flux from positions on the operating floor with access from two sides of the shield.

Vestibule boxes for the B-1 and B-6 beam ports were designed to accommodate the through tube which is located 0.43 m east of the beam tube centerlines. This places the through tube centerline 0.24 m from the core face. The through tube has a permanently placed sleeve on each side set in the concrete shield extending from the B-1 and B-6 beam port vestibule boxes to the reactor pool tank.

The permanent sleeve is a 5-in. Sch 40 pipe of 6063-T6 aluminum. It is welded to the reactor pool tank nozzle at its inner end and bolted to the vestibule box at its outer end. The inside diameter of this sleeve has been bored to 129 mm. The sleeve where the through tube penetrates the east thermal column is a 5-in. Sch. 40 pipe of 3003-H12 aluminum. It has a 9.52 mm thick reinforcement flange at each end which is welded to the sides of the inner thermal column enclosure to form a water seal.

The through tube proper is a 4-in. Sch 40 pipe of 6063-T6 aluminum 5.1 m long. About 1.5 m in from each end a seal ring 120 mm in diameter is welded to the outside of the pipe. This ring is part of the water seal assembly. The seal is made by a ring of carbon packing in the annulus between the through tube outside diameter and the sleeve inside diameter. The packing is expanded radially in the annulus by compressing it axially between the seal ring and a seal retaining tube. The seal retaining tube fits over the through tube and has a flanged outer end. Pressure is applied to the flanged

Fig. 4.2-11. Irradiation ports R-1, S-1, and S-2, before pouring shield concrete.

Fig. 4.2-12. Irradiation facilities, north side, before pouring shield concrete.

end by tightening 16 swivel pad clamps. This in turn compresses the carbon packing. If water leakage develops, the packing can be compressed further or additional packing added.

Figures 4.2-3 and 4.2-12 indicate the through tube installation before the shield concrete was poured. The vestibule boxes and permanent sleeves mentioned previously are clearly visible.

4.2.2.5 Pneumatic Rabbit Facilities - The slow and fast pneumatic rabbit systems are covered in Ch. 10. The portions of those systems that are located within the reactor pool tank will be described here.

Fig. 4.2-13. Reactor pool, east side.

The systems have three termination points in the reactor pool, see Figs. 4.2-13 and 4.2-14. The in-core facilities for both rabbit systems are nearly identical in design. They differ only in size of tubing and some minor details. The lower 0.6 m of the in-core facility consists of a pair of concentric tubes. The outer tube is closed at the bottom with an end plug which also aligns and supports the inner tube. The rabbit travels inside the inner tube. The motive force is a difference in gas pressure between the two ends of the rabbit. When sending the rabbit into the irradiation facility, the gas pressure on the rear of the rabbit is greater. The gas upstream of the rabbit is exhausted through slots in the bottom end plug into the annulus between the two tubes. This gas is eventually released to the atmosphere through appropriate filters. To remove the rabbit from the irradiation facility, the flow paths are merely reversed.

Fig. 4.2-14. Reactor pool, west side.

The higher pressure gas now emerges from the annulus and pushes the rabbit out of the inner tube.

The slow rabbit system has two terminations in the reactor pool tank. One is located in a well in the east thermal column and the other is in a beryllium reflector element in the reactor core. The design parameters are:

Design pressure - external	89 kPa
Operating pressure - external	89 kPa
Test pressure - external	101 kPa
Test pressure - internal	172 kPa
Design temperature	66°C
Outer tube - east thermal column	57 mm o.d. X 1.65 mm wall, 1100-F aluminum
Outer tube - in core	50 mm o.d. X 1.65 mm wall, 3003-H 14 aluminum
Inner tube - east thermal column	38.1 mm o.d. X 0.9 mm wall, 1100-F aluminum
Inner tube - in core	38.1 mm o.d. X 0.9 mm wall, 3003-H 14 aluminum

The fast rabbit system has one termination point in a beryllium reflector element. The design parameters are:

Design pressure - external	68 kPa
Design pressure - internal	68 kPa
Operating pressure - internal	410 kPa
Test pressure - internal	680 kPa
Design temperature	66°C
Outer tube	48 mm o.d. X 1.47 mm wall, 6061-T6 aluminum
Inner tube	35 mm o.d. X 1.47 mm wall, 6061-T6 aluminum

4.2.3. Reactivity Control System

4.2.3.1 Description - The reactivity control system consists of one regulating rod plus four shim-safety rods and their associated components. The shim-safety rods are used for coarse reactivity adjustments and are the primary neutron absorber. The regulating rod is used for fine control of reactivity when going to and at power.

4.2.3.1.1 Actuators - The control rods are inserted and withdrawn by actuators of the rack and pinion type. The actuators for the shim-safety rods are powered by two-phase reversible motors which are fan cooled and include a high speed, ball bearing gear reducer with a one stage worm and gear reducer to prevent back driving or drift under load, see Fig. 4.2-15. The regulating rod actuator is powered

Fig. 4.2-15 LPTR shim/safety rod actuator assembly.

by a syn motor with a single worm gear set. The actuator rack is supported at two locations by a guide roller and spur gear set. They are mounted on opposite sides of the rack near the lower end. The rack is guided above by a pair of rectangular stainless steel strips. The upward travel of the rack is physically restricted by a stop block attached to the actuator frame.

The actuators are designed for mounting in adjacent rows on 76 mm centers. The bottom of the rack is provided with a removable flange to facilitate attachment of extension rods. The extension rods serve as part of the linkage mechanism to directly couple the actuator drive to the control rods. A description of the mechanical connections from the output rack of actuator to the control rods is shown in Fig. 4.2-16 and discussed later. The actuators and their support structure can be seen in Fig. 4.2-17.

Position indication for the control rods is obtained by means of a potentiometer. The position indicating potentiometer for the shim-safety rods is a single, ten-turn precision potentiometer. A double ganged, ten-turn precision potentiometer is employed for the regulating rod. One section of the potentiometer is used to furnish the position indicating signal. The other section of this potentiometer is used for feedback to the controller.

The potentiometer signal is proportional to the travel of the actuator output rack, and hence to the vertical position of the control rods. The shaft of the potentiometer is directly geared to the rack through antibacklash spur gears.

Mating with the flange at the bottom of the rack is the extension tube assembly. It consists of fixed and adjustable length tubes with flanged connections at both ends. The upper flange attaches to the rack and the lower flange carries an electromagnet assembly. In the case of the regulating rod there is no electromagnet and the neutron absorber is connected directly to the extension tube.

4.2.3.1.2 Electromagnets – The electromagnet assembly provides a fail-safe connection between the shim-safety rod neutron absorber section and the extension tube. In normal operation the magnet is energized and the shim-safety rod neutron absorber is firmly attached to it. A scram signal or a power interruption de-energizes the magnetic field and drops the neutron absorber section into the reactor core. The magnet is designed for quick release and controlled holding power with respect to current. Table 4.2-3 outlines the pertinent parameters.

Table 4.2-3. Current vs force parameters, LPTR shim-safety rod holding electromagnets.

	Current mA	Force N	Release time ms
Nominal	22	38	10
Maximum	35	76	10

Fig. 4.2-16. LPTR shim/safety and regulating rod assemblies.

Fig. 4.2-17. LPTR pool top structure.

Also desired are, of course, dependability and consistent performance over a reasonable lifetime.

The design of the present electromagnet, shown in Fig. 4.2-18, has evolved from the early canned type used on many research reactors. The magnet is shaped as a right circular cylinder. Its inside and outside poles are fabricated from a ferro-nickel alloy, Allegheny Ludlum 4750, and are press fitted together. The armature,

Fig. 4.2-18. Shim/safety rod magnet assembly.

attached to the neutron absorber, is made of Armco ingot iron plated with 0.05 mm of copper and 0.076 mm of nickel for corrosion resistance. The magnet coil is 4650 turns of No. 30 coated copper wire which is completely encapsulated within an aluminum oxide epoxy resin formulation (75 wt% Al_2O_3). The coil leads are No. 22 fiberglass insulated copper wire. A 304 stainless steel seal ring supports the coil in the assembly. The ring is press fitted to the bottom of the inner pole piece and has essentially zero clearance between it and the other pole piece. At final assembly any voids in the unit are eliminated by filling them with the Al_2O_3 /epoxy formulation. To assist in the achievement of the design objectives, the interface between the armature and the magnet is shaped. The armature face is machined to a 0.46 m spherical radius and the magnet face to a 0.93 mm radius. Before installation in the reactor the magnets are tested and, if necessary, the air gap at the interface is adjusted to attain the desired current vs force characteristics.

Another feature of the electromagnet assembly is an integrally mounted rod make-up switch. It is actuated by a push rod acting through the center of the inner magnet pole. When the magnet engages the mating armature on the shim-safety rod, the push rod rises and closes a radiation-resistant microswitch mounted on the extension tube. This "clutch switch" not only informs the reactor operator of contact with the shim-safety rod but is also interlocked to prevent the drive motor from withdrawing any control rod unless all magnets are engaged with their armatures. For protection, the electrical leads for the electromagnet and "clutch switch" are routed to the reactor top inside the extension tube.

The magnet armature is attached to the upper end of the shim-safety neutron absorber. Mounted beneath the armature is a hydraulic shock absorber, Fig. 4.2-19. When the shim-safety rod is dropped, the shock absorber base contacts the lower flange of the magnet guide tube. The magnet guide, a 63.5 mm o.d. $\times$ 1.65 mm wall $\times$ 1.02 m long type 304 stainless steel tube that bolts to the top of the special fuel element, guides the electromagnet assembly. The shock absorber dissipates the kinetic energy of the falling rod, e.g. in a scram shutdown. The shock absorber's dashpot is designed to be critically damped or slightly overdamped to prevent oscillation of the shim-safety rod when dropped into the reactor core.

The design of the shock absorber is in accordance with that employed on the Geneva and other research reactors. Shock absorber assembly and details are shown in

Fig. 4.2-19. Shim/safety rod shock absorber assembly.

Fig. 4.2-19, relative position in Fig. 4.2-16. The shock absorber proper is of three piece construction with a fourth piece, the magnetically soft armature, providing the connection to the electromagnet.

4.2.3.1.3 Neutron Absorber Rods (Shim rods) – The neutron absorber sections of the shim-safety rods and regulating rod are similar in external appearance. They both have the same envelope dimensions of 22.2 mm thick and 57.1 mm wide with the ends rounded to a 11.1 mm radius, see Fig. 4.2-20.

The regulating rod provides continuous control of the reactor. A type 304 stainless steel sleeve of cross section described above comprises the regulating rod body which also acts as the neutron absorber. This jacket (wall thickness 1.65 mm) is welded to stainless steel plugs at top and bottom, vented to avoid trapping air, the bottom having a chamfer to facilitate insertion. A 12.7 mm diam stainless steel bar extending 0.235 m up through the center is welded to the top plug. The complete assembly of regulating rod and linkage up to the actuator is shown in Fig. 4.2-16. The

Fig. 4.2-20. Shim/safety and regulating rods.

linkage consists of a 5 m extension tube and 1.28 m adjustable section. Both the extension tube and adjustable section are of 25 mm diam $\times$ 2.76 mm wall, type 304 stainless steel tubing. Flanges are welded to both ends of the adjustable section and at the upper end of the extension tube. The lower section of the extension tube has a sleeve fitting with provision for attaching to the neutron absorber section of the regulating rod.

The shim-safety rods provide regulatory and safety control. The neutron absorber is a 1.65 mm thick type 304 L stainless steel jacket filled with high purity grade boron carbide powder. The powder filled cavity is 0.627 m in length. To strengthen the jacket against internal pressure a "Z" shaped sheet, Fig. 4.2-20, is spot welded inside the jacket over its length. The jacket is welded to stainless steel plugs at the top and bottom. The bottom plug has a 9.52 mm diam tapped hole for inserting the boron carbide powder. A plug is screwed in after filling and the hole plug welded. It is then leak checked with a helium mass spectrometer leak detector.

4.2.3.1.4 Special Fuel Elements – The special fuel element which accepts the neutron absorber section is basically a modified standard fuel element. The center fuel plates have been replaced with a pair of aluminum end plates, Fig. 4.2-1, forming a rectangular slot nominally $28.57 \times 66.5 \times 736$ mm deep. It is open at both ends and has a flange at the top to mate with the magnet guide tube. The nominal radial clearance between the neutron absorber section and the slot wall is 3.17 mm and the minimum is 2.5 mm.

4.2.3.2 Shim-Safety Rod History – The original neutron absorber design had an aluminum sheath without internal stiffening. This type of rod experienced swelling due to internal gas pressure. To guard against a repeat of this behavior and its possible consequences, a number of corrective actions were initiated in 1962.

- 1) The neutron absorber section was changed to incorporate a stainless steel can and was internally stiffened.
- 2) The width of the absorber section is periodically checked with a go – no-go gauge to determine if swelling is taking place. The gauge is set at 24.9 mm, corresponding to a minimum radial clearance of 1.47 mm between the special element slot and the absorber section.
- 3) The shim-safety rod electromagnet pickup current is limited to 35 mA. The minimum current required to lift a shim-safety rod together with a special fuel element out of the core is 45 mA. This effectively removes the possibility of a stuck rod lifting a fuel element free of the core and then having it drop back into the core.
- 4) As a backup to the electrical current limitation the special fuel elements are physically prevented from being lifted by a special fuel element hold down assembly.

Figures 4.2-13 and 4.2-14 show this assembly as the spiders that encircle the tops of magnet guide tubes. The spider to the north encircles two shim/safety rods while the south spider covers two shim/safety rods plus the regulating rod. A 38.1 mm o.d. X 3 mm wall aluminum tube is welded to each spider. These tubes extend up to the actuator support plate and prohibit the special fuel elements from being lifted.

Since the above measures were taken, there have been no problems with swelling or sticking neutron absorber sections.

Another potential problem is that of withdrawing a shim-safety rod completely out of a special fuel element. Under this condition the rod might not be reinsertable. To prevent this situation from occurring, the upper limit switch on the actuator rack stops the actuator drive motor when the bottom of the neutron absorber section is 143 mm from the top of the special fuel element. If the limit switch fails there is a mechanical stop after approximately another 41 mm of travel. Therefore, in the worst case, the engagement is still over 100 mm.

Mechanical failure and/or damage to the control rod actuators and their support structure could effect the ability to raise the rods. However, damage to these components has no effect on the ability of the rods to be scrambled. The electromagnet design is the key feature. The relatively small lifting power of the electromagnet plus the spherical interface between the armature and magnet prevents it from holding the neutron absorber section with undue force. Anything that jars the extension tubes carrying the electromagnet, or moves them rapidly upward or sideways, drops the neutron absorber section into the core.

The reactivity control system has been free of all but very infrequent minor problems since 1962.

4.2.3.3 Inspection — The inspection and testing routines used to demonstrate proper mechanical functioning of the control rods are:

- 1) Rod drop time tests are typically performed each fuel cycle. This test is primarily a check on electrical components but would indicate serious mechanical problems such as rod swelling or misalignment.
- 2) The neutron absorber section is typically checked with a go — no-go gauge every year.
- 3) Control rod actuators are partially disassembled yearly and all wear points inspected. Any worn parts are replaced and the units are cleaned and lubricated.

4.3 Nuclear Design

4.3.1 Design Basis

The LPTR fuel and reactivity control system maintains a minimum of excess reactivity consistent with prudent operational requirements. Fuel distribution and reactivity insertion rates are conservatively established such that operator responses can control all operational transients and such that reactor protective instruments can control all experimental or accidental transients while at the same time providing maximum flexibility and safety of operation.

Sufficient excess reactivity must be available in the core to compensate for reactivity changes that occur over the duration of the fuel cycle, e.g. xenon build-up and temperature effects. This excess reactivity is covered by the four shim-safety rods, which are gradually moved upward in the core as operation continues. Core loading is limited so the reactor cannot be made critical by the withdrawal of only one shim rod.

The typical reactivity losses that occur during operation at 3 MW include:

● Equilibrium xenon poisoning	4.06 %	3.24 %	$\Delta k/k$
● Two-hour xenon override	1.6	1.3	
● Fuel burnup (32 MWD)	0.7	0.56	
● Temperature (from 21°C)	0.26	0.21	

In addition, sufficient reactivity must be available for operational flexibility and to overcome fission product poisons other than xenon. Not only must the shim-safety rods be capable of safely compensating for operational reactivity changes but, when fully inserted in the core, they must provide adequate shutdown margin so that the reactor will remain sub-critical under all conceivable circumstances.

The choice of a light water moderated reactor such as the LPTR necessarily establishes a negative temperature coefficient of reactivity. The limited core size prevents radial or axial xenon oscillations at power. The fuel distribution is such that neutron flux peaking occurs in the core center, consequently any randomization of fuel element locations leads to reduction in neutron flux peaking and some reactivity loss. Loss of moderator leads to a subcritical core.

The design of experimental beam tubes terminating near the reactor core face has a direct effect on the magnitude of reactivity transients in case an empty one becomes flooded. These values are presented in Sec. 4.3.2.9. A start up accident using more conservative parameters is analyzed in Sec. 15.2.2.

4.3.2 Description

The reactor core configuration is arranged as shown in Fig. 4.3-1. The shim and regulating rods, Sec. 4.2.3, have been in this configuration since the very early stages of reactor operation (approximately 15 years). Standard fuel elements, Sec. 4.2.1, however, have been shifted into reflector positions and experimental assemblies have replaced single standard elements from time to time.

4.3.2.1 Nuclear Constants – The nuclear constants for the LPTR have been determined either in the original design calculations for the reactor or by on-going analysis by the reactor staff. Table 4.3-1 lists all the nuclear constants. Those values listed in Table 4.3-1 that are not empirical are based on a calculation using a computer program called FOOBIE described in Sec. 4.3.3.

Table 4.3-1. Nuclear constants for the LPTR.

A. Normal configuration (Fig. 4.3-1)	
Standard elements	20
Special elements	5
Void	1
Beryllium reflector elements	9
B. Normal fuel loading and characteristics	
Cold clean mass ^a (kg)	4.26
Burn mass ^a (kg)	3.66
Fission products ^a (kg)	0.60
Burn ^a (%)	14.16
Fuel burn (g/MW·d)	1.26
Fuel cycle (≈MW·d)	32
Burn/cycle (g/cycle)	40
Burn coefficient ($\% \frac{\Delta k}{k} / \text{MW} \cdot \text{d}$)	-0.016 9
($\phi \text{ MW} \cdot \text{d}$)	-2.1
Temp coefficient ($\% \frac{\Delta k}{k} / ^\circ\text{C}$)	-5.4×10^{-3} (equilibrium xenon)
($\phi / ^\circ\text{C}$)	-0.675 (equilibrium xenon)
($\% \frac{\Delta k}{k} / ^\circ\text{C}$)	-11×10^{-3} (xenon free)
($\phi / ^\circ\text{C}$)	-1.35 (xenon free)
C. Nuclear data	
η	2.07
f	0.749
$\rho\epsilon$	~1.0
k_∞	1.537
k_{eff}	1.012

Table 4.3-1 (Continued)

C. Nuclear data (Continued)

ρ_{excess}	=	1.21%
ρ_{xenon}	=	3.24%
$P_{\text{NL}}(\text{total})$	=	0.658 4
$P_{\text{NL}}(\text{fast})$	=	0.674 3
$P_{\text{NL}}(\text{thermal})$	=	0.976 5
L	=	18.5 mm
B^2	=	0.697 mm ²
τ	=	56.37 cm ²
ℓ^*	=	50.12 μ s
β_{eff}	=	0.008
$\bar{\mu}_0$	=	0.328 1
$\Sigma_a(\text{total})$	=	8.86 m ⁻¹
$\Sigma_a(\text{fuel})$	=	6.56 m ⁻¹
$\Sigma_a(\text{Al} + \text{H}_2\text{O})$	=	1.88 m ⁻¹
D	=	0.305
¹⁴⁹ Sm poison	=	0.71% (equilibrium 3 MW)
¹³⁵ Xe poison	=	3.24% (equilibrium 3 MW)
$\phi_{\text{th peak}}$	=	$7.1 \times 10^{13}/\text{cm}^2 \cdot \text{s}$
$\phi_{\text{th avg}}$	=	$1.8 \times 10^{13}/\text{cm}^2 \cdot \text{s}$

^aBased on averages for calendar year 1972.

4.3.2.2 Neutron Flux Distribution — The overall thermal neutron flux profile for the core resembles a $\cos^2 X$ type of profile where X is the core center. Flux peaking occurs in the core central void and in the control rod channels. The central void peak flux is approximately 7×10^{13} . The core average thermal flux is approximately 1.8×10^{13} n/cm²·s. A typical axial thermal neutron flux profile in a fuel element in a minimum poisoned core is shown in Fig. 4.3-2; a typical profile for this same fuel element in a xenon equilibrium condition is shown in Fig. 4.3-3. Note that the axial flux distribution has moved up into the central part of the fuel element. No radial flux profile is available.

Nuclear instruments located directly underneath the core register thermal neutron flux distribution changes. To hold constant thermal power, one must accommodate these changes in nuclear instrument readouts. Thermal power is determined by a calorimetric check on the primary coolant. The ¹⁶N system also acts as a backup check of thermal power since it is independent of the thermal neutron flux distribution. Historically the thermal neutron flux measuring instruments vary as much as 20% from a clean core to a poisoned core.

Fig. 4.3-1. Typical core loading for the LPTR.

Fig. 4.3-2. Thermal neutron axial flux profile (typical), with minimal xenon poisoning.

Fig. 4.3-3. Thermal neutron axial flux profile (typical), with equilibrium xenon poisoning.

4.3.2.3 Reactivity Coefficients — The temperature coefficient of reactivity for the LPTR typically ranges from minus 0.006% $\Delta k/k$ with a clean core to minus 0.003% $\Delta k/k$ at equilibrium xenon conditions. The temperature effects on reactivity at the LPTR are considered minimal since the temperature range is typically 35°C at startup, through a transient high of 46°C to an operating temperature of 40.5°C. The temperature coefficient of reactivity for the LPTR is negative for all temperatures in the operating range. Given the condition that the bulk coolant temperatures suddenly drops 2.7°C, and the xenon level in the core at zero, the reactivity addition would be 0.03% $\Delta k/k$, resulting in a period of about 290 s; 5.5°C temperature change would result in an asymptotic 90 s period. Asymptotic periods of 290 and 90 s correspond to movements of 19.3 mm and 38.8 mm respectively of the regulating rod in its operating range. Neither of these conditions has occurred in the 17-year operating history of the reactor. Periods of this type are easily compensated for. Reactivity variation due to temperature changes of the moderator, reflector, or control element are negligible for the LPTR since the range of temperatures encountered is small.

Void coefficients of reactivity are introduced, not because of reactor core phenomena, but because irradiation capsules can be introduced into the core.

A standard LPTR aluminum capsule (about 25 cm³) introduces about +6¢ worth reactivity when inserted into the central facility. A similar capsule in a reflector facility produces a fraction of that amount and of opposite sign. A true void coefficient associated with each facility is not practical because of the varied sizes and contents of capsules.

4.3.2.4 Control Requirements — Reactivity control and limitations are predicted on safety and operational scheduling. The magnitude of reactivity requirements are briefly presented in Secs. 4.3.1 and 4.3.2.7.

4.3.2.5 Control — Shim-safety rods are the only normal mode of control for the LPTR. The reactor uses a steady state fueling program requiring no other controls, i. e. burnable poisons. A steady state fueling program is defined as one which permits the withdrawal of the element with the greatest burn from the core, the introduction of a new element into the core, and a rearrangement of elements to essentially re-establish conditions that existed at the beginning of a particular fuel cycle.⁴

4.3.2.6 Control Rod Worths — Under normal operating conditions the core excess reactivity is controlled by the use of four shim rods and one regulating rod whose worths vary according to core configuration. The shim bank worth in the present configuration, Fig. 4.3-1, is 13.2% $\Delta k/k$. For a minimum shim bank to core excess reactivity ratio of 2, the maximum core excess reactivity is 6.6% $\Delta k/k$. The minimum shutdown margin is the worth of the shim bank less the worth of the most reactive

element in position C-4. The worth of a new, 190 g, standard element in C-4 was experimentally measured to be $4.78\% \Delta k/k$,⁵ thus setting a minimum shutdown excess reactivity margin of $8.42\% \Delta k/k$. From the above, however, it can be seen that the maximum operating excess reactivity under current conditions is limited at $6.6\% \Delta k/k$ by the criterion of shim bank to core excess reactivity ratio of 2.

The maximum rates of reactivity insertion during startup and operation are the same and would be the case where all four shim rods and the regulating rod were moved simultaneously near the center of their drive range. The maximum differential worth of the shim bank and regulating rod are approximately 36 and $1.6\% \Delta k/k$ per metre respectively. The maximum drive speeds of the shim and regulating rod are 1.3 and 4.2 mm/s respectively. These two parameters coupled together show that the maximum reactivity insertion rate is 0.051 5 and 0.007 5 $\% \Delta k/k$ per second for manual and automatic operation respectively since only the regulating rod is associated with automatic control.

A typical integral rod worth curve is shown in Fig. 4.3-4 for the shim bank. The regulating rod, whose differential worth is dependent on the shim bank position, has a relative worth curve for each shim bank position. Figure 4.3-5 shows a set of rod worth curves for certain shim bank positions.

4.3.2.7 Critical Mass and Fuel Requirements – The amount of fissionable material required to cause the reactor to go critical, or needed for a given amount of excess reactivity, depends on the core configuration and fission product inventory. Several measurements have been made of the minimum critical mass using the normal fuel element inventory (0-27% maximum burn per element) and beryllium reflection on the north and south faces. With no void in the core (except control rod channels) the critical mass was 2.46 kg ^{235}U (10% average burn). With a void in the center position the critical mass was 2.91 kg ^{235}U (11% average burn). The most recent minimum critical mass for the LPTR was determined in June, 1971 following an extended shutdown of 7 months. At that time the xenon poisoning was zero and the samarium poisoning was at a calculated level of $1.08\% \Delta k/k$ for the fully assembled core. That minimum critical mass was 3.1 kg with the beryllium reflectors positioned in their standard locations and the central void occupied with a round dummy (Fig. 4.3-1).

4.3.2.7.1 Fuel Cycles – Although the fuel cycle is dictated by the operational requirements of the reactor, the following cycle for the LPTR has been used for several years. The fuel elements are arranged in the core so that the elements with the higher masses are in the regions of higher flux and those with the lower masses are in the regions of lower flux. This maximizes flux peaking in the central void. The five special elements in the core are exceptions to the above and are normally left in the same core position throughout their entire life.

Fig. 4.3-4. Relative rod worth curve for LPTR, shim rods fully withdrawn, typical.

Reactor refueling operations are carried out according to specified procedure and all unknown core configurations are first assembled using a critical assembly procedure. In this way the chances of an unexpected reactivity increase are minimized. The single refueling error that would cause the greatest reactivity addition would be the inadvertent insertion of a fuel element into the central void of an already assembled core. (A round dummy element is inserted at all times.) Measurements indicate that this would increase core reactivity by approximately \$5.9. Since this is the greatest reactivity addition possible without a compound error, the worth of a fuel element in the position of most importance in the reactor core is set as the minimum shutdown margin (shim bank $\Delta k/k$ - core excess $\Delta k/k$).

The LPTR core is typically cycled after approximately 30 to 35 MW·d of operation. Since the fuel burn up rate for the LPTR is 1.26 g/MW·d, about 40 g of ^{235}U is consumed per fuel cycle. Each element is cycled out of the core when the burn of the element is approximately 25%. The average burn for the core ranges from 13 to 15%.

Fig. 4.3-5. Regulating rod relative worth curves (typical) for varying shim bank positions.

The fuel replacement program for the LPTR core involves replacing either one standard or one special element at a time. At the time that the new element is introduced into the core certain other elements are rearranged with general adherence to the following guide lines:

- 1) The core is left in such a configuration following any fuel movement, that no inadvertant rearrangement of the standard elements in the core would significantly increase the core excess reactivity.

- 2) The neutron flux in central facilities (C-4) is maintained at a maximum.
- 3) Typically no element is left in the core with more than 30% burn or removed with less than 20% burn.
- 4) Thermal and hydraulic parameters (Sec. 4.4) will not be reduced in quality.

Fuel cycles for the special fuel elements are an exception in that the fresh fuel element is generally placed in the spot vacated by the spent special fuel element. Normally there is about one special fuel element cycle for every four standard fuel element cycles.

Fuel burnup decreases the reactivity of the core by about 2.4¢ per MW·d. Therefore, the above fuel cycle increases the core reactivity by about 75¢. This fuel cycle has the advantage that the nuclear properties of the core remain nearly constant from cycle to cycle so that the core reactivity can be quite closely predicted and controlled.

The above fuel cycle procedure is used only when the reactivity of a new core can be calculated with an accuracy which insures that the reactor will be operated within the limits as defined in the Technical Specifications, Ch. 16.0. A critical assembly procedure is used to assemble a proposed core for which the reactivity cannot be calculated.

4.3.2.8 Significant Poisons – In the LPTR the equilibrium xenon level achieved after approximately 50 h of operation at 3 MW, is 3.24 %Δk/k. The samarium poison level varies slightly depending on the operating history, but changes by only about 0.2 %Δk/k during the course of an operating week. Figures 4.3-6–4.3-8 show the xenon levels for three typical operating schedules for the LPTR.

It can be seen from Figs. 4.3-6–4.3-8 that the amount of excess reactivity required for efficient operation depends very heavily on the operating schedule. Under the current two-shift operating schedule the excess reactivity at standard conditions* is 5.2 to 5.9 %Δk/k for the end and beginning of a fuel cycle respectively.

4.3.2.9 Experimental Facilities – Six beam tubes, positioned radially in the reactor tank, terminate with one end directly adjacent to the reflector elements of the core. Consequently they can affect the reflector savings or albedo of the core. Reactivity changes, which could occur so as to affect the shutdown margin, could be caused by accidents such as flooding of a beam port or operational errors, such as refueling mistake. Reactivity additions due to beam port flooding have been estimated previously⁶ and are tabulated below:

12-in. beam port	\$1.2
6-in. beam port	0.5
4-in. beam port	0.3

* Standard conditions are defined as zero xenon and 21°C.

Fig. 4.3-6. Xenon poisoning for typical three-shift operation.

If all facilities were to flood (a most unlikely event) then the total reactivity addition would be less than \$3.8. Also, since measurements in the DR-2 reactor (similar to the LPTR) showed that a flooded 6-in. beam port is worth 30¢, the above worths are relatively conservative. The only plausible way sudden flooding of a beam tube could occur is by dropping a shielded coffin on it. A shielded coffin is used only when the reactor is shut down. Even then, flooding of all beam tubes on both sides of the reactor would be impossible.

4.3.3 Evaluation

The LPTR has operated for approximately 17 years without any significant flaws being found in the nuclear design. During this time several CDC 6600 computer codes have

Fig. 4.3-7. Xenon poisoning for typical two-shift operation.

been developed to assist in the evaluation of the operating parameters. A brief description of some of these programs and their usage is given below.

1) RHOBURN: a three dimensional code used to calculate the amount of fuel burn in each LPTR fuel element. The core is divided up into a $5 \times 7 \times 24$ matrix with the boundaries being such that each fuel element is divided up vertically into 24 segments. The fuel burn is then calculated for each element of the matrix based on the known power distribution in the core.

2) WIREFIX: a code to process data from the LPTR wire counter. The wires are used to measure the vertical flux profile at sampling points within the core. The program makes necessary corrections to account for the radioactive decay of the wires during

Fig. 4.3-8. Xenon poisoning for typical single shift operation.

counting. The program plots the data gathered as well as normalized curves, and then punches flux profile cards for use by several of the other codes available at the LPTR.

3) SWDII: a code for calculating the thermal neutron flux and cadmium ratios for the various LPTR facilities and experiments using gold, indium, and gold alloy foils.

4) 3D-XENON: a code to calculate the xenon poison present in the core. The code sets up the same $5 \times 7 \times 24$ matrix as does RHOBURN. It solves the differential equations required to calculate the amount of xenon present in each cell. The program

then sums up the amount of xenon in each cell and converts it to a reactivity. The program gives a graphical and tabular power history of xenon poisoning. 3D-XENON can be modified by input parameters to give the same information for samarium poisoning. A controller version of the program called TTYXEN is also available.

5) FOOBIE: a code to calculate the nuclear parameters of the reactor. The calculations are based on a point reactor model and the results are presented in tabular form. FOOBIE uses the output of 3D-XENON, RHOBURN, and the weekly reactivity logs as input.

4.3.4 Tests and Inspections

Weekly reactivity surveys are made to constantly evaluate the excess reactivity at the LPTR. Computer codes described in Sec. 4.3.3 are used to evaluate the amount of fuel, fission products, and poison present in the core.

4.4 Thermal and Hydraulic Design

4.4.1 Design Bases

The LPTR is cooled by low conductivity water flowing downward through the plate type fuel elements. The water is pulled down through the core into a plenum beneath the grid plate and out an 8-in. line in the bottom of the tank. The heat picked up from the core is transferred in a heat exchanger to a secondary water system and ultimately disposed of in a cooling tower. The water is returned through four 4-in. lines located at the bottom periphery of the tank. The reactor tank is open to atmospheric pressure.

The cooling is designed to prevent boiling at any point in the reactor core under normal steady state conditions. Under abnormal or transient conditions a small amount of subcooled boiling might occur in the region of the hot spot but under no condition would bulk boiling or fuel melting be approached. The actual fuel and water temperatures in the reactor are variable depending on the outside wet bulb temperature which influences cooling tower effectiveness. Normally the water outlet temperature is controlled at 40.5°C but may reach 51.6°C before shutdown is initiated. Even at 51.6°C the above criteria hold.

The grid plate which supports the fuel elements is provided with holes for each element nose piece so that a 50 mm diam flow path is available between the 17 channels defined by the 18 fuel plates and the plenum below the grid plate. Flow for the outsides of the outer fuel plates is provided by 22-24 mm diam and 8-13 mm diam holes through the grid plate. Most of the reflector elements have no through flow. The ones that do, as well

as the dummy element in the core center, are orificed to have less flow than a standard fuel element.

The thermal-hydraulic design bases for the LPTR core are:

- 1) A maximum fuel temperature only a few degrees above the maximum cladding temperature and a factor of about 2.5 less than the melting temperature of either fuel alloy or clad.
- 2) A sufficiently high critical heat flux ratio to assure no departure from nucleate boiling at the hot spot under any anticipated circumstance.
- 3) A flow velocity low enough to assure no vibration or fuel element integrity problems.
- 4) A core flow rate well above the point at which an excessive hydraulic instability could occur.
- 5) No boiling of any kind at any point in the core at the normal operating power level.

4.4.2 Description

Tables 4.2-1 and 4.4-1 present the thermal-hydraulic parameters for the LPTR. In Table 4.2-1 a comparison is made with other reactors using similar types of fuel elements. As can be seen the flow velocities and temperatures are much less severe in the LPTR than in some of the other reactors.

The maximum fuel temperature at the minimum core flow ($53.6 \text{ cm}^3/\text{s}$) and loading (16 std, 5 spec. elements) is 115°C at 3 MW. This is well below the melting point of the fuel alloy (660°C) or its softening point (449°C) and below the maximum temperature at which this type element has been run in other reactors (156°C). Figure 4.4-1 shows the maximum fuel temperature rise above the coolant inlet temperature as a function of core flow and power level. These curves are for a nominal core loading of 19 standard and 5 special elements.

The critical heat flux ratio is about 4 with the reactor at 3 MW and minimum flow and core loading. The burnout heat flux was calculated using a correlation suggested by Gambill⁷ with a downflow correction as used by Meem⁸ at the University of Virginia. Details of this calculation are given in a later sub-section. The minimum critical heat flux ratio for the University of Virginia Reactor in 3.2 and for the Plumbrook Reactor of NASA it is 2.0 as a means of comparison.

Table 4.4-1. LPTR thermal-hydraulic parameters at 3 MW.

Core flow rate (NFR) ^a	66.24 dm ³ /s
(MFR) ^b	53.62 dm ³ /s
Coolant velocity (nominal)	0.82 m/s
(minimum)	0.61 m/s
Channel hydraulic diameter (nominal)	5.82 mm
(minimum)	5.50 mm
Channel cross-sectional area (nominal)	202 mm ²
(minimum)	192 mm ²
Coolant temperature rise (NFR)	10.5°C
(MFR)	13.3°C
Fuel element heat transfer area (standard element)	1.39 m ²
(special element)	0.696 m ²
Core loading (nominal)	19 standard
(minimum)	5 special
	16 standard
	5 special
Power density (nominal)	30 MW (th)/m ³
Maximum inlet temperature (NFR)	41.1°C
Maximum channel coolant exit temperature (MFR)	73.3°C
Maximum cladding temperature (MFR)	115°C
Maximum heat flux (minimum core loading)	0.41 MW/m ²
Burnout heat flux (NFR)	1.89 MW/m ²
(MFR)	1.387 MW/m ²

^aNFR = nominal flow rate.^bMFR = minimum flow rate.

The LPTR operates with a dummy element in its center position (C-4) creating a flux trap. The thermal flux in this position is some 3.25 times the core average flux and causes severe flux peaking in the southernmost plate of the fuel element in core position C-3. This is shown in Fig. 4.4-2. By careful flux mapping of the core⁹ the hot spot has been located 445 mm down from the top of the element in this southernmost plate. This has been verified with temperature measurements as shown in Fig. 4.4-3. In this flux mapping the nominal core loading of 19 standard and 5 special elements was present. Although the LPTR has been operated for the last 10 y with this nominal core, it would be possible to operate at 3 MW for a short time with three fewer standard elements increasing the heat fluxes, temperatures and hot spot factors accordingly. Thus this thermal-hydraulic analysis has been carried out with this "minimum" core of 16 standard and 5 special elements as a worst case.

Fig. 4.4-1. Maximum fuel temperature rise above inlet water temperature vs flow rate and power level.

The LPTR has no upper grid plate or structure on top of the core that would affect flow. Thus the coolant is pulled down through the core in proportion to the flow area presented by each element. Each of the 35 grid positions is always filled with either a standard, special, dummy, or reflector element with orificing, where necessary, to assure an even flow distribution.

Anticipated transient conditions which might affect core heat transfer would be a pump stoppage at power or a start up accident as analyzed in Ch. 15.0. With a pump stoppage the reactor is automatically shut down and only decay heat from delayed neutrons and fission products need be dissipated. As shown later, the maximum power immediately after shut down would be 0.96 MW. The LITR,¹⁰ which has thermal-hydraulic characteristics similar to the LPTR, has been operated without boiling to power levels of 1 MW without forced convection flow. By analogy the LPTR would not be expected to boil after a pump stoppage. Even if there were some boiling no damage would result as shown in the same series of LITR tests.

Under the conditions assumed in a start up accident, a power of 6.2 MW could be reached with a minimum reactor period of 138 ms. Under such conditions some boiling would result, but the burnout heat flux would not be reached. For the nominal core of 19 standard and 5 special fuel elements and a nominal flow of 66.2 dm³/s the critical heat flux ratio at 6.2 MW and a coolant outlet temperature of 51.6°C is 3.5. The point where an excursive hydraulic instability would be expected is not reached until 8 MW under these same conditions.

Fig. 4.4-2. North-south flux traverse through center of reactor core.

Fig. 4.4-3. LPTR fuel plate temperature rise as a function of axial position for grid position C-3.

4.4.3 Evaluation

The thermal-hydraulic analysis presented here and tabulated in Table 4.4-1 is based on hot spot and hot channel factors as discussed by Broadman⁹ along with the temperature and flow measurements on which Figs. 4.4-1 and 4.4-3 are based and which are described in the last preceding safety analysis report.¹¹ The percentage of power produced in the hot element (C-3) is calculated for each fuel cycle based on the average thermal flux in the element and its fuel mass using an LPTR operations staff fuel cycle code.¹² The maximum power produced in the hot element is 201 kW with the reactor at 3 MW based on this code. Pertinent hot spot and hot channel factors used in calculating the burnout and peak heat fluxes are listed in Table 4.4-2.

Table 4.4-2. LPTR core hot spot and hot channel factors.

Factor	Core loading	
	19 standard 5 special	18 standard 5 special
$P_{he}/P_{total} = \frac{\text{Power produced in hottest element (C-3)}}{\text{Power produced in core}}$	0.067	0.078
$P_{hp}/P_{ap} = \frac{\text{Power produced in hottest plate (C-3)}}{\text{Power produced in average plate (C-3)}}$	1.36	1.53
$f_{axial} = \frac{\text{Peak flux in hottest plate (C-3)}}{\text{Average flux in hottest plate (C-3)}}$	1.62	1.62
$frac = \frac{\text{Hot channel power produced down to hot spot}}{\text{Total hot channel power}}$	0.58	0.58

The peak heat flux (ϕ_{max}) in the LPTR core, occurring at the core hot spot located 0.445 m from the top of the element in C-3 and on its southernmost plate, was calculated as follows:

$$(\phi_{max}) = \frac{(P_T) \left(\frac{P_{he}}{P_T} \right) \left(\frac{P_{hp}}{P_{ap}} \right) (f_{axial})}{A_{element}}$$

with

$$A_{element} = \text{heat transfer area of the element} = 1.4 \text{ m}^2$$

$$P_T = \text{total core power level}$$

The hot channel exit temperature is calculated as follows:

$$T_{hot \text{ channel exit}} = T_{inlet} + \Delta T_{hc}$$

$$\Delta T_{hc} = \frac{\text{Power produced in hot channel}}{\text{Mass flow rate through hot channel} \times \text{specific heat of water}}$$

$$\Delta T_{hc} = \frac{(P_T) \left(\frac{P_{he}}{P_T} \right) \left(\frac{P_{hp}}{P_{ap}} \right)}{(\rho) (F) (C_p) \left(18 \frac{\text{plates}}{\text{element}} \right) \left(\frac{0.314}{5.34N + 22.9} \frac{\text{hot channel flow}}{\text{core flow}} \right)}$$

where flow divides as the cross-sectional area of the channels with

F = total core flow rate

N = number of fuel plates in core/18

ρ = coolant density

Multi-channel cores, as in the LPTR, are subject to an excursive flow instability in the hot channel when the core flow drops to a point where the pressure drop through the channel starts to increase with decreasing flow. This is analyzed in two ways in a later subsection defining the thermal-hydraulic safety limits. It can be said here, however, that at the minimum core flow $53.6 \text{ dm}^3/\text{s}$ and loading (16 standard, 5 special) with a coolant outlet temperature of 51.6°C no such instability would develop until the reactor power reached at least 6 MW, a factor of two above its operating power level.

If an excursive hydraulic instability or a departure from nucleate boiling were to take place in the LPTR core some fuel melting might result. If the fuel temperature were to rise above the melting point an exothermic aluminum-water reaction could result releasing large amounts of chemical energy. Such reactions have played only a minimal part in accidents with cores of the LPTR type.¹³ Since no temperatures approaching fuel melting are expected in the LPTR under any anticipated operating condition, the chance for a chemical reaction of consequence appears minimal as discussed in Ch. 15.0.

An estimate of the error (one standard deviation) to be assigned to the variables used in this thermal-hydraulic analysis and the following safety limit analysis are listed in Table 4.4-3.

4.4.4 Testing and Verification

The LPTR has been operating at 3 MW with the existing core configuration since 1967. Prior to that, testing of the thermal-hydraulic parameters was carried out for the change from 2 MW to 3 MW operation. Results of these tests are shown in the previous safety analysis report¹¹ and in Figs. 4.4-1 and 4.4-3 of this report.

4.4.5 Instrumentation Application

The LPTR uses no in-core instrumentation and needs none to assure safety. A fuel element with thermocouples attached to its southernmost plate was used to measure

Table 4.4-3. Error estimate for variables used in thermal-hydraulic analysis.

Variable	σ
P_T Total reactor power	2.5%
P_{he}/P_T	3.0%
P_{hp}/P_{ap}	3.5%
h_{cf}/c_{of} Percent of core flow in hot channel	3.0%
f_{axial}	2.5%
$A_{element}$	2.5%
t_{outlet}	2.0%
F	5.0%
$frac$	2.5%
$R = \frac{\text{Power produced in hot channel}}{\text{Power to produce steam at exit of hot channel}}$	2.5%
$\Delta T_{hc}/\Delta T_{sat}$	1.0%

plate temperatures in core position C-3 and the results were shown in Fig. 4.4-3. This instrumented element is not a normal part of the core, however.

Thermohms in the primary coolant line are used to measure the temperature rise across the core and the actual inlet and outlet temperatures. The core flow rate is monitored by measuring the pressure drop across an orifice. This is supplemented by a magnetic flowmeter which determines flow rate by the interaction of the ions in the primary water with a magnetic field. Reactor power is based on the core flow and ΔT measurements. Details of this instrumentation are given in Ch. 5.0 and 7.0 and errors inherent in the readings are reflected in Table 4.4-3.

4.4.6 Thermal and Hydraulic Safety Limits

4.4.6.1 Basis – Safety limits have been established for power level, primary flow, and outlet temperature. These limits are listed in Table 4.4-4 along with the limiting safety system settings and normal operating settings. These safety limits ensure that fuel melting and the attendant fission product release will not occur providing the limits are not exceeded. The LPTR protective system instrumentation is set to shut down the reactor at levels that allow an adequate margin below the safety limits.

A safety limit is defined as a chosen value of a specified parameter for which it can be said with confidence that no serious consequences will occur if the limit is not exceeded. Safety limits have been calculated for power level, flow rate, and outlet temperature for the condition where the two parameters not under evaluation are set

Table 4.4-4. LPTR thermal-hydraulic safety limits^a

Parameter	$P_T > 100 \text{ kW}$			$P_T < 100 \text{ kW}$
	Power level	Flow	Outlet temperature	
Power	at safety limit	4.5 MW	4.5 MW	0.15 MW
Flow	53.6 dm ³ /s	at safety limit	53.6 dm ³ /s	none
Outlet temperature	51.6°C	51.6°C	at safety limit	51.6°C
Inlet temperature	30°C	29°C	39°C	—
Safety limits based on:				
Burnout heat flux	6.7 ± 1.0 MW	34.7 ± 3 4.4 dm ³ /s	87 ± 11°C	
Excursive instability	6.0 + 1.1 MW	40.0 ± 6.3 dm ³ /s	67 ± 8°C	
Safety limit	4.9 MW	46.4 dm ³ /s	59°C	
Limiting safety system setting (LSSS)	4.5 MW	53.6 dm ³ /s	51.6°C	0.15 MW
Operating point	3.0 MW	66.2 dm ³ /s	40.5°C	0.10 MW
Burnout heat flux at LSSS	1.17 ± 0.16 MW/m ²			0.17 MW/m ²
Maximum heat flux at LSSS	0.63 ± 0.063 MW/m ²			0.021 MW/m ²
Minimum burnout ratio	1.45			8
Hot channel exit temperature at safety limit	88 ± 8°C	90 ± 8°C	91 ± 8°C	
Saturated water temperature	111.6°C	111.6°C	111.6°C	

^aErrors shown are two standard deviations from mean.

at their limiting safety system setting. The calculated safety limits are sufficiently below the threshold for fuel melting to preclude actual damage due to uncertainties in the experimental observations and measured values on which the calculations are based.

In each case the safety limit is considered to be reached when either of the following happen:

- 1) There is a departure from nucleate boiling at the core hot spot (burnout).
- 2) There is an excursive flow instability developed in the hot channel.
- 3) The exit temperature in the hot channel reaches the saturation temperature of the water at that pool depth.

The onset of any of these conditions does not mean that fuel melting has occurred but rather is a prelude to it. As can be seen in Table 4.4-4 condition 2 is governing.

Safety limits are established for both the forced convection and natural convection ($P_T < 100$ kW) modes of operation.

4.4.6.2 Burnout Heat-Flux Correlations — The burnout heat flux correlations of Gambill⁷ were used to determine the conditions at which a departure from nucleate boiling would occur. In the reference cited Gambill outlines a procedure by which the minimum value of ϕ_{bo} can be obtained based on the lower bound of all available experimental data. This technique was used in this analysis with a correction for downflow also proposed by Gambill.^{8,14} In addition to using conservative correlations for ϕ_{bod} an error analysis was made on the hot spot and hot channel factors and used to establish a 2σ error band. The lower limit of the band gives ϕ_{bod} at greater than the 95% confidence level.

The burnout heat flux equation used along with the downflow correction are as follows. These equations and coefficients are left in standard English units for simplicity.

$$\phi_{bod} = \frac{\phi_{bo}}{6.58 (N_{Gr}/N_{ka})^{0.39}}$$

$$\frac{N_{Gr}}{N_{ka}} = \frac{(\rho_f)^2 g \beta (T_b - T_{inlet}) D_e}{f_w G^2} \quad \text{evaluated at the average fluid temperature}$$

$$\phi_{bo} = K \lambda (\rho_v)^{0.5} (\sigma g \Delta \rho)^{0.25} (F_{sub}) + h_{nb} (T_w - T_b)_{bo}$$

$$F_{sub} = 1 + 0.8 \left(\frac{\rho_f}{\rho_v} \right)^{0.75} \left[\frac{C_p (T_{sat} - T_b)}{9.8 \lambda} \right]$$

$$h_{nb} = 0.61 \frac{(V)^{0.8}}{(D_e)^{0.2}}$$

$$(T_w - T_b)_{bo} = (T_w - T_{sat})_{bo} + (T_{sat} - T_b)$$

$$f_w = \left[0.0032 + \frac{0.221}{(N_{Re})^{0.237}} \right] \left(\frac{\mu_w}{\mu_b} \right)^{0.14}$$

The symbols have the following meanings:

- ϕ_{bod} = burnout heat flux with downflow correction (Btu/hr-ft²)
- ϕ_{bo} = burnout heat flux based on Gambill technique (Btu/hr-ft²)
- ρ_f = density of saturated water at minimum pool depth (lb/ft³)
- g = acceleration of gravity (ft/hr²)
- β = water volumetric coefficient of expansion (°F⁻¹)

T_b = local bulk water temperature at hot spot ($^{\circ}\text{F}$)
 T_{inlet} = inlet primary coolant temperature ($^{\circ}\text{F}$)
 D_e = equivalent channel diameter (ft)
 f_w = Darcy-Weisbach friction factor
 G = mass velocity through hot channel ($\text{lb}/\text{ft}^2\text{-hr}$)
 $K = 0.1$
 λ = heat of vaporization of water at minimum pool depth (Btu/lb)
 ρ_v = density of saturated steam at minimum pool depth (lb/ft^3)
 σ = liquid-vapor interface surface tension (lb/ft)
 $\alpha = 4.17 \times 10^8 \text{ ft}/\text{hr}^2$
 $\Delta\rho = \rho_l - \rho_v$ (lb/ft^3)
 C_p = specific heat of water ($\text{Btu}/\text{lb } ^{\circ}\text{F}$)
 T_{sat} = saturated temperature of water at minimum pool depth ($^{\circ}\text{F}$)
 $\mathcal{F}$ = physical properties factor based on T_b (Figure 1 of Ref. 7)
 h_{nb} = heat transfer coefficient at hot spot ($\frac{\text{Btu}}{\text{hr} - \text{ft}^2 - ^{\circ}\text{F}}$)
 V = velocity through hot channel (ft/sec)
 T_w = hot spot wall temperature at burnout ($^{\circ}\text{F}$)
 $(T_w - T_{\text{sat}})_{\text{bo}}$ = minimum burnout wall superheat (Fig. 2 of Ref. 7)

The saturated properties of water were taken at a pressure of 0.152 MPa ($T_{\text{sat}} = 112^{\circ}\text{C}$). This corresponds to the hydrostatic pressure with the reactor tank water level at its minimum position 1.1 m below the top. (The reactor would scram if the water level were lower.) For the minimum core loading of 16 standard and 5 special fuel

Fig. 4.4-4. Burnout heat flux, ϕ_{bod} , and core maximum heat flux, ϕ_{max} , vs reactor power. Outlet, 52°C , flow, $53.6 \text{ dm}^3/\text{s}$, core loading, 16 std. and 5 special elements.

elements, a core coolant flow of $53.6 \text{ dm}^3/\text{s}$ (850 gpm) and outlet temperature of 51.6°C (125°F), the burnout heat flux is $1.39 \text{ MW}/\text{m}^2$ ($4.4 \times 10^5 \text{ Btu}/\text{hr}\text{-ft}^2$) at 3 MW. This gives a burnout ratio of 4.4 at the hot spot. The burnout heat flux as a function of reactor power is shown in Fig. 4.4-4 along with the core maximum heat flux.

The burnout heat flux for the LPTR under natural circulation conditions is taken as $170 \text{ kW}/\text{m}^2$ ($54,000 \text{ Btu}/\text{hr}\text{-ft}^2$) based on the data of Gambill and Bundy.¹⁵ Their tests, used to determine this number, were conducted using an MTR type fuel element held vertical with a 10% relative flow area in the return path. The peak heat flux in the LPTR at 150 kW, the

limiting safety system setting under natural circulation operation, is 21 kW/m^2 (6600 Btu/hr-ft^2) giving a burnout ratio of 8 under natural circulation operation.

4.4.6.3 Hydraulic Instability Calculations – Due to the effects of increased pressure loss with subcooled nucleate boiling and adverse buoyancy from heating, an excursive hydraulic instability could occur in the LPTR hot channel. Such an instability occurs in a multi-channel core when the pressure loss in the hot channel begins to increase with decreasing flow rate. This causes a drastic reduction in flow to the hot channel since the pressure drop is governed by the total core, not the hot channel. Such a flow reduction or instability in the hot channel could lead to fuel melting.

A flow instability is determined by calculating channel pressure losses for successively decreasing flow rates at a fixed power level. The minimum point of this curve locates the minimum stable flow rate at that power level.

The important contributions to the pressure drop across the LPTR core are entrance losses, losses due to elevation change, and friction losses. The entrance loss and elevation change loss are given by

$$\Delta p_{\text{entrance}} = K \frac{V^2}{2g} \text{ with } K \approx 1.0$$

$$\Delta p_{\text{elevation}} = \int_{\text{entrance}}^{\text{exit}} \rho dz = L (1 - \beta \Delta T_{\text{hc}})$$

with β evaluated at the average hot channel temperature

L = channel length (ft.)

ΔT_{hc} = hot channel temperature rise ($^{\circ}\text{F}$)

The friction loss was determined using data gathered by Dormer and Bergles¹⁶ for hydraulic instabilities in flow through round tubes. For a given channel length to diameter ratio their data was correlated to give the ratio of the actual channel pressure drop to isothermal pressure drop as a function of the ratio of the actual channel heat input to the channel heat input necessary to raise the channel coolant exit temperature to the saturated temperature (defined in Table 4.4-4 as R). Thus for a given reactor power one can find R and from the curve defined by the data pick off the pressure drop ratio. Knowing the channel friction factor the isothermal pressure drop can be calculated. From this one can find the actual friction pressure drop. This method is essentially identical to that which was used to determine some of the MITR technical specifications.¹⁷ The equations used to determine R and the isothermal pressure drop through the hot channel are as follows.

$$R = \frac{P_{hc}}{P_{saturated}} = \frac{(P_T) \left(\frac{P_{he}}{P_T} \right) \left(\frac{P_{hp}}{P_{ap}} \right)}{(F) \left(\frac{hcf}{cof} \right) (T_{sat} - T_{inlet}) \left(18 \frac{\text{plates}}{\text{elements}} \right) (Cp)}$$

$$\Delta p_{f, iso} = f_w \frac{L}{D_e} \frac{V^2}{2g}$$

The symbols are as defined in Sec. 4.4.3 and on the previous pages of this subsection.

The data of Dormer and Bergles used for the LPTR hot channel is shown in Fig. 4.4-5.

In Fig. 4.4-6 the hot channel pressure drop as a function of core flow rate is shown.

Fig. 4.4-5. Pressure drop ratio vs hot channel power ratio.

Fig. 4.4-6. Hot channel pressure drop vs core flow rate.

In this figure it can be seen that the slope changes at a core flow rate of $53.6 \text{ dm}^3/\text{s}$ (850 gpm.) Thus the reactor power can be raised to 6.0 MW with $T_{\text{outlet}} = 51.6^\circ\text{C}$ (125°F) for a core with 16 standard and 5 special elements before a choked flow condition results. The safety limit for reactor power is based on this curve.

To double check the MITR analysis the data of Forgan and Whittle¹⁸ was used. They studied excursive flow instabilities in both round tubes and narrow rectangular channels and concluded that for the L/D_e ratio of the LPTR hot channel all data correlated as follows.

$$\frac{\Delta T_{\text{hc}}}{T_{\text{sat.}} - T_{\text{inlet}}} = 0.8 \text{ with } \sigma = 0.008$$

They stated this correlation held for down-flow as well as upflow. Their correlation gives results very close to those using the MITR technique and indeed gives the same reactor power safety limit of 6 MW.

4.4.6.4 Error Analysis of Thermal-Hydraulic Safety Limits — Using the standard deviations for the variables involved as shown in Table 4.4-4 an error analysis was performed on the equations defining peak heat flux, burnout heat flux,

excursive hydraulic instability, and channel exit temperature. The standard deviation for each of these calculated quantities was determined by the following equation.¹⁹

$$\sigma_{\psi} = \left[\sum_{j=1}^n \left(\frac{\partial \psi}{\partial x_j} \right)^2 S_{x_j}^2 \right]^{1/2}$$

with

ψ = quantity for which the standard deviation is desired

x_j = variable in equation defining ψ

S_{x_j} = standard deviation of variable

n = number of variables

The $\pm$ tolerances shown in Table 4.4-5 are $2\sigma_\psi$ and thus represent a band within which the quantity ψ should fall with a 95% confidence level. The mean value of $\psi \pm$ its 2σ error, whichever is the most conservative, defines the safety limit.

REFERENCES

1. J. E. Cunningham and E. J. Boyle, "MTR Type Fuel Elements," Proc. U. N. Intern. Conf. Peaceful Uses At. Energy, 1st, Geneva, 1955, (United Nations, Geneva, 1956) vol. 9, p. 203.
2. D. S. Billington, "Radiation Damage in Reactor Materials," op. cit. vol. 7, p. 425.
3. Nuclear Reactor Fuel Elements, A. Kaufmann, Ed. (Interscience Publishers, New York, 1962) p. 578.
4. E. E. Hill and F. J. Shon, Fuel Programming in Pool-Type Research Reactors of Intermediate Power Levels, Lawrence Livermore Laboratory, Rept. UCRL-6324 (1961).
5. W. J. Richards, Core Physics Tests of December, 1972, Lawrence Livermore Laboratory, Memo of August 7, 1973.
6. Livermore Pool Type Reactor Safeguard Report, (with Addenda I, II, III, IV), Foster Wheeler Corporation, Rept. AECD-4258 (1956).
7. W. R. Gambill, Design Curves for Burnout Heat Flux in Forced-Convection, Subcooled, Light-Water Systems, Oak Ridge National Laboratory, Tenn., Rept. ORNL-TM-2421 (1968).
8. J. L. Meem, Revised Safety Analysis Report in Support of Amendment to License R-66 for Two Megawatt Operation University of Virginia Reactor, University of Virginia, Rept. UVAR-8 (1970).
9. G. A. Broadman, Determination of Maximum Fuel Plate Temperatures in a Light Water Moderated Research Reactor with a Central Water Void, Lawrence Livermore Laboratory, Rept. UCRL-7214 (1963).
10. W. M. Breazeale, T. E. Cole, and J. A. Cox, Preliminary Boiling Experiment in the LITR, Oak Ridge National Laboratory, Tenn., Rept. TID-5065 (1953).
11. J. W. Bolstad, G. E. Cummings, J. L. Eagan, and H. F. Finn, Safety Analysis Report for the Livermore Pool Type Reactor, Lawrence Livermore Laboratory, Rept. UCRL-50147 (1966).
12. G. V. Pefley, Jr. RHO BURN: A Nuclear Reactor Code for Calculating Burnup and Operational Parameters for the LPTR, Lawrence Livermore Laboratory, Rept. UCRL-7477 (1963).
13. R. W. Miller, A. Sola, and R. K. McCardell, Report of the Spert I Destructive Test Program on an Aluminum Plate-Type, Water-Moderated Reactor, Phillips Petroleum Company, Rept. IDO-16883 (1964).
14. W. R. Gambill, Nuclear Safety 9, (6) 471 (1968).
15. W. R. Gambill and R. D. Bundy, Burnout Heat-Flux for Low Pressure Water in Natural Circulation, Oak Ridge National Laboratory, Tenn., Rept. ORNL-3026 (1960).
16. T. Dormer, Jr., and A. E. Bergles, Pressure Drop for Water Flowing in Small Diameter Tubes Under Isothermal Nonboiling and Surface Boiling

Conditions, Massachusetts Institute of Technology, Cambridge, Mass., Rept. 8767-31 (1964).

17. Technical Specifications for the MIT Research Reactor, Massachusetts Institute of Technology, Cambridge, Mass., Rept. MIT-NE-62 (1965).
18. R. Forgan and R. H. Whittle, Pressure-Drop Characteristics for the Flow of Subcooled Water at Atmospheric Pressure in Narrow Heated Channels, Atomic Energy Research Establishment, Harwell, Berkshire, Rept. AERE-M1739 (1966) parts I and II.
19. E. B. Haugen, Probabilistic Approaches to Design (John Wiley and Sons, New York, 1968) p. 92.

5.0 REACTOR COOLANT SYSTEM

5.1 Summary Description

5.1.1 Reactor Primary Coolant System

The reactor core is cooled by an open circulation system with low conductivity water (LCW). The cooling water is circulated downward through the core, passing through a holdup tank, the primary coolant pump, and the heat exchanger before returning to the reactor tank, (see Fig. 5.1-1). The present flow rate of the primary system is about $67 \text{ dm}^3/\text{s}$. About 45 m^3 of water are contained in the primary system which is constructed mainly of aluminum but has a cast iron primary pump and valves, and a stainless steel heat exchanger.

Cooling water enters the reactor pool tank through four 4-in. diam mains in the bottom head of the tank. The flow is upward around the side of the core, then down through the core, leaving the tank through an 8-in. main in the center of the bottom of the tank. The water then passes through a 11 m^3 aluminum holdup tank to allow decay of the ^{16}N produced in the core. The holdup tank is located beneath the reactor building floor (see Figs. 5.1-2 and 5.1-3), immediately adjacent to the reactor. After leaving the holdup tank, the coolant line enters the equipment pit and the water passes through the 15 kW centrifugal primary pump and then through the shell side of the heat exchanger. The bulk of the stream returns to the reactor tank. A small amount ($\sim 0.6 \text{ dm}^3/\text{s}$) is drawn off and routed through the beam plug cooling system and the primary purification system. The heat exchanger is a counter-flow heat exchanger with a two-pass shell section for primary flow and a four-pass tube section for secondary flow. Table 5.1-1 presents the specifications of the primary coolant system components.

5.1.2 Secondary Coolant System

The secondary coolant system consists of the tube side of the heat exchanger, the cooling tower, automatic temperature control valve plus several manually operated valves and two 56 kW deep well centrifugal pumps and two 11 kW centrifugal pumps. The secondary coolant pumps may be used either singly or in parallel. With the present heat exchanger one 56 kW pump will circulate about $120 \text{ dm}^3/\text{s}$ and all pumps will deliver about $176 \text{ dm}^3/\text{s}$ total in parallel operation with no throttling by the automatic temperature control valve. After leaving the pump, the secondary coolant flows through the automatic temperature control valve and through the tube side of the heat exchanger. After leaving the exchanger, the water is cooled in a forced draft cooling tower. The secondary piping consists mainly of 12 in. Sch. 40 steel pipe reduced to 8 in. Sch. 40 pipe in the equipment pit.

Fig. 5.1-1. Primary and secondary coolant systems.

Fig. 5.1-2. Reactor containment building elevation.

The secondary coolant is regulated by the temperature control valve to hold the core temperature constant. This is achieved by monitoring the reactor primary coolant outlet temperature after it leaves the holdup tank and throttling the secondary coolant flow rate by means of a pneumatic controller and control valve. At 3 MW the reactor outlet temperature is normally held at 41°C for experimental reasons. The secondary flow rate thus varies throughout the day because of its dependence on atmospheric conditions at the cooling tower. Experience has shown that the 41°C reactor outlet temperature may be maintained on all but the hottest (wet bulb) days of the year with the use of both secondary coolant pumps. On days of very high wet bulb temperature it is necessary to allow the reactor outlet temperature to exceed 41°C in order to dissipate 3 MW in the cooling tower. For instance, with a wet bulb temperature of 21°C, the reactor outlet temperature must be at or above 46°C in order to dissipate 3 MW in the cooling tower. A wet bulb temperature exceeding 21°C occurs only about 18 hours per year at the LPTR. Even with the highest wet bulb recorded at the LPTR (22°C), 3 MW may be dissipated in the cooling tower with the reactor water outlet temperature less than 49°C, well below the temperature at which boiling would occur in the core.

The primary and secondary systems have instrumentation to measure temperature, flow rate, radioactivity, pressure, and pH. In addition, the primary system water

Fig. 5.1-3. Reactor containment building plan view.

conductivity and pool liquid level are monitored. A schematic diagram of the cooling system and its associated instrumentation is shown in Fig. 5.1-1. The primary and secondary systems are independent of one another except for their connection at the heat exchanger. Table 5.1-2 presents the specifications of the secondary coolant system components.

5.1.3 Primary Coolant Drain System

Temporary storage of reactor water is provided for by a 38 m^3 storage tank buried under the floor of the reactor building, see Fig. 5.1-3. Gravity flow will drain the reactor pool tank in 1800 s. Once in the storage tank, reactor water can be pumped back into the main coolant line, or via the pit sump pump to either of the waste tanks in the equipment pit or to portable tanks outside the building. Water enters the storage tank from a direct connection to the reactor tank bottom and from the overflow gutter

Table 5.1-1. Primary coolant system.

Component	Description	Rating	Material	Code
Primary pump (P-1)	Worthington 4 CNE-84 5 in.-125# suction and 4 in.-125# discharge flanges, 20.3 rev/s, 15 kW, single stage centrifugal.	64 dm ³ /s at 180 kPa total discharge head (TOH), 1.2 MPa at 66°C	SS impeller and shaft, nickel plated cast iron casing.	Mfg. Std.
Primary heat exchanger (E-1)	Fuel Engineering, Div. Textron Inc., vertical, 1.0 m diam X 2.7 m length, counterflow two pass shell and four pass tube section.	410 kPa at 66°C, shell and tube side	304 SS both tube and shell side.	ASME Sec. VIII, Code cases 1270 N and 1273 N
Primary flow indicator (F-1)	Fischer & Porter Co., Model 10D1416B, 6 in. size magnetic flowmeter, 6 in.-150# flanges.	95 dm ³ /s, 1.76 MPa at 66°C	SS electrodes, neoprene lining, SS tube and carbon steel flanges.	Mfg. Std.
Primary orifice plate and flange	6 in.-300# flanges 102 mm orifice diam.	4.9 MPa at 66°C	6061-T6 aluminum 304 SS orifice plate.	Mfg. Std.
Primary coolant system valves	Crane Co., 125# cast iron flanges 2-1/2 in. and larger. Valves 2 in. and smaller are 150# screwed connections.	1.21 MPa at 66°C	Cast iron and ferrosteel.	Mfg. Std.
Holdup tank (TK-4)	Consolidated Western Steel, 11 m ³ capacity, 1.8 m diam X 4.70 m length	170 kPa at 93°C	5052-F aluminum	ASME Sec. VIII
Primary coolant system piping	6 in. Sch. 40	2.3 MPa at 120°C	6063-T6 aluminum	ASTM-B241
	9 in. Sch. 40	2.0 MPa at 120°C		
Flanges	150# weld neck	1.9 MPa at 66°C	6061-T6 aluminum	ASTM-B247, ASA-B 16.5
Branch fittings (2 in. IPS and smaller)	Screwed, 150#	2.8 MPa at 93°C	Alcoa B214 or 6061-T6 aluminum	None
	Welding, Sch. 40	1.7 MPa at 93°C	3003-F aluminum	ASTM-B361 ASA-B16.9
Main loop fittings	6 in. Sch. 40 butt welding	3.2 MPa at 120°C		
	8 in. Sch. 40 butt welding	2.8 MPa at 120°C	6061-T6 aluminum	ASTM-B361
Reactor pool tank	See Sec. 5.4			
Strainer	Bailey 100-R-1, 6 in.-125# flange connections	1.2 MPa at 66°C	Cast iron	Mfg. Std.

near the top of the reactor pool tank. The storage tank is lined with a bisonite protective coating and is equipped with a liquid level indicator alarm. The tank is vented through the main building stack. Table 5.1-3 presents the specifications of the primary coolant drain system components.

Table 5.1-2. Secondary coolant system.

Component	Description	Rating	Material	Code
Secondary pumps (P-11, 12)	Peerless Pump, 14 1A-2 two stage vertical turbine type, 8 in.-125# discharge flange, 29.3 rev/s, 56 kW.	105 dm ³ /s at 370 kPa TDH 1.21 MPa at 66°C	Cast iron	Mfg. Std.
Secondary pumps (P-5, 6)	Peerless Pump 6 1A-4 four stage vertical turbine type 4 in.-125# discharge flange, 59.3 rev/s, 11 kW.	1.21 MPa at 66°C	Cast iron	Mfg. Std.
Cooling tower (T-1)	Marley Co., Model 220- 1-01, crossflow type, 3.66 m fan, 4.40 rev/s, 22 kW.	164 dm ³ /s at 30.6°C hot water, 27°C cold water, 21°C WH	Redwood	Mfg. Std. with 1.4 kPa wind load or 1.28 m/s ² (0.131 g) earthquake
Secondary orifice plate and flanges	8 in.-300# flanges 143 mm orifice diam.	4.0 MPa at 66°C	Carbon steel	Mfg. Std.
Control valve (V-97)	Dezurik Corp., Fig. 118, 8 in.-125# flanges, air to close, spring return, fails open.	1.0 MPa pressure rating at 66°C	Cast iron	Mfg. Std.
Valves	8 in.-125# and 12 in.- 125# flange connections	1.21 MPa at 66°C	Cast iron	Mfg. Std.
Piping	8 in. Sch. 40	5.8 MPa at 93°C	Carbon steel	ASTM-A53, GR B
	12 in. Sch. 40	4.7 MPa at 93°C		
Flanges	150# welding neck	1.7 MPa at 93°C	Carbon steel	ASTM-A 181 GR 1, ASA B 16.5
Fittings	Sch. 40 welding	3.5 MPa at 343°C	Carbon steel	ASTM-A 106, GR A.
	300# screwed	6.9 MPa at 66°C	Malleable iron	ASTM A 47 ASA 16.10
Primary coolant heat exchanger (E-1)	See Table 5.1-1	410 kPa at 66°C		

5.1.4 Beam Port Coolant System

Each of the six beam ports is equipped with cooling water from the primary coolant system. The discharge pressure of the primary coolant pump is used to circulate reactor primary coolant water through each facility and return it to the primary system in the low pressure section. The water in the annulus between the sleeve and tube of the B-6 facility is circulated by allowing it to discharge into the reactor pool. Table 5.1-4 presents the specifications of the beam port coolant system components.

Table 5.1-3. Primary coolant drain system.

Component	Description	Rating	Material	Code
Storage tank (TK-1)	Consolidated Western Steel, 38 m ³ capacity, 3 m diam X 5.66 m length	170 kPa at 93°C	SA-7 carbon steel	ASME Sec. VIII
Piping	3 in. Sch. 40	2.3 MPa at 93°C	6063-T6 aluminum	ASTM-B 241
Valves	125# flanged	1.2 MPa at 66°C	Cast iron	Mfg. Std.
Fittings	Sch. 40 welding	2.3 MPa at 93°C	Alcoa B214 or 6061-T6 aluminum	None
Storage tank 2 in. drain line and fittings	150# screwed	2.1 MPa at 66°C	Galvanized iron	ASTM A 120
Flanges	150# weld neck	1.0 MPa at 93°C	6061-T6 aluminum	ASTM B 247

Table 5.1-4. Beam port cooling system.

Component	Description	Rating	Material	Code
Piping	Sch. 40	3.06 MPa at 93°C	6063-T6 aluminum	ASTM-B 241
Fittings	Screwed	2.8 MPa at 38°C	Alcoa B 214 or 6061-T6 aluminum	None
	Welded	2.8 MPa at 93°C	3003-F aluminum	ASTM-B 361
Valves	Crane No. 222 1/2 H and 223 1/2 H	21 MPa at 38°C	Type 316 SS	Mfg. Std.
	125#	1.21 MPa at 66°C	Cast iron	Mfg. Std.

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Design of Reactor Coolant Pressure Boundary (RCPB)

5.2.1.1 Performance Objectives – The basic performance objectives for the reactor coolant system are a heat removal capacity of 3.0 MW and the requirement that no nucleate boiling occur in the core. The necessary parameters for meeting these objectives are:

Primary coolant drain system

Design capacity	38 m ³
Maximum obtainable pressure	120 kPa
Design pressure	170 kPa
Elapsed time to drain system	1800 s

Beam port coolant system

Design pressure	410 kPa
Design Temperature	66°C

The flow rate is not significant. Coolant is required primarily to purge system of radiolytic decomposition products in water.

Operating Conditions

The normal operating¹ conditions are:

Reactor power	3 MW
Primary coolant flow rate	67 dm ³ /s
Core exit temperature	41°C
Core inlet temperature	30°C
Primary coolant pump discharge pressure	260 kPa
Secondary coolant flow rate	92 dm ³ /s
Secondary coolant pump discharge pressure	450 kPa
Heat exchanger secondary inlet temperature	23°C
Heat exchanger secondary exit temperature	31°C
Heat exchanger secondary side pressure	140 kPa

The upset¹ conditions are:

Reactor power	3.15 MW
Primary coolant flow rate	54 dm ³ /s
Core exit temperature	52°C

These conditions may occur separately or simultaneously. If they were to occur simultaneously the largest change in the reactor coolant system conditions would be a temperature rise of 8°C in the core inlet temperature to 38°C.

Emergency¹ conditions that could possibly occur are:

Maximum primary coolant pressure	390 kPa
Maximum secondary coolant pressure	660 kPa

These conditions would be the result of inadvertent closure of manually operated valves on discharge side of main coolant pump or secondary coolant pump respectively.

5.2.1.2 Design Parameters – The design parameters of the various components within the RCPB are listed in Tables 5.2-1 through 5.2-4. Comparing the system design with operating conditions indicates that the pressure-temperature ratings of the components are greater than the imposed conditions. It may therefore be concluded

that the components are adequate. The single exception to the above statement is described below.

The secondary coolant system is capable of attaining a maximum pressure of 660 kPa. However, the primary heat exchanger in the system is rated for 410 kPa. To protect the heat exchanger from possible overpressurization the following arrangement was made:

The secondary system valves downstream of the heat exchanger (V-65, heat exchanger discharge and V-154, cooling tower flow control) are locked in the fully open position. These valves are used very infrequently and never while the reactor coolant system is in operation. In addition locking them creates no safety question. With these valves locked open the secondary system coolant pressure cannot exceed the 410 kPa design pressure rating at the heat exchanger.

At present the secondary coolant pressure reaches 310 kPa at the heat exchanger for approximately 20 days per year. These are days with high dry and wet bulb temperatures, making it necessary to increase the secondary flow to the heat exchanger. This constitutes about 6% of the reactor operating time. The remainder of the year the pressure at the heat exchanger is in the 210-280 kPa range.

5.2.1.3 Compliance with 10 CFR Part 50, Sec. 50.55a — Almost all of the components used in the RCPB are of normal industrial quality. At the time of the LPTR's design and construction the "Codes and Standards" of 10 CFR Part 50, Sec. 50.55a were not in effect. Consequently, there is no compliance with those rules. It may be added that a system with a pressure-temperature rating of 410 kPa at 66°C subjects its components to only mild operating conditions.

5.2.1.4 Applicable Code Cases — The only code case interpretations applicable to the LPTR system are associated with the heat exchanger (see Table 5.1-1).

5.2.1.5 Design Transients — No transient analyses have been performed for the reactor coolant system. An examination of the existing design reveals no transient conditions that could lead to mechanical damage. The system contains no quick closing isolation valves nor pressure relief valves. The only pressure sources within the system are centrifugal pumps and the hydrostatic head. The reactor core is contained in an open pool tank. From the above facts it is apparent that no sudden changes in the system pressure would be expected. Temperature transients are similarly minor in nature. Loss of secondary coolant flow results in a temperature rise in the primary system from 30°C (normal core exit temperature) to 52°C (scram). This change requires minutes to take place.

A damaging water hammer is likewise not a credible occurrence with the present manually operated valves. In addition, no operating problems have ever occurred from sudden loss or application of power to the centrifugal pumps.

5.2.1.6 Identification of Active Pumps and Valves – Pipe breaks in the primary coolant system can be isolated only if they occur in the equipment pit area. The primary coolant pump suction line is buried beneath the containment building floor from the reactor pool tank outlet to the equipment pit. This is a pipe run of about 17 m including the holdup tank, see Fig. 5.2-1. There is another pipe run of 6 m in the equipment pit before reaching the first valve (V-13). The primary coolant pump discharge line is similarly situated. Starting from the reactor pool tank inlet, there is a 11 m buried run and a 7.6 m run in the equipment pit before the first valve (V-29). It is evident from this routing that there are long lengths of piping that cannot be isolated. A postulated pipe rupture between the reactor pool tank and either of these two valves V-13 and V-29 would drain the pool. This constitutes a LOCA and is discussed in Sec. 15.2.

Active valves in the primary system that could be employed to isolate a pipe rupture are all manually operated by entry to the equipment pit. These valves are listed in Table 5.2-1.

Table 5.2-1. Active valves that could be used to isolate a pipe break.

Valve No.	Size	Function
V-13	6 in.	Flow control
V-17	6 in.	Main coolant pump suction
V-28	6 in.	Heat exchanger inlet
V-29	6 in.	Heat exchanger outlet
V-34	6 in.	Heat exchanger bypass, NC

5.2.1.7 Protection Against Environmental Factors – The primary coolant system is contained entirely within the reactor containment building. The system components are either buried 2.7 m deep beneath the floor or located in the equipment pit, see Fig. 5.1-2. The equipment pit is covered by an 0.3 m thick reinforced concrete slab. This configuration provides protection against any credible missiles originating within or outside the containment building. Missiles generated by failure of rotating equipment in the pit external of the coolant boundary possess insufficient energy to penetrate the coolant system. The primary pump impeller could rupture the pump housing if it disintegrated.

CONTINUED BLUE WALL

EQUIP. PIT WALL

Fig. 5.2-1. LPTR process piping and component schematic.

Fires originating within the equipment pit are the only ones that could conceivably damage the reactor coolant system. Protection against damage in this area would be provided by conventional firefighting techniques.

Flooding of the equipment pit could occur from a primary or secondary coolant system pipe rupture. The only protection against flooding is a $2.5 \text{ dm}^3/\text{s}$ sump pump installed in the equipment pit. This pump is adequate for small leaks but large leaks in the primary system would ultimately drain the pool and half of the holdup tank and fill the equipment pit to a depth of 0.72 m. Since the main coolant pump motor is only slightly above this level, it is possible that it would suffer some water damage. However, the motor would have been automatically shut off before this time. A secondary line rupture that drained the entire secondary system would fill the equipment pit to a depth of 1.2 m. This would definitely submerge the main coolant pump. Regardless of the depth of flooding the only damage to the reactor coolant system is water damage to the pump motor.

5.2.2 Overpressurization Protection

The primary coolant system contains no pressure relief valves. The system is not a closed loop, being open to the atmosphere at the reactor pool surface. Therefore, the system cannot develop pressure in excess of the primary pump shutoff head plus static pressure. This amounts to a maximum of 390 kPa. The lowest pressure rated component in the system, the heat exchanger, is rated for 410 kPa and 66°C. Consequently, adequate overpressurization protection is provided by the system design.

The secondary coolant system is capable of being subjected to a pressure of 680 kPa (660 kPa pump shutoff head plus 22 kPa elevation head). All components in the secondary coolant system are pressure rated in excess of this value with the exception of the heat exchanger. The pressure rating of the heat exchanger is 410 kPa at 66°C. The overpressurization protection for it consists of locking in the fully open position the two valves (65 and 154) between the heat exchanger and the cooling tower, see Figs. 5.1-1 and 5.2-1. With this arrangement the heat exchanger cannot be subjected to the pump shutoff head. The highest pressure the heat exchanger can now experience is 320 kPa when all secondary pumps are in operation and the flow control valve 97 is fully open.

5.2.3 General Material Considerations

5.2.3.1 Material Specifications — The specifications for components within the RCPB are shown on Tables 5.1-1 through 5.1-4. The information that was available, in most cases, failed to identify the material by ASTM or any other nationally recognized

standard. In view of the reactor's construction date (1955-1957) and the relatively low pressure-temperature conditions (410 kPa and 66°C) this was not unexpected.

5.2.3.2 Compatibility with Reactor Coolant - The reactor pool tank and its internals are fabricated from aluminum as is the primary system piping. In the piping system there are a number of ferrosteeel and cast iron valves, a nickel plated pump casing, a stainless steel heat exchanger, pump impeller and valves.

RCPB materials of construction are listed below:

Aluminum alloy	5052-F
Aluminum alloy	1100-F
Aluminum alloy	6061-T6
Aluminum alloy	B214 (sand casting)
Aluminum alloy	3003-F
Aluminum alloy	6063-T6
Stainless steel	Type 304
Cast iron	
Nickel plated cast iron	
Ferrosteeel	
Neoprene	

The aluminum alloys exhibit negligible corrosion rated under LPTR conditions of service. The water temperature, velocity, chemistry, and conductivity are such that corrosion problems are minimal. Water leaks that have been discovered appear, from all evidence, to be the result of corrosion that has penetrated from the outside surface, see Sec. 5.2.3.3.

The stainless steel components have, to this date, been trouble free. They all have small areas relative to the aluminum area, except for the heat exchanger. Hence, galvanic effects would be small. The heat exchanger is electrically isolated from the piping system to prevent galvanic effects. Type 304 stainless steel has excellent corrosion resistance to high purity water but may suffer from crevice and stress corrosion. As mentioned above, no problems have been noted.

The corrosion resistance for the cast iron and ferrosteeel components is fair to good. The maximum corrosion rate is most likely to be around 0.15 mm per year. With the relatively massive wall thicknesses involved in these components, overall corrosion will not be a problem.

The nickel plating on the cast iron pump casing has excellent corrosion resistance to high purity water.

The neoprene lining is found in the magnetic flow meter in the equipment pit. The radiation levels in this area are about 20 mrad/h; not enough to adversely effect the neoprene's excellent corrosion resistance for high purity water.

5.2.3.3 Compatibility with External Boundaries — The reactor pool tank is cast in magnetite concrete. In the presence of moisture corrosive attack takes place on the aluminum surfaces. Tests performed by Reynolds Metal Company on various aluminum alloys indicated that 5052 aluminum offered superior corrosion resistance. Consequently, the reactor tank was fabricated from that alloy. As further protection, all aluminum surfaces in contact with the concrete were given two coats of Dupont RA-190 Alkyd resin lacquer, each at least 25 μm thick.

LPTR experience thus far indicates that the pool tank itself has suffered minimal corrosion. However, corrosion has taken place on the aluminum lines attached to the tank and buried in concrete. The corrosion on a number of drain and vent lines has been sufficient to cause leakage of reactor coolant into the concrete. The lines are 6063-T6 aluminum with threaded 6061-T6 fittings except at the pool wall where the threaded fittings are B214 aluminum. The threaded joints were seal welded with an unspecified filler alloy after assembly.

The corrosion appears, from all evidence, to have progressed from the outside surface inward. The proposed explanation is that there existed at the time of initial reactor operation at least one faulty connection that allowed water to enter the concrete.² Approximately four years later the leak first manifested itself by a very slight seepage through the concrete in the East Thermal Column. Thereafter, corrosion started raising bumps on the insides of the instrument thimbles.³ These bumps resulted from the pressure of the confined corrosion products on the tube outside diameter. The leakage was arrested to a degree when in 1970 most of the drain and vent lines were filled with epoxy and sealed. At that time the leak rate dropped from 95 cm^3/s to 350 mm^3/s indicating that these lines were the primary source of leakage. The corrosion still taking place is more of a nuisance than a serious problem.

A second instance of corrosion was noted in 1970. During excavation beneath the containment building floor a 6061-T6 aluminum elbow on the primary system showed corrosion on its outer surface. The line is embedded within the soil and has a bituminous coating. The coating had been damaged during the original backfilling operation and corrosion was confined to the exposed area. The elbow was replaced, recoated, and a pair of anodes attached. One anode is aluminum alloy KA-46 and the second is magnesium alloy H. No further problems have been noted.

5.2.3.4 Chemistry of Reactor Coolant — Control of reactor coolant chemistry is limited to the use of deionized water for make-up and the periodic use of the primary

purification system. The latter consists of a cation exchanger and a mixed bed exchanger. The effectiveness of this treatment is monitored hourly by measuring the water's electrical resistance.

Coolant quality is also checked periodically by pH measurements. The pH is normally around 6.9.

5.2.4 Fracture Toughness

Not applicable.

5.2.5 Austenitic Stainless Steel

The heat exchanger, a few small valves and a portion of the magnetic flow meter are fabricated from type 304 stainless steel. There are no other austenitic stainless steel parts in the RCPB. These components have received no heat treatment to remove carbide precipitates. The corrosion resistance in the weld effected zones of the heat exchanger and flow meter is therefore reduced. However, the high resistivity and purity of the water enables us to use these materials in the as-welded condition without any corrosion problems.

5.2.6 Pump Flywheels

Not applicable.

5.2.7 RCPB Leakage Detection Systems

Significant primary water losses can be detected by observing the reactor pool tank level. Several ways of doing this include:

- A manometer in the control room
- Actual measurements taken at top of reactor pool tank
- A low level float switch 1.1 m below top of pool tank.

The float switch alarms in the control room, scrams the reactor and shuts off the main coolant pump.

Normal water loss rate records come from measurements taken at the top of the pool during the week. Normal water loss is $\sim 300 \text{ mm}^3/\text{s}$ due to evaporation from the surface of the pool.

No maximum allowable loss rates have been established. The pool level is kept between 0.33 m and 0.46 m from the top and usually requires filling about once in three weeks. Plant demineralized water is added at the reactor pool tank top. The fill line

stops short of the pool water surface, avoiding any back flow contamination problem. Water can be added at $0.5\text{--}0.6\text{ dm}^3/\text{s}$.

Large water losses from the RCPB above floor elevation would run into drains on the containment building floor and then into two 5.7 m^3 waste tanks in the equipment pit. Water can be reprocessed through filters and ion exchange columns and returned to the reactor pool tank. The system is arranged for automatic operation. If pool water level drops 76 mm below normal a float switch starts a pump (P-5) in equipment pit, pumping water from the waste tank, through the purification system and into the main coolant line on the inlet side of the main coolant pump (P-1). The system will process 150 m^3 at a flow rate of $0.6\text{--}1\text{ dm}^3/\text{s}$, before a demineralizer regeneration is necessary.

Underground water losses present a more difficult problem. The pool water could be drained into the 38 m^3 storage tank (TK-1) to reduce static head pressure if a leak occurred but this would present radiation problems in the building if the reactor core were still loaded with fuel elements. See Sec. 15.2 for analysis of the radiation hazard.

Water losses to the equipment pit from the reactor primary coolant system can be detected through a sump high level alarm in control room. All water on the equipment pit floor runs to the sump, which alarms at 0.15 m^3 – equivalent to 50 mm pool water level loss.

All these leakage detection system parameters are condensed in Table 5.2-2.

Table 5.2-2. RCPB leakage detection.

Detection instrument	Set point	Remarks	Automatic action
Manometer in control room	None	Not on reactor operator's hourly log, read frequently.	None
Visual indication	NA	No reliability, requires entry to containment building.	None
Reactor pool tank float switch	610–740 mm below normal pool water level	$\sim 2\text{ m}^3$ coolant loss before alarm.	Alarms in control room, scrams reactor and shuts off primary coolant pump
Sump float switch	150 dm^3 or 50 mm drop in pool water level	Requires leakage to drain to sump before action can be taken.	Alarms in control room

5.2.8 Inservice Inspection

With the exception of the buried portions of the reactor coolant system all sections are accessible for inspection. The equipment pit, which houses most of the coolant system, may be entered at any time. Likewise, the reactor pool tank may be visually inspected from above at any time. The only formal inspection program is the day-to-day check by the operators in the course of their start-up routine.

5.3 Thermal Hydraulic System Design

The reactor coolant system for the LPTR is relatively simple. The primary coolant loop removes heat from the reactor core and transfers it to the secondary coolant loop by means of the heat exchanger. A forced draft cooling tower is used to dissipate the secondary coolant loop's heat load to the atmosphere. There is no boiling connected with any heat transfer surface and both loops are open to the atmosphere.

5.3.1 Primary Coolant System

The underground portion of the primary loop was sized for 5.0 MW operation. Consequently, design calculations contend with two differently sized systems as far as the underground permanently installed piping is concerned. All accessible piping is sized to handle heat capacities for the present 3 MW operation of the reactor. The possibility of later power increases was resolved by the installation of 8 in. lines underground and 6 in. lines in accessible sections of the primary loop.

A $67 \text{ dm}^3/\text{s}$ flow rate, established to yield a reasonable temperature drop through the heat exchanger, sets conditions around which the piping system is designed. At this flow rate linear velocity in the 6 in. piping will be 3.58 m/s, in the 8 in. pipe, 2.1 m/s. The 8 in. underground piping was sized by anticipating certain operating conditions at five megawatt operation. It was assumed that a flow rate of $95 \text{ dm}^3/\text{s}$ and a linear velocity in the 8 in. piping of 2.93 m/s would be satisfactory.

Calculating pressure drops for the primary loop for $67 \text{ dm}^3/\text{s}$ yields the results presented in Table 5.3-1.

The head losses in Table 5.3-1 were calculated from the standard Fanning equation for flow in uniform circular channels.⁴ Friction factors in straight runs of pipe are calculated to be 0.0042 for 8 in. lines and 0.0040 for 6 in. lines. Enlargement and contraction head losses are calculated from the same reference. Head losses for fittings and valves are taken from published data based on experimental results.⁴

Table 5.3-1. Pressure drops in the primary coolant loop for $67 \text{ dm}^3/\text{s}$ flow.

Section	Head loss kPa
Contraction loss at reactor tank outlet	1.0
Reactor tank to basement wall	7.06
Basement wall to pump	21.2
Pump to heat exchanger	22.4
Heat exchanger to basement wall	21.5
Basement wall to manifold	5.05
Manifold to reactor tank	3.35
Expansion loss in reactor tank	8.41
Through holdup tank	3.35
Through heat exchanger	50.0
Total loop friction head loss	143.0

From the above head loss calculations operating conditions for the primary loop pump were established as follows:

Flow rate	$67 \text{ dm}^3/\text{s}$ at 41°C
Available net positive suction head (NPSH)	189 kPa
Discharge head	258 kPa
Suction head	97.8 kPa
Total head	163 kPa

The discharge head is the sum of the pump discharge side friction head losses, a 141 kPa static head and 33.1 kPa velocity head (based on 4 in. pump discharge). The suction head is the algebraic sum of the pump suction side head losses, a 141 kPa static head and a 13.4 kPa velocity head (based on 5 in. pump suction). The net positive suction head (NPSH) is the sum of the suction head and atmospheric pressure minus the vapor pressure at 41°C (7.5 kPa).

The primary pump selected and installed is rated for 170 kPa total head at $67 \text{ dm}^3/\text{s}$ and requires a NPSH of 45 kPa. The correspondence between the calculated total head required and that available from the pump performance curve is good. The available NPSH is far in excess of that required and no cavitation problems or operating restrictions are anticipated.

The primary loop flow rate is constant at $67 \text{ dm}^3/\text{s}$ or 67 kg/s . The temperature differential across the primary side of the heat exchanger is maintained at 10.7°C with the primary inlet temperature normally held at 41°C . This results in a nominal heat exchange of 3 MW.

5.3.2 Secondary Coolant System

Treated secondary cooling water from the cooling tower enters the equipment pit from outside the reactor containment building through a 12 in. pipe running under the reactor room floor. A schematic of this system is shown in Fig. 5.2-1. Upon entering the equipment pit, the pipe size is reduced to 8 in. and runs to the tube side of the heat exchanger E-1. The 8 in. line leaves the heat exchanger and returns to the equipment pit wall where it enlarges back to the 12 in. size. The underground 12 in. piping provides for a possible increase to 5 MW reactor operation.

The secondary loop was designed to supply secondary cooling water to the tube side of the heat exchanger E-1 at a rate of $88 \text{ dm}^3/\text{s}$ and at an inlet temperature of 26°C . At present, secondary flow rates up to $179 \text{ dm}^3/\text{s}$ are sometimes required. Using this latter figure, flow velocities are 5.57 m/s in the 8 in. piping and 2.46 m/s in the 12 in. section.

Permanently installed underground piping is sized to accommodate high flow rates for reactor operation at 5 MW. For such a condition, secondary flow would be increased to $300 \text{ dm}^3/\text{s}$ entering the heat exchanger at 26°C and leaving at 31°C . Heat exchanger capacity would have to be increased proportionately. Flow velocity in the 12 in. piping would be 4.18 m/s .

Operating under 3.0 MW and $88 \text{ dm}^3/\text{s}$ conditions produces the head losses presented in Table 5.3-2.

Table 5.3-2. Pressure drops in the secondary coolant loop for $88 \text{ dm}^3/\text{s}$ flow.

Section	Frictional head loss kPa
Cooling tower to heat exchanger inlet (less flow control valve)	41.9
Heat exchanger	35.3
Flow orifice	14.4
Heat exchanger exit to cooling tower	9.0
Total	100

There is an additional static head loss of 72 kPa due to the difference in elevation between the water level in the cooling tower basin and the top distribution basin. At $88 \text{ dm}^3/\text{s}$ there is approximately a 300 kPa head loss caused by throttling in the flow control valve (97).

When operating at $88 \text{ dm}^3/\text{s}$ or 88 kg/s , the temperature differential across the secondary side of the heat exchanger is 8.1°C .

5.3.3 Heat Exchanger

The stainless steel heat exchanger is of the tube and shell type. It has a two pass shell (primary side) and four pass tube section (secondary side). The unit's design conditions are listed in Table 5.3-3.

Table 5.3-3. Design conditions for the stainless steel heat exchanger at 2 MW heat load.

	Flow rate dm^3/s	Temperature $^\circ\text{C}$	Heat transfer coefficient ^a $\text{kW}/^\circ\text{K}$
Shell side	63		3.21^b
Inlet		41	
Outlet		32	
Tube side	88		5.51^b
Inlet		26	
Outlet		32	
Log mean temperature difference (Corrected)		7.1^a $(6.8)^a$	
Overall			2.03

^aManufacturer's calculations.

^bFilm coefficient.

The observed characteristics in December 1973 are listed in Table 5.3-4. From a thermal and hydraulic standpoint the present heat exchanger functions satisfactorily.

5.3.4 Thermal and Hydraulic Characteristics Summary

The primary system of the LPTR operates at a constant flow rate of $67 \text{ dm}^3/\text{s}$. It is possible to operate at a higher flow rate by by-passing the heat exchanger; however, in practice this has never been done. A lesser flow rate could result from partially closed valves, increased flow friction or declining pump performance. This has not occurred during past operation. Therefore, the normal primary flow rate is $67 \text{ dm}^3/\text{s}$ and this covers 100% of the operating time.

Concomitant with the above characteristic is the fact that the LPTR operates almost exclusively at 3 MW with a core outlet temperature of 41°C (which incidently is also the

Table 5.3-4. Operating characteristics for the stainless steel heat exchanger at 3 MW heat load, observed in December 1973.

	Flow rate dm ³ /s	Temperature °C	Heat transfer coefficient kW/m ² .K
Shell side	67		
Inlet		39	
Outlet		28.6	
Tube side	100		
Inlet		22	
Outlet		29.4	
Log mean temperature difference (corr.)		(8.1)	
Overall			2.02

primary water inlet temperature to the heat exchanger). The usual departure from the above conditions is when the outlet temperature is allowed to exceed 41°C because of a cooling tower heat dissipation limitation. This occurs a few days of the year during the summer when there is a high wet bulb temperature (see Sec. 5.1 for further discussion). With the primary water inlet temperature to the heat exchanger and the power fixed, the primary outlet temperature from the heat exchanger becomes 30°C.

The secondary system's flow rate is varied to hold the 41°C core exit temperature. The secondary water inlet temperature is greatly influenced by ambient conditions at the cooling tower and, therefore, varies considerably. There is no normal inlet temperature although it seldom exceeds 25°C or drops below 16°C. The flow rate corresponding to an inlet temperature of 25°C is 177 dm³/s.

Using the above conditions the reactor coolant system would exhibit the characteristics shown on Table 5.3-5.

Table 5.3-5. Thermal and hydraulic parameter summary.

Power	3.0 MW
Primary system flow rate (max)	67 dm³/s
Secondary system flow rate	177 dm³/s
Primary system flow velocity	
8 in. underground piping	2.1 m/s
6 in. piping in equipment pit	3.59 m/s
Secondary system flow velocity	
12 in. underground piping	2.47 m/s
8 in. piping in equipment pit	5.58 m/s
Reactor pool temperature	30°C
Core entrance temperature	30°C
Core exit temperature	41°C
Heat exchanger inlet temperature	
Primary	41°C
Secondary	25°C
Heat exchanger exit temperature	
Primary	30°C
Secondary	29°C
Primary system pressure	
Core exit	75 kPa
Pump inlet	89 kPa
Pump exit	255 kPa
Heat exchanger inlet	206 kPa
Heat exchanger exit	158 kPa
Secondary system pressure	
Pump discharge	450 kPa
Heat exchanger inlet	320 kPa
Heat exchanger exit	170 kPa

5.4 Reactor Pool Tank and Appurtenances

5.4.1 Summary Description

The reactor pool tank is used to house the core of the reactor, direct the flow of cooling water, contain the demineralized water shielding, and act as the inner liner for the concrete shield. It also provides access nozzles through which various experimental and instrument thimbles protrude into the tank from the concrete shield. These components are described in detail in Sec. 4.2.2, reactor internals.

The tank is penetrated by six beam tubes, one through tube, four irradiation tubes, one pneumatic tube, two thermal columns, and five instrument thimbles, all terminating in close proximity to the core face and all welded or gasketed to maintain a leak-proof tank, see Fig. 5.4-1. In addition, the tank bottom is penetrated by an 8-in. Sch. 40 coolant outlet, four 4-in. Sch. 40 inlets, and a 3 in. Sch. 40 drain line. The grid plate support is welded to the tank bottom. A circumferential rack for storage of spent fuel elements is located about 2.9 m above the bottom of the tank. The rack provides space for 40 fuel and reflector elements in a noncritical configuration. It can be used for convenient temporary storage during core assembly and disassembly, or for extended storage of spent fuel elements permitting them to cool before transfer for shipment to the reprocessing facility.

Fig. 5.4-1. LPTR pool tank.

Aluminum plates 6.3 mm thick, removable in sections, cover the top of the tank. The top plate sections merely form a working platform, not a pressure seal. The plates rest on the top of the reactor shield and on cantilevered beams that support the control rod actuator housing. Access to the pool is obtained by removing one or more sections of the top plate.

5.4.2 Tank and Welds

The tank is 8.08 m in overall height, 2 m i.d. with a 9.53 mm wall and bottom thickness. The maximum operating water depth is 7.8 m and the tank contains roughly 25 m³ of water. The maximum internal pressure in the reactor pool is simply the hydrostatic head, 75 kPa. Surface temperature of the tank wall varies depending upon the distance from the core; however, the measured maximum temperature of the vessel is below 38°C.

The original LPTR contract specifications called for the reactor pool tank to be designed, where applicable, in accordance with the ASME Unfired Pressure Vessel Code, Sec. VIII, 1952 edition. However, the tank has an operating pressure of less than 103 kPa and therefore is outside the jurisdiction of Sec. VIII. Fabrication details available on the construction drawings neither confirm nor deny that Sec. VIII design practice was followed.

The design calculations indicate the discontinuity stresses in the bottom-shell junction have been investigated. Superposing the discontinuity and pressure stresses at this point yields the maximum stress in the pool tank.

Calculations⁵ indicate that for a cylindrical tank with a built-in bottom subjected to liquid pressure the maximum stress at the junction is 30 MPa. The maximum allowable stress value for 5052-0 aluminum is 43 MPa, according to Sec. VIII. Therefore the tank is presumed safe. The completed tank was subjected to a field hydrostatic test by merely filling it with water prior to pouring the concrete shield.

During normal operation there are virtually no mechanical stresses in the tank due to external containment by the concrete shield. Since the concrete restrains the radial expansion of the tank, however, there will be thermal stresses. The temperature rise hypothesized under the most extreme conditions during a reactor shutdown is 28°C. The maximum thermal stress for a 28°C temperature rise, assuming the concrete does not expand, is 29.3 MPa.

The reactor pool tank is fabricated from 5052-F aluminum. The choice of alloy was the result of tests performed by Reynolds Metals Company for the Foster Wheeler

Corporation. The intent of the tests was to determine the corrosion rates for aluminum alloys in contact with magnetite aggregate concrete. The results were:

1. Aluminum alloys 3003, 5052, and 6063 cast in magnetite aggregate concrete and maintained dry are subject to only slight attack during the setting period.
2. The above mentioned alloys cast into magnetite aggregate concrete and maintained wet are subject to more aggravated attack.

The entire tank and the outer thermal column enclosures are fabricated from aluminum alloy 5052-F. All connections and members welded to the shell are made of 5052-F or some alloy that is compatible with it insofar as obtaining sound ductile welds. Screens covering the bottom inlet and outlet connections are manufactured from 14 gauge $\times 5/8$ in. wire mesh made from aluminum alloy 5052-F.

Upon the recommendation of the Reynolds Metals Company all aluminum surfaces that would come into contact with concrete were given two protective coats of Dupont RA-190 alkyd resin lacquer. Each coating was at least 25 μ m thick.

The entire pool tank, thermal column enclosures and attachments to the tank are welded by the inert-gas shielded metal arc process with tungsten electrode or the metal arc welding process.

These processes were selected because they require no flux. Therefore, difficult-to-reach places or completely inaccessible interiors of welded assemblies have no flux contamination to act as a potential source of corrosion. Sound ductile welds can be obtained by these processes and there is a minimum of distortion of the metal near the weld.

All welders were qualified in accordance with the provisions of the ASME Boiler and Pressure Vessel Code 1953, Sec. IX. See Table 5.4-1 for summary.

5.4.3 Thermal Columns

The thermal column inner enclosures that project into the pool tank are supported by a frame that is welded to the tank bottom and wall. The thermal column supports are necessary to position the two thermal columns exactly opposite the reactor core and at the same time support the heavy lead nose of the thermal columns. These supports are designed with emphasis on rigidity. For the existing supports the maximum stress is under 3.44 MPa. Five vertical 76 mm $\times$ 38 mm $\times$ 9.5 mm channels with a 19 mm thick top plate form the platform for the lead nose section of each thermal column. This platform is stabilized by five 51 mm $\times$ 51 mm $\times$ 9.5 mm angles extending radially outward and welded to the pool tank wall. Each support is independent of the thermal column inner enclosure to permit thermal column alignment.

Table 5.4-1. Reactor pool tank design.

Applicable code	ASME Sec. VIII 1952 Edition (see description)
Tank o.d.	2.03 m
Tank height	8.068 m
Tank wall thickness	9.52 mm
Tank bottom thickness	9.52 mm
Tank wall and bottom material	5052-F aluminum
Tank capacity	25 m ³
Pressure — design	76 kPa
— operating	76 kPa
Temperature — design	93°C
— operating	30°C
Hydrostatic stresses — design	30 MPa
— maximum allowable	43 MPa
Thermal stresses — operating	29.3 MPa
	28°C ΔT
Nozzle material (in magnetite aggregate concrete)	3003-H12 aluminum
Nozzle material (in conventional concrete)	6061-T6 and 6063-T6 aluminum
Flange material	5052-F

5.4.4 Spent Fuel Storage Rack

The spent fuel element storage rack is used to store up to 40 spent elements during their period of radioactive decay prior to removing them from the tank. The rack is mounted along the inside of the tank under 4.8 m of water.

The rack is in the shape of 210° segment whose diameter is the same as that of the tank. In this shape the rack cannot possibly form a critical array of the fuel elements. The rack is composed of a flat bar 114 mm wide by 19 mm thick with holes drilled in it the same as the grid plate to accommodate the nose sections of the fuel elements. Thirteen inches above the bar are guide boxes. These guide boxes give positive assurance that the stored elements will not topple out of the rack. The guide boxes are made from sheet 4 mm thick. The cross section of each box is 114 mm long by approximately 94 mm wide and the depth is 114 mm.

The flat bar is fabricated from aluminum alloy 5052-F and is stitch welded to the inside of the pool tank. The boxes are fabricated from aluminum alloy 1100-0 bent plate, also welded to the inside of the tank.

5.4.5 Thermal Shield

The high level of gamma radiation in the region of the reactor core not surrounded by the two thermal columns can induce appreciable thermal stresses in the concrete shield. Internal generation of heat in the concrete due to captured gamma rays must be held to a minimum. For this reason lead is clad on the outside of the Reactor Pool tank between the two thermal columns enclosures. The cladding is applied in two 90° segments, one on the north and one on the south side of the tank. Each clad section is in the form of a belt 64 mm thick, starting 0.3 m from the bottom of the tank and extending upward for 1.8 m. The belted sections are framed top and bottom for support by 9.5 mm bent aluminum plates and by 9.5 mm thick bars on the sides that butt against the thermal column enclosures. On each side of the tank the lead is joined to the aluminum tank with a 25 μ m thick coating of a nickel-phosphorus compound in order to obtain a good heat transfer bond.

The heat transfer calculations for the lead belt area indicate that even without a good heat transfer bond there is a negligible temperature rise due to gamma heating.

5.4.6 Experiment and Instrument Tubes

The through-tube and beam-tube nozzles penetrate the pool tank shell horizontally at distances between 1.1 m and 1.4 m above the tank bottom. The nozzles vary from 6 in. Sch. 40 pipe to 400 mm outside diameter by 9.5 mm wall tube. They are double fillet welded to the tank wall and are provided with flanges. The above facilities are oriented either radially or tangent to the reactor core.

There are five additional nozzles for the ion chambers. They are aimed at the area just under the core between 460 and 760 mm above the tank bottom. The nozzles are 5 in. Sch. 40 pipe double fillet welded to the tank wall.

The remaining irradiation facilities penetrate the tank wall at elevations between 3.1 m and 3.7 m from the tank bottom. They are angled downward toward the reactor core. The nozzles in this case are 5 in. Sch. 40 pipe or 76 mm o.d. by 1.7 mm wall tube. A circular plate, like a large washer, is slipped over the pipe or tube and fillet welded to it and to the tank shell to form the seal.

5.5 Component and Subsystem Design

The Foster Wheeler "Design Analysis Manual" is the principal source of design information on the LPTR coolant systems. Experience to date indicates that the vast majority of components have performed satisfactorily from a safety standpoint. Component or subsystem weaknesses have been corrected or eliminated so that the present system configuration provides adequate safety.

5.5.1 Reactor Coolant Pump and Piping

The reactor coolant pump is a standard centrifugal type pump. Its maximum allowable working pressure is 1.21 MPa at 66°C. The operating pressure is 255 kPa at 40°C. The pump is described in Table 5.1-1. It is located in the equipment pit where the

Table 5.1-1. Primary coolant purification system (demineralizer).

Component	Description	Rating	Material
Demineralizer inlet filter	Cuno MK-126, Series 1B2G, 3/4 IPS	1.0 MPa, 1 μ m	Stainless steel housing, fiber filter element
Cation unit (D-1)	Ilco MR-590 with C-211 resin	520 kPa, 1 dm ³ /s 57°C	Stainless steel shell
Mixed bed units (D-2, -3, -4)	Ilco MB-590 with C-211 cation and A-244 anion resins	520 kPa, 1 dm ³ /s 57°C	Carbon steel shell, rubber lined
Demineralizer outlet filter	Fulflo BRX 10 - 3/4 IPS	1.0 MPa, 20 μ m	Stainless steel housing, fiber filter element
Pipe and fittings	Sch. 80	1.65 MPa at 24°C	Rigid PVC
	Sch. 40	10.0 MPa at 37°C	304 SS
	Sch. 40	5.2 MPa at 37°C	6061-T6 aluminum
Valves		861 kPa at 24°C	Type I, PVC

the radiation exposure rate seldom exceeds 20 mR/h. The pump is accessible for inspection at all times. The 15 kW electric motor that drives the pump is interlocked with the reactor pool tank water level float switch. If the water level drops to a point 1.22 m from the top, the reactor scrams and power to the pump motor is removed. This prevents overspeeding of the pump in the event of a LOCA.

The reactor coolant system components, like the reactor coolant pump, are common, off-the-shelf parts with no extraordinary design features or performance requirements.

Except for those that are buried, they are visually inspectable and in low radiation exposure areas. They are typical of systems designed to the code of their time, ASA B31.1-1955, Code for Pressure Piping, for installation in a high purity water system operating at low pressure and temperature.

5.5.2 Reactor Coolant Cleanup System

A by-pass stream taken from the primary coolant system is used as a means of decontaminating the water of ionic impurities that might accumulate either as a result of chemical or radiation corrosion of the piping and equipment in contact with the coolant. A schematic of this system is shown in Fig. 5.5-1.

The water first passes through a pair of 1- μ m filters in parallel and then through a cation exchanger and a mixed-bed anion exchanger. The cation exchanger removes most of the radioactive contaminants, including sodium produced in the $^{27}\text{Al} (n,\alpha) ^{24}\text{Na}$ reaction, relieving much of the burden from the mixed-bed exchanger. This protects the mixed-bed anion exchange resin, which is more easily damaged by excessive radiation. Finally, a second filter unit removes any small particles of exchanger resin from the stream.

Water may be pumped through the ion exchangers by the primary coolant pump (P-1) or by a second pump (P-6) that draws from the top of the pool. At present the demineralizer system is not operated during 3 MW operation in order to minimize ^{24}Na radiation damage to the resin. The system is normally started up 24 h after reactor shutdown and continues to run until reactor startup. Flow rate is 0.56 dm³/s. Water on a Monday startup will have a specific resistance of 50 k Ω -m and will decrease to about half that amount during a normal (105-110 h) week. A 36 h cleanup during shutdown will again bring it back to the 50 k Ω -m level. During reactor operation a bypass flow of 0.6 dm³/s passes through a 1- μ m filter.

Since most of the radioactivity is concentrated in the cation unit, the container and connecting piping for this unit are fabricated from Type 316 stainless steel rather than plastic-lined carbon steel or other materials. This affords good radiation and chemical resistance where it is most necessary. A standard rubber-lined carbon steel vessel is specified for the mixed bed unit since it is exposed to much lower levels of radioactivity. Interconnection pipe is plastic and stainless steel.

Fig. 5.5-1. Primary coolant purification system (demineralizer).

REFERENCES

1. Definition taken from ASME code NB 3113, Sec. 3.
2. C. W. Garrard and I. D. Randall, A Survey of Pool Linings for Swimming Pool Reactors, Texas Agricultural and Mechanical University, College Station, Texas, Technical Rept. 15 (1963).
3. G. E. Cummings, Instrument Thimble Corrosion Problems at the LPTR, Lawrence Livermore Laboratory, Rept. UCRL-12477 (1965).
4. Chemical Engineer's Handbook, J. H. Perry, Ed. (McGraw-Hill, New York, 1950) ed. 3, pp. 377, 388.
5. S. Timoshenko, Theory of Plates and Shells (McGraw-Hill, New York, 1959) ed. 2, p. 410.

CHAPTER 6.0 ENGINEERED SAFETY FEATURES

6.1 General

The engineered safety features provided for the LPTR are the containment system and the emergency core cooling system (ECCS).

6.1.1 The Containment System

The containment system consists of a controlled-leakage steel building and filter systems designed to limit release of radioactive effluents. The building is a cylindrical shell of welded steel plates 6.35 mm thick (reinforced with stiffener angles) with a domed, self-supporting roof of 7.94 mm steel plates. The base of this structure is welded to large anchor bars buried in a massive reinforced-concrete ring wall. The building is 24 m in diameter, 16 m high, with a working floor area of nearly 420 m². Volume is 6260 m³. Section and plan views of the building are shown in Figs. 3.8-20 and 3.8-21.

Connected to the building shell is a decontamination system capable of filtering the entire air volume of the building in 2700 s whether the building is pressurized or not. The decontamination fan and filter box are situated outside the containment building so that they may be serviced, if necessary, during a period when the building is radioactively contaminated. The flow of air is from the containment building supply duct, through a bank of HEPA filters and then through the decontamination blower and back into the building. The blower, including fan and motor, is located wholly within the ducting in order to prevent any leakage to the environment.

The building is designed to withstand an internal pressure differential of 14 kPa. However, the containment philosophy is based on maintaining a very small positive pressure in the building. An emergency filter/scrubber system is connected to the building to vent any overpressure. If the building pressure slightly exceeds atmospheric pressure, the emergency filter/scrubber system will exhaust air from the building. This air, after passing through absolute filters and activated charcoal filters, is exhausted through 12 m stacks into the atmosphere essentially free of particulate fission products and the radioiodines. Two separate systems, each with its own filters and scrubber, are connected to the emergency power system. If the building overpressure reaches 0.5 kPa, blowers in the system start automatically to increase the discharge flow rate.

6.1.2 The Emergency Core Cooling System

The emergency core cooling system (ECCS) provides cooling for the core during a loss-of-coolant accident (LOCA). The ECCS uses city water or site deionized water. The system is shown in Fig. 6.1-1. It can be activated from either the control room or the containment building. The only active component in the system is a motor-operated valve.

Fig. 6.1-1 Emergency core cooling system (ECCS) piping and instrument layout.

6.2 Containment Systems

6.2.1 Containment Functional Design

The containment system was originally designed for low pressure containment at 14 kPa internal pressure and a leak rate not in excess of 3% per day. Pressure tests at 14 kPa yielded a leak rate of approximately 10% per day. After extensive

modification the leak rate was reduced to 8% per day. Further efforts to reduce the leak rate were unsuccessful and it was concluded that the main source of leakage was diffusion through the concrete floor.¹ The containment system was then modified to a pressure-venting type to achieve an acceptable leak rate. This type of containment system maintains a given pressure in the building by continuous pressure venting during the course of an accident.

The pressure-venting system is designed to limit the containment building pressure so that an acceptable unfiltered gaseous leak rate is possible. The venting is accomplished by use of the air scrubber and filter system, see Sec. 6.2.3.2.2.

6.2.1.1 Design Basis — The design basis for the containment building structure was the maximum expected pressure rise in the sealed system due to diurnal temperature effects and a concurrent decrease in barometric pressure. A diurnal change in the ambient temperature of 39°C was assumed. This would correspond to a night-time temperature of 4.4°C and a peak midday temperature of 43°C. These temperatures are the maximum daily extremes found in the meteorological records over a 30-y period. Tests conducted on the LPTR containment building indicate that the building temperature variation is always less than 75% of the ambient air temperature variation. With a 28.9°C building air temperature rise after being sealed at 9.4°C and a simultaneous drop in barometric pressure of 3.4 kPa the total effective overpressure would be:

$$P_2 = P_1 \left(\frac{T_2}{T_1} \right) + P_3$$

P_2 = containment building internal pressure, kPa

P_1 = containment building internal pressure at time of building seal = 101.3 kPa

P_3 = barometric pressure change = 3.4 kPa

T_1 = temperature at time of building seal = 282.8 K

T_2 = maximum temperature of air in sealed building = 311.7 K

$$\begin{aligned} P_2 &= 101.3 \left(\frac{311.7}{282.8} \right) + 3.4 \\ &= 115.1 \text{ kPa (absolute)} \\ &= 13.8 \text{ kPa (gauge)} \end{aligned}$$

As stated in Sec. 6.1, the containment system is now operated in a pressure-venting mode. The design basis for the condition is the capacity to vent gas fast enough to

keep the unfiltered gaseous leak rate at ≤ 0.25 w%, per day. Based on the original containment leak tests this would indicate that the average daily containment pressure should be <406 Pa.

The postulated accident condition that results in a pressure rise in the containment is loss of reactor coolant. It is assumed that the reactor has been operating for an infinite period of time. The energy release to the containment building during the accident is assumed to be entirely due to the radioactive decay of the core fission products. The decay heat generation is determined from the proposed ANS Standard ANS 5.1.² Stored energy in the coolant and reactor components is neglected since the system is open to atmospheric pressure. Also, the coolant is at an average temperature of 31°C and the remainder of the containment is at 21°C .

No metal-water reaction takes place. The ECCS is assumed to fail to function. Therefore, the fission product decay heat is transferred to the containment building air, the reactor pool tank and its supporting structure. The containment air is assumed to transfer heat to the containment building concrete foundation and reactor shield. The supposition is that there is perfect mixing of all gases in the containment and the ideal gas law applies. There are no differential pressures within the containment.

Post-accident pressure reduction is accomplished by use of the air scrubber and filter system. The system vents containment pressure without requiring any initiating action. The larger gas flow rates obtainable by using blowers are not essential to the system operation.

6.2.1.2 Containment Structure Design -

6.2.1.2.1 Shell - The containment building is 24 m in diameter by 16 m high. It has a volume of 6260 m^3 . Section and plan views of the building are shown in Figs. 3.8-20 and 3.8-21. The cylindrical shell is fabricated from 6.35 mm thick steel plates with suitable stiffeners. The spherically shaped roof is fabricated from 7.94 mm thick steel plates and is self supporting. The entire structure is of welded construction. The shell footing is a massive reinforced concrete annular ring 76.5 m in circumference. The shell is buried in the footing to a depth of 0.46 m and is welded to numerous anchor bars.

6.2.1.2.2 Containment Building Floor - The building floor is a 0.15 m thick concrete slab, with a 0.10 m curb around entire circumference. An antileak ring is welded to the containment building shell steel below the concrete surface. To facilitate decontamination, the floor is surface hardened and covered with five coats of a strip-pable plastic paint. The floor has several drains and a number of covered pits, including the large equipment pit. Expansion joints are provided in the floor to prevent the

floor slab from buckling or cracking and to isolate any vibrations which might be transmitted to the steel building shell. All construction joints in the concrete floor slab and between concrete floor and walls have a copper water stop, which is soldered to form an unbroken periphery to eliminate leakage into the subsoil. These joints are keyed and have a minimum thickness of 0.2 m.

6.2.1.2.3 Compartments – The containment has a few spaces which can be considered compartments. They include the floor storage pit, beam plug storage, magnet pit and equipment pit.

The floor storage pit consists of thirty 5 in. Sch. 40 aluminum pipes sunk to a depth of 4.0 m in the containment building floor; Sec. 9.1.2. These are capped on the bottom and can be filled with water. Similarly, there are five 6 in. Sch. 40 and one 0.34 m i.d. $\times$ 0.91 m long pipe for temporary or extended storage of other radioactive items. Originally designed for spent-fuel-element storage, this pit now provides storage of the core only during periods when removal of the elements from the tank is required for in-tank maintenance or modifications.

The beam plug storage facility is located on the southeast side of the containment building (Fig. 3.8-21). It is 4.3 m long $\times$ 7.6 m wide $\times$ 2.4 m high and contains 20 horizontal storage tubes of various diameters from 0.12 m to 0.44 m. All the tubes are 3 m long. The structure stands on a concrete footing and has concrete walls on all four sides 0.30 m thick. This concrete "box" is filled with compacted earth. The pipes are covered with at least 1.1 m of earth topped with 0.05 m of mastic sealer. Each tube is closed with a magnetite concrete plug 0.15 m thick in a 6.3 mm steel shell.

The magnet pit is a 8.8 m square and 0.9 m deep hole in the floor. It is covered by metal plates.

The equipment pit is located on the west side of the containment building as shown in Fig. 3.8-21. It is 10.2 m long, 4.0 m wide and 5.0 m deep. It houses the pumps, heat exchanger and demineralizer of the primary loop, waste tanks, air purge filter and blower, pneumatic tube blower and control cabinet, and exposed process piping.

The equipment pit is constructed entirely of reinforced concrete, and is covered with five 0.30 m thick removable concrete slabs set flush with the operating floor. The removable slabs are capable of carrying a uniformly distributed load of 9800 kg/m^2 . Access to the basement is through a hatchway at the north end of the equipment pit. The hatchway is 0.61 m square and is not covered.

For all practical purposes the containment building has no separate compartments. All parts of the containment building experience the same pressure and temperature through unsealed openings.

6.2.1.2.4 Roof Penetrations – The roof has a single 2.4 m diam penetration at its center. A plenum chamber over this opening houses the building exhaust blower and isolation valve. The plenum is 2.4 m in diameter, 1.5 m high and has a domed roof.

Access to the plenum is through a 0.5 m diameter manway reached by a ladder and safety cage from outside the containment building. The bottom of the plenum chamber collects gases from the waste tank vents, air purge blower and air sampling pit as well as the containment building.

For ease of maintaining equipment in the plenum a perforated plate is bolted across the bottom to form a floor. It does not, however, materially impede flow of air from the containment. Inside the plenum there is a vane axial fan for exhausting collected gases. Downstream of the fan the exhaust isolation valve is installed in 6.3 mm thick steel duct work. The isolation valve is a 0.5 m diam butterfly type with 25# ASA flanges. It is operated by a direct coupled air cylinder which is spring loaded to close on loss of air or electrical power. Closure time is less than 2 s. The duct work is welded to the plenum shell and is designed for 14 kPa pressure. After the duct penetrates the plenum shell there is a louvered damper and a bird screen of 13 mm wire mesh hardware cloth.

6.2.1.2.5 Shell Penetrations –

6.2.1.2.5.1 Airlock – Located on the west side of the containment building is a personnel airlock. It is a 2.4 m diam by 3.0 m long cylinder fabricated from 6.3 mm thick steel. The bulkheads at the ends are 9.53 mm thick with doors 1.14 m by 2.1 m high. The lock structure and both doors are designed and constructed for the same pressure as the containment. The doors are interlocked to prevent both being opened simultaneously. An interior lighting system with an emergency power supply is available. Also, there is a communication outlet.

The doors are designed to swing outward from the airlock. This facilitates seating of the interior door if internal pressure exists in the containment building. The door perimeters are sealed with inflatable elastomer gaskets 28.6 mm thick x 85.7 mm wide x 9.5 mm wall. The gaskets are normally deflated and are inflated with 400 kPa compressed air when the containment building is sealed. The building seal operating sequence inflates the gasket of the door that is closed and locked by the control circuit. If the last use of the airlock was to exit from the containment building, then the airlock

interior door is sealed. The exterior door is sealed if the last use was to enter the containment.

6.2.1.2.5.2 Escape Mandoors - There are two escape mandoors located respectively on the north and south side of the containment building. They are similar in construction to the airlock doors except that they are smaller, 0.76 m wide by 1.8 m high. They use the same inflatable gasket design but the inflating pressure is reduced to 140 kPa. The doors open inward so that a positive pressure in the building would help seal them. The doors are in the sealed condition whenever the reactor is operating. To exit the building, the gasket is deflated by using the door control panel. It requires 10-15 s for the gasket to deflate. The door can then be opened manually. After an exit, a commercial door closer installed on each door will close the door but it is necessary to physically seat the door by pulling it. Upon seating the gasket inflates automatically to reseal the door.

6.2.1.2.5.3 Truck Door - The truck door is on the east side of the containment and measures 3.0 m wide by 4.2 m high. It employs an inflatable gasket 38 mm thick by 127 mm wide with a 13 mm wall. Inflation pressure is 140 kPa. The door opens inward and is in the sealed condition whenever the reactor is operating. The door is fabricated from 9.5 mm thick plate with stiffeners on the inside and outside surfaces. Because of its size and weight this door has a supporting wheel on the lower corner opposite the hinge line. This removes some of the load from the hinges and helps to maintain alignment. To seat the door and to crack it open a pneumatic cylinder is used. The door can be swung by hand the rest of its travel distance.

6.2.1.2.6 Materials - The shell and roof are fabricated from ASTM A283C steel. The inside surface is painted with a coat of water base white paint. The outer surface has been covered with a 25-32-mm-thick layer of polyurethane foam. The polyurethane is nonflammable, has a maximum service temperature of 77°C and is covered with a water base exterior fibered paint. The concrete floor is painted with Bisonite M-101 vinyl paint.

There is one known case of material incompatibility under an accident condition. A fire could damage the door inflatable gaskets and cause loss of containment if not extinguished promptly.

6.2.1.3 Design Evaluation -

6.2.1.3.1 Leak Rate Determination - The 0.25 wt% per day unfiltered gaseous leak rate used in the DBA, Sec. 15.4.1, is predicted on the containment system average daily pressure being below 406 Pa. The relationship between gaseous leak rate and containment pressure was derived from the leak tests performed in 1960.¹ The leak rate was determined by the reference vessel method, eliminating the necessity for

precise measurements of absolute pressure and for multiple measurements of temperature.

This method of leakage rate determination depends on the changes in pressure of the containment building air compared with that of hermetically sealed reference vessels. The reference vessels are so placed in the containment building and of such a geometry that they rapidly assume the temperature of the containment air, providing automatic temperature compensation.³

The results of that test indicated a leakage rate of 8.0% per day at 14 kPa differential pressure. This data was extrapolated to meet the DBA conditions. The measured leakage rate was determined to result mainly from leakage through the concrete floor. The actual leakage results from the combination of several types of flow. With the relatively low pressure differential, however, it was considered safe to assume that laminar flow would predominate.

The extrapolation factor was derived on the basis of four simplifying assumptions:

1. All leaks in containment systems behave in accordance with the single mode of flow for which the extrapolation factor is applicable.
2. All leak geometries (area of flow, leak path length, roughness of passage, etc.) remain constant for the pressure and temperature range of interest.
3. The number of leaks in the containment system does not vary with pressure or temperature and is not influenced by the composition of the containment atmosphere (no plugging).
4. The conditions within the containment system remain constant for the time interval of interest, as defined by the leakage rate.

The extrapolation factor analysis starts with the Darcy formula for laminar flow which is of the form

$$w = K(P_a^2 - P_o^2)$$

where

w = mass flow rate of flowing gas for given conditions (kg/s)

K = constant accounting for geometric factors and gas properties (kg/s·kPa)

P_a = containment pressure (kPa)

P_o = barometric pressure outside of containment (kPa).

To simplify the equation the pressures P_a and P_o can be expressed in atmospheres

$$w = K'(P_a^2 - 1).$$

The definition of the leakage rate is

$$L = \frac{w}{W_0},$$

where

$$W_0 = \frac{P_a V}{RT}$$

W_0 = initial mass of containment gas (kg)

V = containment volume (m^3)

T = containment temperature (K)

R = gas constant.

Therefore,

$$L = \frac{wRT}{P_a V}.$$

The extrapolation factor is expressed as a dimensionless ratio of the leakage at the extrapolated conditions to that at the test conditions,

$$\frac{L_e}{L_t} = \frac{w_e \frac{RT}{P_e V}}{w_t \frac{RT}{P_t V}},$$

where the subscripts e and t stand for extrapolated and test condition. Since R , T , and V are constant,

$$\frac{L_e}{L_t} = \frac{w_e P_t}{w_t P_e}.$$

Substituting for w yields

$$\frac{L_e}{L_t} = \frac{(P_e^2 - 1)(P_t)}{(P_t^2 - 1)(P_e)} = \frac{P_e - \frac{1}{P_e}}{P_t - \frac{1}{P_t}},$$

where

$$L_t = 8.0\% \text{ per day}$$

$$P_t = 1.136 \text{ atm}$$

$$P_e = 1.004 \text{ atm}$$

$$L_e = 8 \left(\frac{1.004 - \frac{1}{1.004}}{1.136 - \frac{1}{1.136}} \right)$$

$$= 0.25\% \text{ per day.}$$

6.2.1.3.2 Containment Pressure Vent Rate – The design of the filter/scrubber system (see Sec. 6.2.3.2.1) forms the basis for the containment pressure-venting system. The pressure venting takes place whenever the containment pressure exceeds

atmospheric pressure. The system will operate without the scrubber blowers although their use results in a higher flow rate and hence lower containment pressure. Figure 6.2-1 shows the volume flow rate as a function of containment pressure as determined by experimental measurements in this facility

Fig. 6.2-1 Scrubber system flow rate vs containment pressure above atmospheric. Blowers not operating.

6.2.1.3.3 Energy Release to Containment – The energy release to the containment system during the design basis LOCA is due solely to the radioactive decay of the core fission products. The decay energy is determined from the proposed ANS Std. 5.1, Decay Energy Release Rates Following Shutdown of Uranium Fueled Thermal Reactors.² The fission product decay power for infinite reactor operating time at 3 MW is shown in Fig. 6.2-2. The total heat produced that could be transferred by convection to the containment building air is shown in Fig. 6.2-3. The curves were determined by using the appropriate analytical expressions from ANS Std. 5.1. They are

$$\frac{P}{P_o} = At^{-a},$$

Fig. 6.2-2 Fission product decay power after infinite reactor operating period at 3 MW.

Fig. 6.2-3 Total heat absorbed by containment building air after a loss of coolant accident (LOCA).

where

$\frac{P}{P_0}$ = fraction of operating power

t = cooling time in seconds.

Values of A and a appear in Table 6.2-1.

Table 6.2-1 Values of coefficient and exponent in the analytic expression for pressure change.

Cooling time s	A	a
$10 \leq t \leq 150$	0.0766	0.181
$150 \leq t \leq 4 \times 10^6$	0.130	0.283

The total heat is determined by integrating these expressions.

$$H = AP_0 \int_0^t t^{-a} dt$$

$$H_1 = 0.0766 \times 3000 \int_0^t t^{-0.181} dt$$

$$= 280.6 t^{0.819} \text{ (kJ)}$$

for $10 < t \leq 150$ s.

Similarly

$$H_2 = 0.130 \times 3000 \int_0^t t^{-0.283} dt$$

$$= 544 t^{0.717} \text{ (kJ)}$$

for $150 < t \leq 4$ Ms.

To facilitate the later calculations for containment pressure as a function of time, the heat input term was simplified to:

$$H = 544 t^{0.717} \text{ (kJ)}$$

for $0 < t \leq 4 \text{ Ms.}$

This conservatively overestimates the heat input in the time period below 150 s. Note that the heat up to 68 s is absorbed by the draining pool water. This is also ignored in the calculational model.

6.2.1.3.4 Containment Heat Balance Following a LOCA – As stated previously, it is assumed that there is perfect mixing of all gas in the containment. Therefore, any heat addition to the containment air causes a temperature rise in the entire volume. This follows from the assumption that the pool deck is not in place. This is a conservative approach since it results in the highest containment pressure. If the pool deck were in place (as is normal practice) the core fission product decay heat would be transferred primarily to the reactor shield. This was the case examined in Sec. 15.2.3.1 where the object was to calculate the highest fuel temperature. That analytical approach causes a negligible rise in containment pressure.

The initial temperature at the start of the LOCA is 21°C. This is the usual temperature that the containment building air-conditioning system maintains. The pool water is at 31°C but during a LOCA it would drain to the equipment pit. There its stored energy in excess of 21°C would be transferred through the equipment pit walls into the earth.

Heat absorbed by the containment building air is transferred to various structural components within the building. The only ones used in the analysis are the concrete reactor shield and floor slab. Ignored are the 110 Mg steel shell, the 22.7 Mg steel crane structure and any heat losses through the shell.

a) Heat Losses to Concrete – The 525 Mg reactor shield with its 145 m² vertical surface area is assumed to be an infinite heat sink at 21°C. The same assumption applies to the 467 m² floor area. The natural convection heat transfer coefficients that apply to these surfaces are calculated from equations in Ref. 4, converted to the International System of Units (SI),

$$h_v = 1.31 \theta^{1/3}$$

$$h_H = 1.52 \theta^{1/3}$$

where

$$h_v = \text{convection coefficient for vertical surface (W/m}^2\text{·K)}$$

h_H = convection coefficient for horizontal surface ($W/m^2 \cdot K$)

θ = temperature difference between surface and gas (K).

These equations are for air at atmospheric pressure and turbulent free convection conditions. For the conditions to qualify as turbulent the Grashof-Prandtl number product ($Gr \cdot Pr$), evaluated at film temperature, must be $> 10^9$. The Grashof number is

$$Gr = \frac{g \beta \theta L^3}{\mu},$$

where

g = acceleration of gravity

β = volume coefficient of expansion

θ = defined above

L = vertical or horizontal dimension of surface

μ = kinematic viscosity.

When the film temperature is $21.5^\circ C$ assuming

$$\theta = 1^\circ C$$

$$L = 3.35 \text{ m (minimum vertical or horizontal surface length)}$$

the Grashof number is

$$Gr = 5.4 \times 10^9.$$

At $21.5^\circ C$ the Prandtl number is

$$Pr = 0.71.$$

The Grashof-Prandtl number product is therefore always greater than 10^9 for the surfaces under consideration.

The heat transferred into the concrete is the sum for the vertical and horizontal areas:

$$\begin{aligned} q_{\text{conc}} &= h_v A_v \theta + h_h A_h \theta \\ &= (1.31 \times 145 + 1.52 \times 467) \theta^{4/3} \\ &= 900 \theta^{4/3} \text{ (J/s)} \end{aligned}$$

or

$$Q_{\text{conc}} = 0.9 t \theta^{4/3} \text{ (kJ)}$$

where

$$t = \text{time (s)}.$$

b) Internal Energy Gain — The heat transferred into the containment building air from the core increases the air internal energy. Part of this air is expelled from the building by virtue of the increase in building pressure. Designating the volume of expelled air as ΔV , the increase in internal energy for the respective volumes are

$$\begin{aligned} Q_{\text{CB}} &= MC\phi \\ &= \rho C(V_0 - \Delta V)\phi \end{aligned}$$

and

$$Q_{\text{EX}} = \rho C(\Delta V)\delta$$

where

Q_{CB} = change in internal energy of containment building air (kJ)

Q_{EX} = change in internal energy of air expelled (kJ)

C = specific heat of air at constant pressure

$$= 1.0 \text{ kJ/kg}\cdot\text{K}$$

ρ = mass density of air at 21°C

$$= 1.2 \text{ kg/m}^3$$

$V_0 - \Delta V$ = volume of gas in building (m^3)

ΔV = volume of gas expelled (m^3)

ϕ = change in temperature of gas in building (K)

δ = change in temperature of gas expelled (K).

c) Heat Balance — The fission product decay heat equals the heat transferred to the concrete plus the internal energy change of the containment building air plus the internal energy change in the expelled air.

$$H = Q_{\text{conc}} + Q_{\text{CB}} + Q_{\text{EX}}$$

$$544t^{0.717} = 0.9 t \theta + 1.2(V_0 - \Delta V)\phi + 1.2(\Delta V)\delta$$

d) Containment Building Gas Temperature - The heat balance equation can be solved for the building gas temperature by making the following substitutions:

$$\theta = T_2 - 294 \text{ (K)}$$

$$\phi = T_2 - T_1 \text{ (K)}$$

$$\delta = \frac{T_2 + T_1}{2} - T_1 \text{ (K)}$$

where 294 K is the concrete surface temperature which is assumed constant,

T_2 = temperature of gas at time t_2

T_1 = temperature of gas at time t_1

To simplify the solution for T_2 , the exponent of θ will be reduced to unity. This significantly underestimates the heat transfer to the concrete.

$$T_2 = T_1 \left[\frac{1.2 V_0 - 0.6 \Delta V}{0.9 (t_2 - t_1) + 1.2 V_0 - 0.6 \Delta V} \right] + \frac{544 (t_2^{0.717} - t_1^{0.717}) + 264.6 (t_2 - t_1)}{0.9 (t_2 - t_1) + 1.2 V_0 - 0.6 \Delta V}$$

6.2.1.3.5 Containment Pressure Following a LOCA - The containment pressure can be calculated using the perfect gas law.

$$\frac{P_0 V_0}{T_0} = \frac{P_2 V_2}{T_2}$$

Let

$$V_2 = V_0 + \Delta V$$

$$P_2 = P_0 \left(\frac{T_2}{T_0} \right) \left(\frac{V_0}{V_0 + \Delta V} \right)$$

where T_2 can be found from the last equation in Sec. 6.2.1.3.4. ΔV can be found from Sec. 6.2.1.3.2 and Fig. 6.2-1 by the following method. The volume flow rate is:

$$w = 0.0416 \sqrt{P - 101.33}$$

but

$$u = \Delta V_i / (t_2 - t_1)$$

where

ΔV_i = volume of gas expelled in time period $t_2 - t_1$.

Therefore:

$$\Delta V_i = 0.0416 (t_2 - t_1) \sqrt{P - 101.33}.$$

However, P is not constant over the time period $t_2 - t_1$. Hence ΔV_i must be evaluated at t_2 and t_1 and the average value used. The total volume of gas expelled (ΔV) that is used in the prior equation for P_2 and T_2 is

$$\Delta V = \sum (\Delta V_i).$$

A computer routine was written to solve for ΔV , T_2 and P_2 . The initial conditions are:

V_0 = containment volume = 6258 m³

P_0 = containment pressure = 101.33 kPa

T_0 = containment temperature = 294.44 K

$t_1 = 0$

and

T_1 = temperature at $t_1 = 294.44$ K.

The calculated results at t_2 equals 150 s are:

$P_2 = 102.2$ kPa

$T_2 = 297.0$ K

$\Delta V = 3.0$ m³

The computer routine continues to solve for these items as a function of time. The time steps were arbitrarily selected to be 900 s. In successive time steps, the prior calculated value of T_2 is substituted for T_1 and the prior value of P_2 is used to determine ΔV_i . The results are shown in Figs. 6.2-4 and 6.2-5. The containment pressure is seen to peak early and then return to atmospheric after 14 ks. The containment gas temperature reaches its peak about the time the pressure is back to atmospheric. The total gas expelled (ΔV) is 610 m³.

Fig. 6.2-4 Air pressure in containment building following a LOCA with reactor pool uncovered.

Fig. 6.2-5 Temperature of containment building air following a LOCA with the reactor pool uncovered.

6.2.1.3.6 Summary – The original design basis for the containment structure was a set of nonnuclear circumstances that caused the pressure to rise to 13.8 kPa (Sec. 6.2.1.1). The analysis in Sec. 6.2.1.3.5 shows that for the worst nuclear event the maximum containment pressure is 3.0 kPa.

The DBA analysis (Sec. 15.4) was based on assumptions of 0.25 weight percent per day unfiltered gaseous leak rate and 1630 m³ per day of gas escaping through the filter/scrubber system. Ignoring any pressure changes due to diurnal temperature effects, the gas expelled during a LOCA via the filter/scrubber system is 610 m³. The prior assumption was therefore conservative. The 0.25 wt% leak rate requires an average containment overpressure of 406 Pa or less (see Sec. 6.2.1.3.1). As can be seen from Fig. 6.2-4, the containment pressure is only above atmospheric for 14 ks or <4 h. The integrated leak rate was calculated using the leak rate extrapolation equation and the data in Fig. 6.2-4. On a 24 h basis the leak rate is 0.12% or about half the value assumed for the DBA. Again, the DBA assumption was conservative. The maximum containment air temperature of 323 K should cause no problems.

In conclusion, the containment system is able to meet its design requirements with adequate safety margins.

6.2.1.4 Testing and Inspection - The testing and inspection of the containment system is confined to that associated with its subsystems. Sections 6.2.3.4 and 6.2.4.4 describe the tests performed.

6.2.1.5 Instrumentation Requirements - The instrumentation available for monitoring the containment system is that used for its subsystems. Airborne radioactivity instrumentation is described in Sec. 12.2.4. The filter/scrubber instrumentation as well as that used for the emergency air decontamination system is detailed in Sec. 6.2.3.4.3. The instrumentation for the containment isolation system is found in Secs. 7.3 and 7.5.

6.2.1.6 Materials - The containment structure is constructed of steel and concrete. The thermal insulation of the steel shell is applied on the outer surface and its decomposition would not interfere with the safe operation of the structure.

6.2.2 Containment Heat Removal Systems

Not applicable.

6.2.3 Containment Air Purification and Cleanup System

The containment building has three air purification and clean up systems. The first is the ventilation system which changes the containment building air twice every hour. The second system is the emergency filter/scrubber system. This system cleans the purged air of nongaseous fission products and radioiodines and exhausts the cleansed air from the containment building. The third and final system is the emergency air decontamination system, which recirculates the sealed containment building air through a bank of four absolute filters. The entire containment building air can be recirculated through these filters every 2.7 ks. Figure 3.5-1 shows how air flows through these systems. Air monitoring is discussed in Sec. 12.2.

6.2.3.1 Design Bases -

6.2.3.1.1 Heating, Ventilation, and Air-conditioning - The general design of heating, ventilating, and air-conditioning system conforms to the recommendations of the A. S. H. and V. E. Guide, latest edition:

- a) Building is equipped with air-conditioning ducting capable of normal and emergency usage.
- b) Extreme winter outside condition is -10°C (db) with 6.7 m/s average wind velocity.
- c) Maximum summer outside condition is 40°C (db) and 22°C (wb).
- d) Building shall be heated by heating coils with a value of 41 MJ/m^3 .

- e) Heating calculations are based on approximately two (2) air changes per hour within the breathing level, 0-4.6 m.
- f) Heated or conditioned air is supplied through equally-spaced registers in a duct system concentric with the building.

6.2.3.1.2 Filter/Scrubber System – Tests concluded in November 1958 indicated that the containment building did not meet the 3%/day leakage rate specification. During these tests, leaks were observed in the following areas.

- a) The three dampers did not seal tightly against their gaskets. This was particularly true at relatively low pressure conditions.
- b) The two emergency escape doors leaked air past their gaskets.
- c) The main truck door had very large air leaks past the gasket under low pressure condition, but sealed fairly tight at 14 kPa.
- d) The air lock leaked air through both doors. If the dogs were tightened enough to eliminate these leaks the door was inoperable.
- e) It was noted that air leaked through the floor in several places, particularly cracks and construction joints. These leaks in the floor were observed after the application of caulking compound in all construction joints, and the application of the five coats of Bisonite paint.

During the summer months of 1960, the reactor was shut down and an extensive construction program was carried out to modify the containment system. Upon conclusion of these modifications the building was again tested to determine the leakage rate. Final tests indicated that the building system leaked at a rate of approximately 8%/day at 14 kPa pressure. It has been concluded that the main source of leakage from the static containment system is by diffusion through the concrete floor.

It was decided to convert the static LPTR containment system to a dynamic system. A system was installed to exhaust air from the building as necessary to maintain a very small pressure differential between the building and the outside pressure. The air exhausted by such a system would be passed through a filter/scrubber system before being vented into the atmosphere through a high stack.

6.2.3.1.3 Emergency Air Decontamination System – A system of dampers and ducting is designed to permit air to be recirculated within the sealed building to clear it of radioactive particulates. The flow is out through the air-conditioning duct on the north wall of the building, through a bank of four absolute filters, and in through the duct on the south wall. Clean up of air is thus accompanied by sweeping air through the building from south to north. This system is designed to filter $2.6 \text{ m}^3/\text{s}$, or the equivalent of the entire building volume in 2.7 ks. The filters are almost 100% efficient for particles larger than $0.3 \mu\text{m}$. The equipment for this recirculation system is located just outside the air lock where the filter is housed in a unit specially designed

to facilitate replacement of contaminated filters. The cleanup operation may be performed continuously until a minimum level of activity is achieved.

16.2.3.2 Descriptions ~

6.2.3.2.1 Air Conditioning, Heating, and Ventilation - The air-conditioning unit for the reactor building is located in the laboratory building east of the mechanical equipment room, Room 1328. About 29% ($1.10 \text{ m}^3/\text{s}$) make-up air enters the unit through louvers in the $0.8 \times 0.4 \text{ m}$ fresh air duct. One Trane Co. self-contained air conditioner, Model Number 202 SCW vertical unit is used for the reactor rooms.

Air leaves the unit via a $0.5 \times 1.1 \text{ m}$ duct which runs overhead through the supply damper and enters the reactor building above the air lock. Once inside the reactor building, the duct splits into two $0.5 \times 0.5 \text{ m}$ ducts running concentric with the reactor building. One branch runs clockwise and the other counter clockwise. Over 70% or $2.77 \text{ m}^3/\text{s}$ of conditioned air is recirculated through a $0.76 \times 1.2 \text{ m}$ return-air duct which returns to the plenum of the air-conditioning unit.

A fan in the stack cupola on the top of the dome, Sec. 11.3.2.2, exhausts the make up air through the stack damper and louver. Replacing the exhausted air with fresh outside air at the rate specified is sufficient to change the air at the breathing level twice every hour. This is desirable to minimize the build-up of airborne radioactivity, primarily ^{41}A , Sec. 11.3.

The extreme winter design condition was taken as -10°C (db) with 6.7 m/s average wind velocity. The reactor building is heated by hot water coils with a value of 41 MJ/m^3 .

Heating calculations are based upon approximately two air changes per hour within the breathing level. The height of the breathing level is assumed to be 4.6 m above the building floor.

Under winter conditions it takes 166 kW to maintain a temperature of $24 \pm 3^\circ\text{C}$ in the reactor building. The heating coils are Drayer-Hanson type CT, Series 96, 2 row bend types. All capacities are based on 14°C entering air and 82°C hot water in and 71°C hot water out.

Cooling of the reactor building is accomplished by the self-contained 20 ton Trane air-conditioning unit located in the Room 1320 of the laboratory building, which removes the additional heat produced by the lighting system and potential occupancy by 10 people. The total quantity of heat to be extracted by the air conditioning unit in order to maintain an ambient (room) temperature of $24 \pm 1 - 3^\circ\text{C}$, when the outside temperature is

40°C (db) is 57.6 kW. The unit, under these design conditions, is capable of extracting 67 kW.

A flat type cleanable filter section serves to keep large particles out of the reactor building air-conditioning unit. The unit's cooling coil is of the direct expansion type, having a face area of 1.24 m^3 and a final temperature of 24°C (db) and 16°C (wb) for 3 m/s air velocity through the coils. Two forwardly curved cleanable fans deliver $3.8 \text{ m}^3/\text{s}$ at 0.25 kPa external static pressure. The fans are mounted on a common shaft driven by a 3.7 kW motor operating on 440 V, 3-phase, 60 Hz current. A 15 kW, 170 rad/s, 400 V, 3-phase, 60 Hz motor operates the cooling unit compressor.

The air-conditioning supply and return ducts are equipped with 0.9 m diam Rockwell butterfly valves. These units are spring-to-close, air-to-open, fast-acting positive-sealing valves. Upon receipt of the closing signal to the solenoid-operated pneumatic valve, the butterfly valves close within 2 s.

These butterfly valves serve to isolate the air conditioner and prevent it from becoming contaminated in case of an incident involving airborne radioactive materials. They remain open except during a building seal.

The containment building is also insulated to reduce the building temperature variation. A 32-mm layer of polyurethane foam has been sprayed on the exterior surface of the containment building. This polyurethane is nonflammable and has a nominal density of $30 \text{ kg}/\text{m}^3$ and a compressive modulus of 5.5 MPa. There is also a maximum service temperature of 77°C. This foam is covered with a water-base exterior fibered paint. The inside of the containment building has a coat of semigloss water base white paint with a light reflectance of over 80.

6.2.3.2.2 Air Filter/Scrubber System – The dynamic containment system operates to prevent excessive pressure from building up inside the containment building. A pressure switch starts the scrubber system at 0.5 kPa above atmospheric pressure and keeps it going until the building pressure falls to 0.22 kPa. These moderate pressure gradients reduce unfiltered leakage from the building to acceptable levels. The air filter/scrubber system, shown in Fig. 6.2-6, consists of the following elements.

a) **Scrubber** – The ^{131}I scrubber consists of a stainless steel tank approximately 0.35 m diam and 1.2 m high. This tank is completely filled with chemical porcelain intalox saddles. The scrubber solution flows down through the tank to a rectangular sump tank located under the scrubber. A circulating pump returns the KOH solution to the top of the scrubber. The air flows into the bottom of the scrubber and out the top.

Fig. 6.2-6 Air filter/scrubber system schematic.

The chemical reactions between I_2 , KOH, and H_2O produce KI, KIO_3 , and HIO. These reactions essentially go to completion in the presence of a strong base such as KOH, thereby removing nearly all iodine or other halogens present in the air (except for certain organic iodines such as methyl iodide). Tests performed on identical units, already installed at LLL, indicate an iodine removal efficiency of 99.9%.

b) Dehumidifier Assembly – The dehumidifier, a mechanical system to trap KOH solution aerosols, sits directly on top of the scrubber. It consists of a series of baffle plates that entrap any moisture particles, arranged to let the fluid drip back into the solution stream.

c) Air Filter – A $0.3 \text{ m}^3/\text{s}$ absolute filter is installed to prevent carry-through of any particulate material. Its effectiveness for particles greater than $0.3 \mu\text{m}$ is 99.9%. This filter is purposely much larger than the rest of the system, to provide ample filtering capacity for extensive periods of time, without increasing the pressure drop across the filter.

d) Impregnated Charcoal Beds – Two charcoal beds (MSA No. 85851) are located just after the absolute filters. They have a rated capacity of $0.16 \text{ m}^3/\text{s}$ with

Fig. 6.2-7 Efficiency of an impregnated charcoal filter bed for methyl iodide removal vs relative humidity.

0.2 m/s bed velocity, and about 0.25 s dwell time. Figure 6.2-7 shows an expected efficiency rating vs relative humidity.

e) Blower — The exhaust pump is an ILG-10P centrifugal-type blower. This blower has ample capacity to exhaust the air from this system at the rated $14 \text{ dm}^3/\text{s}$.

f) Piping and Stack — The entire piping system is made up of 4 in. tubing connecting each of the elements. Upon leaving the blower the effluent air is carried up a vertical stack, approximately 12 m high, to be exhausted above the roof of the containment building.

g) Pressure Switch — The pressure switch is a standard air-conditioning type pneumatic switch, set to close contact at 500 Pa positive, and to open the contacts at 220 Pa positive. It is undesirable to attempt to operate such a switch to open at exactly zero differential. The manufacturer therefore recommends that there be a slight positive differential.

h) Vacuum Break Valve – The LPTR containment building is equipped with two vacuum break valves which open at -370 Pa. Each valve is designed to pass approximately $240 \text{ dm}^3/\text{s}$ at this pressure. These vacuum break valves are installed to prevent the building from ever attaining a negative pressure greater than -370 Pa. The building is designed to withstand approximately -750 Pa.

i) Emergency Power – The entire scrubber system is connected to the emergency power supply that automatically starts upon loss of the normal a-c power. This emergency power supply is a 50 kW gasoline-powered unit, located at the sub-station just south of the LPTR building.

Each of the above elements is duplicated so that there are actually two complete independent systems. This provides emergency scrubbing capacity even though one unit were to fail. Each system is rated at $14 \text{ dm}^3/\text{s}$ for a total capacity of $28 \text{ dm}^3/\text{s}$. Each separate system is controlled by its own pressure switch, sensing building pressure through its own pneumatic tube leading into the containment building.

To determine the pumping capacity required, a diurnal change of 21°C is assumed in the ambient temperature. This would correspond to a night-time temperature of 4.4°C and a peak midday temperature of 43°C . These temperatures are the maximum daily extremes expected in the Livermore Valley on the basis of 30 y of meteorological records. Tests conducted on the LPTR containment building indicate that the building temperature variation is always less than 75% of the ambient air temperature variation. Assuming a 29°C building air temperature variation, that occurs over a 12-h period, and a simultaneous drop in barometric pressure of 3.39 kPa , the total effective overpressure would be 13 kPa . This overpressure would require exhausting 800 m^3 of air during this 12-h period to maintain the building pressure at zero. This would, therefore, require a pumping rate of $19 \text{ dm}^3/\text{s}$.

6.2.3.2.3 Emergency Air Decontamination System – The reactor building is provided with an airborne particulate decontamination system for use in emergencies. It is capable of filtering the entire air content of the building whether or not the building is pressurized. The decontamination fan and filter box is situated outside the reactor building so as not to obstruct the valuable floor area of the reactor building and so that they may be serviced, if necessary, during a period when the building is contaminated with radioactive materials.

For reasons of economy the ducting of the air-conditioning system, Sec. 6.2.3.2.1, is used to channel the flow of air in the decontamination system. This flow of air is from the reactor building supply duct, through a bank of absolute filters, and then through the decontamination blower. The blower, including fan and motor, is located wholly within the ducting in order to prevent any leakage to the environment. It is capable of

delivering $2.6 \text{ m}^3/\text{s}$ against a static pressure head of 0.92 kPa. The fan is specially designed to perform efficiently in an ambient atmosphere of 14 kPa. Its rotor is of the adjustable pitch type, dynamically and statically balanced to minimize vibration. The 5.6 kW motor operates at 353 rad/s on 220/440 V, 3-phase, 60 Hz current. All electrical connections are made in an external conduit box.

After leaving the blower filtered air enters the reactor building through the decontamination damper, then the air-conditioning duct along the periphery of the south side of the building. This air is then discharged from the registers in the ducting, sweeps northward across the reactor room and into registers of the ducting on the north side of the building, which lead back to the filter boxes. Cleanup of the air is thus effected by continuously sweeping air through the building from south to north.

Four absolute filters rated at $0.52 \text{ m}^3/\text{s}$ at 0.25 kPa are used. Each filter consists of a plywood frame 0.76 m deep weighing 18 kg, holding a glass asbestos filtering medium with asbestos separators and a self-extinguishing rubber-base sealer. The filters are individually tested with $0.3 \text{ }\mu\text{m}$ dioctyl-phthalate smoke and certified to have an efficiency of not less than 99.97% and a static pressure drop not greater than 0.25 kPa when operated at rated capacity of air at 21°C .

The particle filters are housed in a specially designed filter box, tight to 14 kPa internal pressure, located outside the reactor building. It is approximately 2.1 m long by 0.86 m high by 1.7 m wide and is bolted to a concrete foundation. For protection from the weather it is made of $3/4$ in. fire-resistant exterior plywood. The cover of the box is removable for easy replacement of the dirty or contaminated filters. A draft gauge complete with three-way cocks and static pressure taps provides a check on the pressure drop across the filters.

Inside the box each bank of two filters has its own linkage system to compress the filter frame against a rubber seal in order to prevent bypass leakage. These links are capable of remote release, and they incorporate inherent adjustment to overcome any set in the rubber seals.

Several condensate drain couplings on the bottom of the filter box provide a way to drain it if it should ever be flooded, e.g., during decontamination.

The air decontamination system damper is located in the air conditioning duct inside the reactor building, just south of the southwest crane column. During normal operation it closes off the return duct of the air decontamination system, preventing air flow through the filters. At the same time it lets conditioned air flow unimpeded through the air conditioning duct, entering the building through nine supply registers. During a building seal or power failure it opens the decontamination system return duct and

reroutes the air in the air-conditioning duct to establish a south-to-north flow through the building for decontamination. It does this by blocking the air-conditioning duct, returning decontaminated air to the building through three supply registers on the south side and converting the six north-side registers into intakes for the decontamination system.

The decontamination damper need not be tight to pressure since it is entirely within the reactor building. A gasketed removable cover in the light gauge sheet metal duct makes it possible to clean and service this damper.

6.2.3.3 Design Evaluations -

6.2.3.3.1 Filter/Scrubber System - Using the design basis accident described in Sec. 15.4, we can arrive at some idea as to the amount of longer term methyl iodide which might be formed. The ^{131}I inventory is about 65 kCi. We assume that 25% of this is released or 2.54 Ci/m^3 in the containment building. Assuming that 2% of this is methyl iodide, that is 50 mCi/m^3 or about 325 Ci total of methyl iodide. The scrubber system presently removes 99% of this. Thus we would be exhausting air containing 0.5 mCi/m^3 of methyl iodide-131 to the atmosphere, or about one hundredth the emission rates shown in Sec. 15.4.1.3.

6.2.3.3.2 Emergency Air Decontamination System - The emergency filter system was checked using $0.3 \mu\text{m}$ dioctyl phthalate smoke and found to be almost 100% efficient for particles larger than $0.3 \mu\text{m}$.

6.2.3.4 Tests and Inspections -

6.2.3.4.1 Filter/Scrubber - The filter/scrubber system is checked in stages. The blowers and pumps are checked every week and the pressure switches are checked monthly. The caustic level is checked monthly and filters are tested annually.

6.2.3.4.2 Emergency Air Decontamination System - Each emergency air decontamination system filter is tested for efficiency and pressure drop at an AEC checking station before being put in service at LLL. In-place checks are made immediately after installation and periodically during its lifetime. Tests are conducted with appropriate instrumentation and a $0.3 \mu\text{m}$ dioctyl phthalate (DOP) smoke. AEC installations normally use a Q-127 Penetrator for testing; our in-place testing uses an Air Techniques Inc. TDA-2C Photometer. The recirculation damper is checked every week along with the blower system.

6.2.3.4 Instrumentation Application - A panalarm system in the control room is equipped with an annunciator that indicates the status of the air-conditioning system

valves and the recirculation damper. The recirculation blower has to be started by hand and can be checked by monitoring the streamers on the air-conditioning outlets either by T. V. cameras or visually observing in the reactor room.

The filter/scrubber system is hooked into the control room by a panel of lights. The lights indicate failure of blower or pump a-c power and the mode of operation of the pumps and blower, either automatic or hand operated.

The filter/scrubber building outside the reactor is equipped with manometers and pressure switches, and a control panel to operate the system either automatically or manually. Outside the north wall of the filter/scrubber building a manometer, readable from a distance, indicates the pressure in the reactor building.

6.2.4 Containment Isolation System

6.2.4.1 Design Bases — The LPTR was designed and erected prior to 1958. The philosophy with regard to piping penetrations at the time was that fluid-filled lines would not be equipped with isolation valves since the fluid lines were connected to closed, leak-tight equipment or systems on either or both ends. The only containment penetrations equipped with automatic isolation valves are those associated with the air-conditioning system. Closure of these valves takes place upon a building seal.

Radiation monitoring stations that may participate in initiating a building seal include:

- (a) Containment building exhaust stack.
- (b) Exhaust duct system.
- (c) Top of reactor pool.
- (d) Primary reactor coolant line.
- (e) Balcony level of reactor shield.
- (f) West thermal column.
- (g) Northwest side of containment building.
- (h) Northeast side of containment building.
- (i) South side of containment building.

Of these, high radiation readings from either (a) or (b) alone will initiate a building seal. Also, a building seal may be initiated by either (c) or (d) plus any other station from (c) to (i).

At the time the LPTR was built, AEC General Design Criteria 54-57 were not in force.

6.2.4.2 System Design — Figure 6.2-8 indicates the containment penetrations with the exception of the doors. Valves are indicated when they are installed within a reasonable distance from the containment boundary, usually 2.4 m. Table 6.2-2 is a

Fig. 6.2-8 Penetrations through the containment shell, schematic.

de penetrations

de penetrations

2

Table 6.2-2 Containment penetrations.

Item and number	Normal operating condition	Action on total loss of power	Action on total loss of instrument air	Parameter sensed to close automatically
Vacuum relief valves (2)	Closed	None	None	Normally closed
Air conditioning supply valve (1)	Open	Close	Close	High radiation level
Air conditioning return valve (1)	Open	Close	Close	High radiation level
Air conditioning stack exhaust valve (1)	Open	Close	Close	High radiation level
Air scrubber inlet lines (2)	Open	None	None	Manual valves designated remain open.
Truck door (1)	Closed and gasket inflated.	None	None - has independent air supply reservoir.	Door must be closed for tor to withdraw control r
Escape manddoors (2)	Closed and gasket inflated.	None	None - has independent air supply reservoir.	Door must be closed for tor to withdraw control r
Double interlocking airlock manddoors (2)	Both doors closed with gaskets deflated.	One gasket inflates to seal building.	None - has independent air supply reservoir.	High radiation level infla one door gasket.
Fast pneumatic rabbit system 1.25 in. id-nitrogen line (1) 1.25 in. id-rabbit line (1)	Nitrogen line has process valve closed inside and outside building. Rabbit line does not open to containment building and exhausts through HEPA filters outside building.	None	None	NA
12 in. secondary coolant lines (2)	Open	None	None	NA
12 in. RW 137 and 139		None	None	NA
Foam generator 4 in. water line (1) 0.5 in. foam solution (1)	Closed Open to foam concentrate tank.	None	None	NA
Slow pneumatic rabbit system lines 1.5 in. (2)	System has HEPA filters at both points that are open to containment building atmosphere.	None	None	NA
<u>Miscellaneous water lines</u>				
1 in. demineralized water (1)	Open	None	None	None
3 in. LCW-101 and V80 (1)	Open			
3 in. LCW-104 and V81 (1)	Open			
1 in. cold water (1)	Open			
3/4 in. RW-118 and V77 (1)	Open			
2 in. RW-158 and V79 (1)	Open			
2-1/2 in. fire hose (2)	Closed			
<u>Miscellaneous gas lines</u>				
3/8 in. instrument air (3)	Open	None	None	None
1 in. sample lines (3)	Closed			
2 in. LPG and V76 (1)	Closed			
1/2 in. instrument line (1)	Open			
1/4 in. instrument lines (6)	Open			
3/4 in. pumpout (2)	Closed			
4 in. spare (1)	Closed			
<u>Compressed air</u>				
1/4 in. tube (1)	Open	None	None	None
2 in. CA 105 and V78 (1)				
1/2 in. CA (1)				
1-1/2 in. CA air lock (5)				
1 in. CA (2)	Open			
<u>Tank and sump lines</u>				
2 in. RW 142 and V86 (1)	Closed	None	None	None
2 in. RW 143 and V82 (1)				
2 in. RW 144 and V83 (1)				
1-1/2 in. RW 141 and V84 (1)				
1-1/2 in. RW 130 and V42 (1)	Closed			

ntainment penetrations.

Action on total loss of instrument air	Parameter sensed to close close automatically	Possible action if item jeopardizes containment integrity
None	Normally closed	Can be manually sealed with blind flange.
lose	High radiation level	It is possible to physically close valve from outside containment building.
lose	High radiation level	It is possible to physically close valve from outside containment building.
lose	High radiation level	None
None	Manual valves designated to remain open.	Valves can be closed manually.
None - has independent air supply reservoir.	Door must be closed for reactor to withdraw control rods.	None
None - has independent air supply reservoir.	Door must be closed for reactor to withdraw control rods.	None
None - has independent air supply reservoir.	High radiation level inflates one door gasket.	Outer door gasket could be inflated by improvising an air supply.
None	NA	Process valves internal and external to containment can be closed remotely.
None	NA	Process valves internal and external to containment can be manually closed.
None	NA	Manual valve on foam solution line can be closed.
None	NA	Seal laboratory exit terminal.

summary of the containment penetrations and shows, among other things, the status and action on loss of air or power.

The normal source of air for the inflatable door seals is the building instrument air supply. If this supply is interrupted, there is a 0.34 m^3 accumulator charged to 690 kPa. This is in turn connected to a nitrogen cylinder containing 6.2 m^3 at STP, which is charged to 14 MPa.

6.2.4.2.1 Missile Protection — The stack exhaust isolation valve is protected against missile damage by virtue of its location inside the plenum on the containment roof. The air-conditioning supply and return isolation valves have metal boxes to protect their position-indicating limit switches and exposed cylinder rod. In addition the containment building to the east and the laboratory building to the west provide partial shielding. Only missiles that fall within 0.26 rad of a vertical north-south plane can strike the valves. Missiles outside this aperture would strike the containment building shell or the laboratory building wall.

The two 1 in. Sch. 40 above-grade compressed air lines penetrate the containment in the same area as the air conditioning isolation valves and are offered the same degree of protection described above. The air lock compressed air lines, the demineralized water line, the slow pneumatic rabbit lines, the 1/4 in. and 1/2 in. instrument lines all pass directly from the containment building into the laboratory building and are not exposed to missiles from an external source. The tank and sump lines are below-grade penetrations and terminate in an underground valve pit. Burial similarly protects the rest of the below-grade penetrations.

The remaining above-grade lines and valves shown on Fig. 6.2-8 are not protected against missiles.

6.2.4.3 Design Evaluation — The only containment penetrations that are normally open to the ambient atmosphere are provided with isolation valves. Specifically these are the stack exhaust, the air-conditioning supply, and the air-conditioning return. These valves close on loss of air or power. They are spring-loaded-closed and air-to-open valves. Routine testing of these valves on a weekly basis has been in effect for many years.

The LPTR has a relatively small inventory of core fission products and fuel meltdown does not occur in the case of a LOCA. Also, there is a small positive differential pressure across the containment under any condition. Upon review of these facts it appears that installation of isolation valves according to AEC General Design Criteria 54-57 would not significantly increase the safety of the present containment system.

6.2.4.4 Tests and Inspections – The in-service surveillance of the containment system is accomplished through the weekly checklist and periodic maintenance tests. The weekly checklist consists of operational tests of the containment building isolation valves, the escape manddoors, truck door, and personnel air lock.

The isolation valve operation is monitored by initiating a building seal and observing the valve position indicators. The building seal is initiated by applying an external radiation source to one of the monitors or by manual operation from the control room console. The doors are checked by physically opening and closing the doors, which requires gasket inflation and deflation. The filter/scrubber system and emergency air decontamination system are tested per Sec. 6.2.3.4.

6.3 Emergency Core Cooling System

6.3.1 Design Base

Fission product decay heat is normally removed from the core by forced or natural convection of the pool water. The emergency core cooling system is designed to function in the event of a LOCA. The system is separate from the primary coolant system.

The analysis presented in Sec. 15.2.3.1 indicates without an operable ECCS the maximum fuel temperature following a LOCA is 504°C. This is 138°C below the melting point of aluminum. The LPTR emergency core cooling system provides an additional measure of safety.

6.3.1.1 Range of Coolant Ruptures and Leaks – The ECCS is designed to supply coolant in the event that a leak in the primary system lowered the reactor pool water level enough to expose the core. The most significant breaks are double-ended rupture of the largest pipe in the primary coolant system and complete shear of a beam port pipe. The double-ended pipe rupture results in the shortest drainage time while shear of a beam port pipe results in partial uncovering of the core. The LOCAs analysed in Sec. 15.2.1 indicate that not even instantaneous drainage would damage the core.

The severing of a beam port pipe is a special case. Although the drainage is slower than the double-ended pipe rupture, higher fuel plate temperatures could result because the level of the water left in the pool hinders natural convection air cooling of the fuel elements. However, the fuel plate temperatures still remain below the melting point.

Also this type of accident is not considered credible with the reactor at power. The reasoning behind this statement is that the beam port pipes are surrounded by the

reactor concrete shield for all but their in-pool portion. The in-pool portion is exposed to very minor temperatures and pressures. The only conceivable way that the pipes could be severed is by dropping a heavy load on them, i.e., inside the pool. However, the pool top is usually covered with the deck plates and the crane is restricted from handling loads over the pool when the reactor is at power. The conclusion is that the accident will not occur even though for completeness it is analyzed in Sec. 15.2.3.2.

6.3.1.2 Time to Drain Pool - The LOCA is assumed to take place following the double-ended failure of the 8 in. Sch. 40 primary coolant pipe where it enters the equipment pit. This is the largest primary coolant pipe and constitutes the worst case.

Examination of the pump suction piping between the reactor pool and the equipment pit indicates the following:

Length of 8 in. Sch. 40 pipe	14.0 m
Equivalent length of fittings	21.3 m
Entrance and exit losses	$3\left(\frac{V^2}{2g}\right)$
Initial static head	$8.8 \text{ m} - H_2O = 86 \text{ kPa}$

By equating the static head to the flow losses the maximum flow velocity can be obtained. Assuming the initial velocity is 7.0 m/s, the entrance and exit losses are

$$3\left(\frac{V^2}{2g}\right) = 7.5 \text{ m (73 kPa)}.$$

Total equivalent length of 8 in. Sch. 40 pipe is then

$$L = 14.0 + 21.3 + 7.5 = 42.8 \text{ m (418 kPa)}.$$

The head loss per meter is

$$\frac{H}{L} = \frac{8.8}{42.8} = 0.206 \text{ m} - H_2O/\text{m} - \text{of pipe}$$

or

$$\Delta P = 2.0 \text{ kPa/m}.$$

This value can be matched against a chart of friction loss versus velocity of water for an 8 in. Sch. 40 pipe⁵ to prove the validity of the initial assumption of the velocity.

Knowing the initial velocity, the flow velocity as a function of static head is obtained from the above data in the following manner.

The flow equation is

$$H = K \frac{V^2}{2g},$$

where K is the resistance coefficient

V = flow velocity, m/s

g = acceleration of gravity

$$= 9.8 \text{ m/s}^2$$

H = static head, metre of water.

Using the initial set of conditions

$$V = 7.0 \text{ m/s}$$

$$H = 8.8 \text{ m}$$

the values of V and K are found to be

$$K = \frac{2gH}{V^2} = 3.53$$

and

$$V = 2.36 \sqrt{H}.$$

The above expression can be changed to yield the flow rate in m^3/s by multiplying by the flow area of the 8 in. Sch. 40 pipe.

$$Q = VA$$

where

$$A = 0.032 \text{ m}^2$$

$$Q = 0.076 \sqrt{H} \text{ m}^3/\text{s}.$$

From the Darcy equation

$$H = f \left(\frac{L}{D} \right) \left(\frac{V^2}{2g} \right)$$

it is seen that

$$K = f \frac{L}{D}$$

where

f = friction factor

L = pipe length, m

D = pipe diameter, m.

Since the friction factor is a function of velocity it follows that so also is the resistance coefficient. However, for the case under consideration it can be shown that friction factor increases by less than 5% as the depth of the water decreases from 8.8 m to 3.4 m, the elevation at the bottom of the reactor active core. Therefore, for this analysis, K will be assumed constant.

The remaining primary piping can be examined in a similar manner to determine the flow velocity from the second end of the double ended pipe break. The flow restrictions in this section of piping are much larger due to the heat exchanger, orifice plate and numerous fittings. The friction head loss is found from Sec. 5.3.1 to be 153 kPa (15.6 metre of water) at $67 \text{ dm}^3/\text{s}$. The resistance coefficient for this piping section can be expressed as

$$K = \frac{H}{Q^2},$$

where

Q = flow rate, m^3/s

$$K = \frac{15.6}{.067^2}$$

$$= 3475$$

$$Q = 0.017 \sqrt{H}.$$

The total flow rate as a function of static head is the total of the two legs.

$$Q = 0.093 \sqrt{H}.$$

The flow rate as a function of time is also defined as

$$Q = A \left(\frac{dH}{dt} \right)$$

where

dH = decrease in pool water level, m

dt = time interval, s

A = flow area of reactor pool tank, m^2 .

The flow area is equal to the cross sectional area of the 2.00-m i.d. pool tank from the reactor top down to the thermal columns at 4.5 m elevation. From that point on the flow area is approximately 1.67 m^2 .

Therefore, the elapsed time for the water level to drop from the 8.8 m elevation to the bottom of the core at the 3.4 m elevation is:

$$\int dt = \int \frac{A_1}{Q} (-dH) + \int \frac{A_2}{Q} (-dH)$$

$$\int_0^t dt = \frac{3.16}{0.093} \int_{8.8}^{4.5} \frac{-dH}{\sqrt{H}} + \frac{1.67}{0.093} \int_{4.5}^{3.4} \frac{-dH}{\sqrt{H}}$$

$$t = 68 (\sqrt{8.8} - \sqrt{4.5}) + 36 (\sqrt{4.5} - \sqrt{3.4})$$

$$= 68 \text{ s}$$

The reactor automatically scrams when the pool surface drops to 8.8 m above the ruptured line. Therefore, this is time zero and the first 68 s of fission product decay heat would be transferred to the pool water. After that it would be desirable to supply water from the ECCS to hold the fuel plate temperature below the 504°C maximum.

Calculation of the time to drain through a severed beam port pipe is handled in a similar manner. The difference in elevation between the B-1 beam port centerline (1.07 m) and the reactor pool water level (6.40 m) is 5.33 m. The cross sectional area of the pool changes at an elevation of 2.07 m.

The flow through the severed beam port pipe is assumed frictionless and the discharge coefficient for the severed end of the pipe is assumed to be unity. These are conservative assumptions yielding the smallest drain time.

The flow rate equation is:

$$Q = VA$$

where

$$Q = \text{flow rate} \quad \text{m}^3/\text{s}$$

$$V = \text{velocity} \quad \text{m/s.}$$

But

$$V = \sqrt{2gH}$$

$$= 4.43 \sqrt{H}$$

and

$$A = \frac{\pi}{4} d^2 = 0.785 (0.146)^2 \times 0.0167$$

$$= 0.0167 \text{ m}^2.$$

Therefore

$$Q = 4.43 \times 0.0167 \sqrt{H}$$

$$= 0.074 \sqrt{H}.$$

The time to drain is:

$$\int_0^t dt = \frac{3.16}{0.074} \int_{5.33}^{1.00} \frac{-dH}{\sqrt{H}} + \frac{1.67}{0.074} \int_{1.00}^0 \frac{-dH}{\sqrt{H}}$$

$$= 157 \text{ s.}$$

6.3.1.3 Fission Product Decay Heat – The energy release to the containment system during the LOCA is due solely to the radioactive decay of the core fission products. The decay energy is determined from the proposed ANS Std. 5.1, Decay Energy Release Rates Following Shutdown of Uranium Fueled Thermal Reactors.² The fission product decay power for infinite reactor operating time at 3 MW was shown in Fig. 6.2-1.

The heat removal capacity of the water spray system assuming an initial temperature of 21°C and complete vaporization is

$$P = MC(\Delta h_f + \Delta h_{fg}).$$

Here

M = mass flow rate, kg/s

P = power, kW

Δh_f = enthalpy change of liquid from 21° to 100°C = 330.3 $\frac{\text{kJ}}{\text{kg}}$

Δh_{fg} = enthalpy change for evaporation
= 2256.9 kJ/kg

C = conversion factor

or

$$P = 2587 \text{ M.}$$

The volume flow rate for the present core spray system is $1.0 \text{ dm}^3/\text{s}$ which results in a mass flow rate M of 1.0 kg/s . Of this amount approximately 8% falls on the fueled positions of the core. The heat removal capacity is, therefore, about

$$P = 2587 \times 1 \times 0.08$$

$$P = 200 \text{ kW.}$$

The heat removal capacity ranges from 2 times the decay power at 68 s to 7 times at the end of 1 d.

6.3.1.4 Reactivity Required for Cold Shutdown – Not applicable.

6.3.1.5 Capability to Meet Functional Requirements – The ECCS utilizes city water supplied to the containment building via valve V79 and 2 in. line RW-158. The line forms a manifold after entering the building and encircles the containment building. It has eight branch connections, only three of which are used. One is connected to an emergency shower and eyewash, the second to a decontamination sink and the third to the ECCS. The city water system suffers about one unscheduled outage every 4 y.

Based on past performance the probability of a water outage at the time of a LOCA is slight.

An alternate source of water is the deionizer water system. Its routing is similar to the city water system. It has eight branch connections, only two of which are used. They are the decontamination sink and the ECCS. This source may experience an outage on the average of once every 2 y.

6.3.2 System Design

6.3.2.1 Schematic Piping and Instrumentation Diagram – Figure 6.1-1 indicates the system piping and instrumentation.

6.3.2.2 Equipment and Component Descriptions – The components comprising the ECCS system are described in Table 6.3-1.

6.3.2.3 Applicable Codes and Classifications – The design and installation is generally to the USA Standard Code for Pressure Piping, USAS B 31.1.0 – 1967. The hose and couplings are those typically used in low pressure installations and meet no national code.

Table 6.3-1 ECCS components

Component	Description ^a	Rating MPa	Material
Copper tubing	1.125 o.d. X .050 wall	3.1	Copper, Type L
	0.500 o.d. X .035 wall	4.8	
	2.125 o.d. X .070 wall	2.4	
Stainless steel tubing	1.000 o.d. X .049 wall	6.2	ASTM A269, Type 304
Plastic pipe	3/4 Sch. 80	1.7	ASTM D-1785 Type I, Grade 1, PVC
	1/2 Sch. 80	2.1	
Stainless steel pipe	1/2 Sch. 40	17	ASTM A312, Type 304
	1 Sch. 40	13	
Rubber hose	1.0 i.d.	1.4 working 5.5 burst at 93°C	Rubber, utility grade, 2 ply, 2 braid rayon, Goodyear type ARTAC
Strainer	1 IPS, 16 mesh stain- less steel screen	0.86	Type 316 SS
Spray nozzle	1/2 NPT, 7/32 orifice, 15° injector type	0.86	Type 303 SS
Valves:			
Motor operated	1-1/2 IPS ball valve 115 V, 60 Hz	1.0	PVC, normal impact
Manual bypass	1 IPS globe valve	2.8	Bronze
Isolation	1/2 IPS ball valve	1.0	PVC, normal impact type
Check	1/2 IPS swing check	2.1	Bronze
	1 IPS swing check	1.4	Bronze
Pressure regulating	2 IPS diaphragm type with strainer, 50 to 100 psi outlet pressure range.	1.7	Bronze

^aAll measurements in inches. One in. = 25.4 mm.

6.3.2.4 Material Specifications and Compatibility – The materials used in the ECCS are listed in Table 6.3-1. There are no material compatibility problems involved in the use of city water/deionized water and the materials listed.

6.3.2.5 Design Pressure and Temperatures – The design pressure and temperature for the city water system and the deionized water system are 1034 kPa and 24°C. The actual maximum pressure the city water system is subjected to is 827 kPa and the deionized water system's maximum is 483 kPa. The deionized water system operating pressure is typically 414 kPa. The city water system pressure is reduced to 552 kPa at the pressure regulating valve.

The pressure ratings for the ECCS components are equal to or greater than the design pressure rating (1034 kPa) of the water systems they are supplied from with two exceptions. The strainer and spray nozzles are rated for 862 kPa. However, these components are acceptable because:

1. They are downstream of the pressure regulator and hence actually experience less than 552 kPa.
2. They are on the open end of the ECCS and cannot be exposed to the full system pressure.
3. There are flow friction and static head losses that cause the pressure to be below 862 kPa at the components locations even if the pressure regulator failed.

6.3.2.6 Coolant Quantity – There is no meaningful limit to the quantity of coolant available from the city or deionized water systems.

6.3.2.7 Pump Characteristics – Not applicable.

6.3.2.8 Heat Exchanger Characteristics – Not applicable.

6.3.2.9 ECCS Flow Diagrams – Not applicable.

6.3.2.10 Relief Valves and Vents – Not applicable.

6.3.2.11 System Reliability – The reliability of the system depends primarily on operation of the motor-operated ball valve, assuming that the city water supply system is intact. The power to operate the valve is drawn from the normal building power circuit with backup provided by the 50 kW emergency power system, see Sec. 7.0. There is a degree of redundancy incorporated in the water supply system in that the deionized water system can be used if the city water system is unavailable. This switch to an alternate water supply must be performed manually. Bypass of the motor operated ball valve can also be done manually if the valve malfunctions.

6.3.2.12 Protection Provisions – The water supply line for the ECCS is a buried pipe that enters the containment building below ground level. This routing provides missile protection for the line outside the containment. After passing through the containment floor slab the line is attached to the steel shell with pipe clamps. Downstream of the valving, the copper tubing is replaced by a rubber hose. The hose runs through an underground conduit from the containment periphery to the reactor concrete shield which houses the pool. The rubber hose connects to stainless steel tubing attached to the reactor shield as it is routed to the top of the shield. The terminal end of the system is stainless steel pipe which is attached to the control rod actuator support structure.

Sections 3.7 and 3.8 indicate the containment building shell and foundation, the reactor shield, and the actuator support structure will survive a DBE without damage. Furthermore, they will move essentially as a unit without significant relative displacement between the structures. Since the ECCS lines are firmly attached to those structures, the lines should be adequately protected against seismic forces.

6.3.2.13 Provisions for Performance Testing – The provisions for performance testing are incorporated in the Tests and Inspections, Sec. 6.3.4.

6.3.2.14 Net Positive Suction Head – Not applicable.

6.3.2.15 Control for Motor-Operated Isolation Valves (PWR) – Not applicable.

6.3.2.16 Motor-Operated Valves and Controls (General) – Not applicable.

6.3.2.17 Manual Actions – The ECCS requires manual actuation to start the coolant flow. The motor operated ball valve is the only active component in the system. The valve is opened by use of a key-lock switch in the control room or containment building. A bypass valve, accessible during a LOCA (Sec. 15.2.4), can be opened manually if the motor-operated valve fails to operate.

Verification of operation is determined by observing the spray on the reactor pool tank television monitor. The operation can be terminated by the close switch on either of the two control panels. The need for ECCS actuation is determined by the rate of coolant loss. After the reactor is scrammed because of low pool water level, there is 4.8 m of water remaining above the core. The decrease in pool water level can be observed from the pool manometer in the control room and the rate of loss estimated. The operator can then actuate the ECCS if the rate is high or proceed with an investigation to locate the cause of water loss.

The failure requiring the fastest operator response is a double-ended rupture of the largest main coolant pipe. The drain time under that condition is 68 s. If the motor operated valve fails to open it requires 60 s for the operator to leave the control room, enter the containment building, and manually open the bypass valve. If the containment is sealed it requires an additional 60 s to pass through the air lock. To determine the consequence of a delay in initiating the ECCS refer to Fig. 15.2-2, the maximum fuel temperature as a function of time after shutdown.

From the figure it can be seen that failure of the system to operate at all results in a fuel temperature rise to 504°C. This is still 138°C below the melting point of aluminum and is not viewed as a serious problem. However, the sooner the system is placed in

operation the lower the maximum temperature. This is true up to approximately 1200 s at which time the peak temperature is reached.

6.3.2.18 Process Instrumentation – The control room instrumentation available to assist the operator in assessing post-LOCA conditions is:

- a) pool water level manometer
- b) audio transmitter at reactor pool top
- c) pool television which monitors top of core area
- d) radiation monitors in containment building.

The pool water level manometer is of value if the pool is not completely drained. The audio and visual instrumentation allow the operator to see and hear the conditions in the pool space. The radiation monitors aid the operator in determining containment building conditions.

6.3.2.19 Materials – The only materials in the ECCS that could suffer radiolytic or pyrolytic decomposition are the neoprene rubber hose and the polyvinyl chloride used for some of the pipe and valves. These components are so located that there is no danger at all of radiolytic decomposition. In the event of a fire, pyrolytic decomposition of these components is to be expected. This would render the system inoperable.

6.3.3 Performance Evaluation

The ECCS is installed to provide additional safety margin in the event of a LOCA. As demonstrated in Sec. 15.2.3.1, fuel melt will not occur even if the system fails to function. The system will function even if it loses its primary electrical power supply, emergency power supply (use manual bypass), and primary water supply (use deionized water).

The ECCS is independent of all reactor systems. The most probable failure that would prevent its operation is loss of water supply. The likelihood of this occurring is remote, as pointed out in Sec. 6.3.1.4.

6.3.4 Tests and Inspections

The ECCS will be tested on a semiannual basis. The test will check the operation of the system from initiation of the start signal to actual spray into the reactor pool tank. This will provide assurance that all electrical circuits are functional, the ball valve motor operates, and the spray nozzles are free from debris.

The test will include use of both sources of water, city and deionized. Also the control room and containment building actuation panels will be used to verify their functioning.

The spray pattern on the top of reactor pool water surface will be visually checked to establish that the nozzles are not plugged. The flow rate will be determined from a calibration curve for the system. The curve is a plot of water pressure versus flow. The pressure plotted is that immediately after the strainer. Also, the pressure drop across the strainer will be checked for possible debris accumulation.

Inspection of the ECCS will be performed concurrently with the functional test. It will include visual checks of the accessible lines and fittings for damage and leaks.

6.3.5 Instrumentation Requirements

The ECCS is a manually activated system and its instrumentation requirements are covered in Sec. 6.3.2.17, manual actions.

6.4 Habitability Systems

The reactor control room is subject under almost all conditions to background environmental radiation only. Figure 15.2-4 displays radiation values for a loss of coolant accident, which are anticipated to be below 5 R/h for the duration of the accident. Section 15.4.1.6 shows that the radiation associated with a design basis accident (DBA) would undoubtedly exceed the 5 R/h level. The DBA is the only situation where radiation in the control room would not meet Criterion 19.

REFERENCES

1. Livermore Pool Type Reactor Safeguard Report, Addendum IV, Foster Wheeler Corp., Rept. FW-56-006 (1960).
2. Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors, American Nuclear Society, Hinsdale, Ill., Proposed Standard ANS-5.1 (1971).
3. W. B. Cotrell and A. W. Savolaine, U.S. Reactor Containment Technology, VI, Oak Ridge National Laboratory, Oak Ridge, Tenn., Rept. ORNL-NSIC-5 (1965) pp. 10.71-76.
4. J. P. Holman, Heat Transfer, 2nd ed. (McGraw-Hill Book Co., New York, 1968) p. 199.
5. H. F. Rase, Piping Design for Process Plants (Wiley and Sons, New York, 1963) p. 132, Fig. 4.6.

CHAPTER 7.0 INSTRUMENTATION AND CONTROLS

7.1 Introduction

7.1.1 Identification of Safety Related Systems

7.1.1.1 Nuclear Instrumentation — These systems respond directly to the neutron flux and initiate protective actions based on that response. Table 7.1-1 and Fig. 7.1-1 show the subsystems that provide scrams or shutdowns. Reactor status readouts are also indicated. Figure 7.1-2 shows the range and overlap of nuclear instruments for the standard core.

7.1.1.2 Non-Nuclear Instrumentation — These systems supply information related to control of the reactor, auxiliary systems, and engineered safety features. Figure 7.1-3 shows magnet current and condition monitoring, rod position monitoring for regulating and shim rods, and coolant parameter monitoring.

7.1.1.2.1 Control Rod Monitoring — Rod position limit and range switches are provided and are interlocked with their respective control systems. There are no scrams or shutdowns initiated by the position detection circuits.

Each of the four shim rods and the regulating rod have a continuously working position indicator so that the operator will know at all times the positions of the rods. In addition, there is a single precision digital position indicator which can, by selection, be connected to any of the five rods for more exact position indication.

Each of the shim rod magnet circuits contains two current-indicating meters, which are checked daily to determine that the magnet currents are in balance and within the normal operating range.

Each circuit is equipped with a "first to drop" indicator to aid in trouble-shooting and repair when a rod drop occurs.

7.1.1.2.2 Coolant Monitoring — The reactor primary and secondary coolant temperatures and flow rates are recorded. A primary coolant temperature above 52°C (~125°F) will cause a slow scram. The pool water level is monitored, and if the level drops to 6.6 m (21.5 ft), a float switch trips a slow scram and turns off the primary coolant pump. The primary coolant inlet-outlet differential temperature (ΔT meter), in conjunction with the fixed flow rate, is used to determine reactor power at equilibrium. The other instrumentation is calibrated from this data.

Table 7.1-1. Nuclear instrumentation.

Subsystem	Sensor	Readout		Scram trip levels		Shutdown	Annunciator
		Meter	Recorder	Fast	Slow		
Log count rate (1) ^a	Fission chamber (10)	X	17	--	--	--	--
Count rate period (1)	Fission chamber (10)	X	18	--	X	X	X
Log N power (2)	Ion chamber -- compensated (11)	X	19	--	X	X	X
Log N period (2)	Ion chamber -- compensated (11)	X	18	--	X	X	X
Linear level (3)	Ion chamber -- compensated (12)	X	19	--	X	X	X
Level safety-A (4)	Ion chamber (13)	X	20	X	X	--	X
Level safety-B (5)	Ion chamber (14)	X	--	X	X	--	X
Level safety-C (6)	Ion chamber (15)	X	--	X	X	--	X
Period level safety-D (7)	Log N system (2)	X	--	X	X	--	X
Regulating rod servo system (PAT 60) (8)	Linear level recorder (19) and power set control (21)		--	--	X	X	X
Signal conditioner (modifies linear level signal to PAT 60) (9)	¹⁶ N gamma detector (NaI) (16)	Digital readout	--	--	--	--	--

^aThese numbers refer to Fig. 7.1-1.

Fig. 7.1-1 - Reactor nuclear instrumentation.

Fig. 7.1-2 — Typical instrumentation overlap.

Fig. 7.1-3 — Reactor nonnuclear instrumentation.

7.1.1.3 Containment Building Seal System – This system is part of the containment system described in Sec. 6.2. Building seal initiation is manual or by radiation detectors located in various parts of the reactor structure. Section 7.3.1 discusses the associated circuitry, interlocks, and displays.

7.1.1.4 Safety and Interlock Chains – Reactor safety is maintained by the various safety and interlock chains that assure proper configurations of control and instrumentation for start-up and operation. Opening of the chains either inhibits certain operations or initiates a rundown or scram. The chains are described in detail in following sections.

7.1.1.5 Alarms and Annunciators – All of the reactor systems are adequately equipped with visible and audible annunciators which inform the operators of the occurrence of events of importance. Some of these result in operator action, some are coincident with the initiation of scrams or rundowns, while the balance are of peripheral interest and require only operator acknowledgment. These systems are described in more detail in Sec. 7.5.

7.1.1.6 Control Systems – These include: the regulating rod (nuclear control system), Sec. 7.7.1.3; shim rods, Sec. 7.7.1.1; scram trips, Sec. 7.2; the ventilation system, Sec. 12.2; the scrubber/filter system, Sec. 6.2.3; the water cooling system, Ch. 5; and other systems including doors, airlocks, etc., Sec. 6.2.

7.1.1.7 Shutdown Systems – The reactor controls are designed to be fail-safe. Loss of control power results in a slow scram by dropping the shim rods. There are no power requirements for the safe shutdown of the reactor.

7.1.2 Identification of Safety Criteria

7.1.2.1 Design Bases – The safety related systems have been evaluated with regard to single component failure. No single component failure can cause a system to fail and leave the reactor without a scram capability within safe limits. The following documents were reviewed and their contents compared to the design of the LPTR systems: Institute of Electrical and Electronic Engineers (IEEE) Standard: Criteria for Protection Systems for Nuclear Power Generating Stations, IEEE, New York, Rept. 279 (1971); and Design Principles of Reactor Protection Instrument Systems, Oak Ridge National Laboratory, Tenn., Rept. NSIC-51 (1968).

7.1.2.2 LPTR Design Bases – The LPTR safety plan requires the following controls. Start-up and operation shall be accomplished with safety to the reactor,

operators, laboratory and off-site personnel, and surroundings. Instrumentation and control shall be provided which will ensure adequate indications of reactor system conditions and prevent undesired transients. There shall be more than one capability for initiating a rundown or scram in the event that abnormal conditions should develop. Scram trips shall occur at such nuclear levels and speeds that no damage to the reactor or hazard to the environment occurs. In the event of accidental release of radioactive material within the reactor building, the event shall be monitored, so that the spread of contamination may be controlled within safe limits.

7.2 Reactor Scram System:

7.2.1 Description

The reactor scram system, shown in Fig. 7.1-1, can initiate a reactor scram in three general ways:

- (1) Fast scram – Electronic interruption of the magnet current to the four shim rods, Sec. 7.2.1.1.
- (2) Slow scram – Relay interruption of the a-c power to the magnet power supplies. A delay of up to 100 ms longer than the fast scram is introduced by this method, Sec. 7.2.1.2.
- (3) Slow scram safety chain – Any of the following will open the slow scram safety chain and cause a scram trip: the manual scram pushbutton switch; the coolant low level, high temperature, and low flow rate switches; the electronic system equipment trouble monitor; and the remote scrams, e.g., the manually operated pushbutton switch on the reactor top and the automatic experiment-related switches.

7.2.1.1 Fast Scram Detection Sensors

7.2.1.1.1 Uncompensated ion chambers (UCIC) – Current from each of three UCICs is amplified by its corresponding composite safety amplifier to drive a common trip bus (sigma bus). One of the amplifiers is monitored by means of the megawatt-day recorder.

7.2.1.1.2 Compensated ion chamber (CIC) – Current from this chamber is amplified in the log N amplifier to a power signal. The change in this power signal is converted to a level that is proportional to the reactor period in a composite safety amplifier driving the sigma bus. The composite safety amplifiers are described in 7.2.1.3.

7.2.1.2 Slow Scram Sensors – The four sensors described in Sec. 7.2.1.1 above also generate a slow scram by means of a relay in each of the composite safety amplifiers. The relay is set to trip at a slightly lower level than the fast scram.

The compensated ion chamber in Sec. 7.2.1.1,2 is the source of signals for the log N power and period recorders which have level and period trips in accordance with technical specifications (Sec. 16.3).

A compensated ion chamber is the signal source for the linear level micro-micro ammeter recorder which has a switch trip setting in accordance with technical specifications (Sec. 16.3).

A fission chamber is the signal source for the log count rate period recorder during the start-up operation. The recorder has a switch trip setting in accordance with technical specifications (Sec. 16.3).

Relays from experiments whose safety is related to the neutron flux of the reactor can cause a slow scram via the experiment scram circuit. Typical inputs are from the fast neutron irradiation facility (fission plate temperature), through tube targets, and from the pool water level.

7.2.1.3 Fast Scram Initiation By Means of the Composite Safety Amplifiers

7.2.1.3.1 Amplifier Description – Figure 7.2-1 is a simplified block diagram of the amplifier. Each of the four amplifier chassis contains the following:

- An ion chamber preamplifier that converts the input to a voltage level.
- A period amplifier that differentiates a log N power input to a period voltage level.
- A sigma amplifier (No. 1) that drives sigma bus (a wire common to all four safety amplifier chassis) from the voltage level of the selected input (ion chamber or period).
- A magnet amplifier that lowers the magnet holding current to a drop level when the voltage on the sigma bus exceeds the trip settings or when the bus voltage drops to an abnormally low level (as in the case of a short on the bus).

7.2.1.3.2 Composite Safety Amplifier Inputs – Three amplifiers have uncompensated ion chamber inputs and are labeled A, B, and C. These provide the power level trips. One amplifier, D, has a log N power input and provides the period trip.

Fig. 7.2-1 - Composite safety amplifier.

Each of these amplifiers is tied to the sigma bus by means of isolation diodes that allow its sigma amplifier No. 1 to drive the bus up (towards trip) but prevent it from pulling the bus down.

7.2.1.4 Slow Scram Initiation By Means of the Composite Safety Amplifiers - A level scram is initiated by sigma amplifier No. 2 (Fig. 7.2-1). The level signal from the ion chamber preamplifier or the period amplifier is passed through the sigma amplifier No. 2 to the scram level detector which has been set to actuate relay No. 5 at a preassigned power level (Sec. 16.2.2). Relay No. 5 initiates a slow scram via the trouble monitor chassis (Fig. 7.1-1).

7.2.1.5 Slow Scram Initiation Due to Amplifier Trouble - The amplifier trouble monitor (a group of relays) is actuated by any of the composite safety amplifiers when a component fails or a wire is disconnected initiating a slow scram. The conditions that will initiate this action are described below. See Fig. 7.2-1.

A chamber cable disconnect or failure of the 385-V supply will de-energize relay No. 1.

Failure of the sigma amplifier tube or the 210-V supply will de-energize relay No. 2.

Failure of the sigma preamplifier will de-energize relay No. 4.

If any of the relays 1, 2, or 4 above should de-energize or if the 300-V supply should fail, relay No. 3 will de-energize.

If relay No. 3 should de-energize or if the sigma bus cable should disconnect, the amplifier trouble chain will be opened. This disables the magnet power supply. Trouble indicators are illuminated on the panelarm and on the particular chassis in trouble.

7.2.1.6 Power Simulator Test Chassis - A power simulator chassis can be switched into each of the level safety amplifier inputs, and its output to the sigma bus compared to a trip voltage reference. The simulator provides a test to insure that the safety amplifiers will apply the proper trip voltage to the sigma bus. The simulator can only be used with one safety amplifier at a time, and its use initiates a rundown.

7.2.1.7 Bypass, Fast Scram System - It is possible to bypass the period fast scram input to the D safety amplifier, Fig. 7.1-1. This bypass does not affect the integrity of the other three fast scram level safety inputs.

7.2.1.8 Bypass, Slow Scram System - It is possible under administrative control to bypass the low coolant flow switch and the period trip switch in the period recorder.

7.2.1.9 Interlock, Scram Chain - The normally closed contacts that make up the magnet chain are shown in Fig. 7.2-2. The nuclear levels are described in Sec. 7.2.1.2. Amplifier trouble inputs are described in Sec. 7.2.1.5. Other interlocks are listed in Sec. 7.2.1.

When turned on, the rod drop tester inserts a relay contact into the magnet chain. This relay contact is used to interrupt the magnet chain when rod drop time is being measured.

The amplifier trouble relay chain is shown in Fig. 7.2-3. In addition to the inputs from the safety amplifiers described in Sec. 7.2.1.5, the following must be in the chain:

- The power simulator must be connected (Sec. 7.2.1.6).
- The log N chassis and its period output to the D safety amplifier must be connected and its internal supply must be at voltage to make up this link.
- The linear level and the log N chamber power supplies must be connected and be at voltage to make up their links.

Fig. 7.2-2 — Scram chain.

Fig. 7.2-3 - Amplifier trouble chain.

7.2.1.10 Rundowns - A rundown causes the four shim safety rods to drive in for as long as the initiating signal exists. A rundown can be caused by those nuclear instrument recorders shown in Fig. 7.1-1 and by building seal violations (emergency doors) as shown in Fig. 7.3-1.

7.2.1.11 Failure to Scram - A detailed fault tree analysis revealed that no single component failure will prevent at least three of the shim safety rods from falling into the core.

7.3 Engineered Safety Feature Systems

7.3.1 Description - The reactor building containment system is designed to minimize the release of radioactive materials. When a building seal occurs, large valves close on the inlet air (from the air conditioner) and on the stack effluent system exhaust at the top of the reactor containment structure. The doors are sealed by means of an inflatable gasket around the edge of each door. Doors other than the airlock are

monitored by the panalarm and, if opened, cause a rundown mode of operation until they are closed.

7.3.1.1 System Description — Radiation detectors in seven locations in the reactor building initiate automatic seal of the containment structure whenever the radiation level of the primary coolant or the top of the reactor and any one of the other radiation detectors reach their set points. Figure 7.3-1 shows the control block logic for

Fig. 7.3-1 — Automatic building seal system.

initiating a building seal. Section 12.1.4 describes the monitors, typical set points, and typical actions. A multipoint recorder provides a record of the radiation levels and serves as the scanner for the relay logic that initiates the building seal for the above seven detectors.

The reactor operator can initiate a building seal by means of a console pushbutton.

Either the stack or duct detectors described in Sec. 12.1.4 seal the building. They are recorded on a chart recorder.

A power failure will cause the building to seal.

An evacuation horn sounds 30 s after the building seal is initiated, unless the operator determines it is not necessary and disables the circuit.

7.3.1.2 Filter/Scrubber System – The filter/scrubber system, Sec. 6.2.3, is designed to relieve the sealed containment structure of any overpressure. The system is made up of two separate filter units. The redundant system control is shown in Fig. 7.3-2. When the building is sealed, the pumps circulate fluid to wet the filter beds of the scrubber system. If the building air pressure becomes positive by more than 0.63 kPa, blowers force air through the scrubber. (The blowers can turn on without a building seal.) If the blower motors turn on, this starts the filter bed pumps. Indicator lamps at the console inform the operator of the system's condition.

7.3.1.3 Emergency Decontamination System – This system, described in Sec. 6.2.3, is operated manually by a switch in room 1328. Air flow through the system is checked visually at the filter bank with a manometer.

7.3.1.4 Emergency Core Cooling System (ECCS) – Control of the ECCS, Sec. 6.3, is by means of a manual valve or an electrical pushbutton in the reactor containment room or by a keyed electrical pushbutton at the console.

7.4 Systems Required for Safe Shutdown

There are no specific systems needed for shutdown since loss of power will allow gravity to drop the shim rods and scram the reactor. Decay heat is sufficiently small so that forced cooling is not required. Although an emergency core cooling system is provided, total loss of water will not lead to core melting, Sec. 15.2. Emergency power, described in Ch. 8, is supplied to the reactor console and engineered safety systems to insure system availability during power outages.

7.5 Safety Related Display Instrumentation

7.5.1 Description

Information is available to the operator for determining and adjusting control rod positions. The building and coolant radiation detector readouts are visible at the

Fig. 7.3-2 - Filter/scrubber system.

console. A water level alarm, water height gauge, and TV core monitor camera supply the operator with the information necessary to start the core spray in an emergency condition.

7.5.1.1 Reactor Power Readouts – Readout and chart recorders that provide reactor power displays are listed below. (See Fig. 7.1-1.) These include: (1) a linear level power meter and recorder, (2) a log N power meter and recorder, (3) a megawatt-day integrating recorder (signal from A-level safety amplifier), (4) a ^{16}N digital readout, (5) A, B, and C power level meters on level safety amplifiers, and (6) a log count rate meter and recorder start-up power indicator.

7.5.1.2 Reactor Period Readout – Reactor period is displayed on a log count rate period meter and a log N period meter. The period recorder is shared by means of a switch between the log count rate period signal at start-up and the log N period signal at higher powers.

7.5.1.3 Control Rod Position Readout – The regulating rod and each shim rod have extended scale meters that monitor their positions (five panel meters). An accurate ± 0.25 mm (± 0.01 in.) digital readout can be switched to any one of the five rods for precise measurement of position. See Fig. 7.1-3.

7.5.1.4 Composite Safety Amplifiers – Amplifiers A, B, and C have meters that monitor the percent of reactor power. Amplifier D has a meter that monitors the period. Each of these meters can be used to monitor the internal amplifier conditions by means of a multiposition switch. Each amplifier has a meter that monitors the magnet current.

7.5.1.5 Water Systems – Primary and secondary coolant flow and temperatures are displayed on chart recorders. In addition to causing a scram, loss of flow activates the panalarm and turns off the primary coolant pump. Pool water level is measured by manometers in the equipment pit and on the console. A low level condition actuates the panalarm. A primary coolant high temperature condition is displayed on the recorder and the panalarm and causes a scram.

7.5.1.6 Primary Coolant Chemistry – Figure 7.5-1 shows a block diagram of the primary coolant conductivity and resistivity monitors.

7.5.1.7 Area Radiation Detectors – Strip chart recorders and meters monitor the radiation detectors in the reactor containment building and those in the primary and secondary cooling water, Sec. 12.1.4.

Fig. 7.5-1 - Miscellaneous reactor monitors.

7.5.1.8 Atmospheric Parameters - Wind speed and direction are displayed on meters at the console, Fig. 7.5-1. The control has a barometer and wet and dry bulb thermometers.

7.5.1.9 Containment Building Surveillance - Two TV cameras monitor the reactor containment structure and one camera monitors the core. These are displayed at the console.

7.5.1.10 Panalarm System - A panalarm, shown in Fig. 7.5.2, indicates the status of the reactor.

7.5.1.11 Duct System Airflow - A pneumatic dial indicator displays the vacuum level in the duct system intake. A lamp on the console indicates when the duct blower motor is on.

7.5.1.12 Bypass Status - An administratively controlled panel has the following bypass switches and indicator lamps:

- Reg rod upper limit
- Reg rod lower limit
- Shim range rod A
- Shim range rod B
- Shim range rod C
- Shim range rod D
- Low coolant flow
- Building seal
- Period recorder
- Period amplifier.

7.5.1.13 Display Indicator Lamps - The console has the following indicator lamps:

- "ON" lamps for main coolant pump, duct blower, and exhaust fan.
- "Cause of Scram" lamps for recorder period scram, linear level recorder scram, and A, B, C, and D composite safety amplifiers.

	1	2	3	4
S1	Scram	Scram-manual	Scram low coolant flow	Experimental scrams
S2	Rundown	Amplifier trouble	Primary coolant hi temp scram	Primary coolant hi temp
S3	Bldg seal hi radiation	Bldg seal manual	Regulating rod manual operation	Rad restriction
S4	Stack hi radiation	Ducting hi radiation	Primary coolant radiation	Secondary cool. hi radiation
S5	Reactor top radiation	Balcony hi radiation	W thermal col. radiation	Room No. 6 S hi radiation
S6	Room No. 5 NE radiation	Room No. 4 NW hi radiation	Regulating rod limit	Demineralizer hi radiation
S7	Air monitor hi radiation	Water monitor hi radiation	Hallway hi radiation	Primary coolant conductivity
S8	Pool hi level	Pool low level	Hi level sump pit	S-tubes low level
S9	Waste 1 and 2 hi level	Pool water low temp	Storage hi level	Through tube hi radiation
S10	Stack valve	Air conditioning inlet valve	Air conditioning return valve	Recirculation damper
S11	Air lock	Doors	Low air pressure	Secondary coolant low level
S12	Bypass in circuit	Clutch switch	Experiment alarm	Power failure

Fig. 7.5-2 - Annunciator CO815.

- Lamps for "withdraw permit," "east thermal column door closed," and "containment building doors open,"
- "On/Off/Contract Failures" lamps for the filter/scrubber system pumps and blowers.

7.6 All Other Systems Required for Safety

Additional systems are added from time to time to govern the safety of an experiment that interacts with the reactor or that may have hazardous consequences. These systems are established in accordance with requirements of Sec. 13.4 and Ch. 17.0.

7.7 Control Systems Not Required for Safety

7.7.1 Description

The reactor is controlled by four shim rods and one regulating rod. The rod actuators are driven by 2-phase motors. The shim rods operate under manual control only.

7.7.1.1 Shim Rods – Shim rods can be withdrawn by manual switches one at a time or by a gang switch four at a time. In order to withdraw the shim rods, the shim rod permit circuit must be enabled. To energize the permit, the following conditions must be met:

- The scram chain, Sec. 7.2.1, must be completed.
- The reactor escape, truck doors, and the east thermal column door must be closed.
- The count rate meter must be in the operate mode.

Shim rods can be inserted at any time by the manual switches. All of the conditions must be constantly met or the system will automatically insert all shim rods (rundown). Automatic rundown inserts all the shim rods, prevents withdrawal of any rod, and forces the regulating rod system to manual control. Rod insertion stops when the condition initiating rundown is corrected or when the insertion limit is reached.

7.7.1.2 Regulating Rod – The regulating rod is controlled by a manual switch or by an automatic control when the shim rods are in the operating range, the regulating rod is within its range, and the system is not in a rundown condition.

7.7.1.3 Automatic Control – After the reactor has been manually taken to power, it may be desirable to automatically keep the reactor at a fixed power level. To do this, the regulating rod motor is placed in a control loop, Fig. 7.7-1, that holds the power at a level determined by the power set potentiometer and sensed by the linear level ion chamber. For more accurate power control (the chamber is subject to error due to flux pattern changes), the ^{16}N control system¹ is switched into the control loop. Figure 7.7-1 shows the regulating rod system. The ^{16}N detector

Fig. 7.7-1 – Regulating rod control.

¹H. P. Spracklen, IEEE Transactions on Nuclear Science NS-14 (1), 271 (1967).

examines the primary coolant discharge from the core and measures the ^{16}N gamma activity. This activity is directly proportional to the reactor power regardless of the rod positions.

7.7.2 System Analysis

In the event of a complete failure of the control system in a mode that would cause maximum withdrawal of the regulating rod and failure of the linear level system at the same time, one slow scram capability could be lost (the trip on the linear level recorder). All other scram trip circuits would be operational and the reactor would trip at a preassigned power level (Sec. 7.2.1.4). Refer to Sec. 4.3.2.6 for discussion of reactivity insertion rates for automatic and manual modes.

CHAPTER 8.0 ELECTRIC POWER

8.1 Introduction

The electrical system at Building 281 is energized from an on-site, underground, 12-kV feeder, and distributes the power to the auxiliary loads, including the instrumentation and control systems for reactor operation. The electrical system is a radial, cascaded distribution system with an emergency power source.

The a-c electrical system starting at the 480-V outdoor unit substation, 160A at Building 281, is energized from transformer T160 (see Fig. 8.1-1). This system includes distribution panels 160A5, 160A5-5, 160A5-7, 160A5-23, the 50-kW emergency generator, and the automatic transfer switch.

A secondary distribution system (208/120-V, three phase, four-wire) is energized from transformers T-161, T-162, T-199 and T-574.

There are two separate battery-powered d-c emergency systems: 1) a 120-V battery which operates the criticality alarm system; and 2) 24-V battery which energizes the fire alarm and building equipment alarm systems.

8.2 Offsite Power System

Power at the Livermore site is purchased from Pacific Gas and Electric Company (PG&E) and is distributed on the property over an AEC-owned distribution system. Two 115-kV transmission lines feed the main substation, No. 1, at the Laboratory. One line comes in from the east and the other line comes in from the south.

8.3 Onsite Power Systems

8.3.1 A-C Power Systems

8.3.1.1 Description – Distribution on the Laboratory property is from Substation No. 1 at 13.8 kV, see Fig. 8.3-1. Area Substation No. 5, which supplies the reactor building, can be energized from either one of two 13.8-kV underground feeders out of Substation No. 1, a distance of approximately 4500 ft. The 12-kV feeder which energizes the 480-V unit substation at Building 281 comes from Area Substation No. 5, a distance of approximately 1000 ft.

8.3.1.1.1 480-V Power System – The 480-V distribution system energized from the outdoor unit substation includes motor control center (MCC) 160A-6 at the reactor cooling tower, and distribution panel 160A-5 in room 1328 at Building 281.

Fig. 8.1-1 Block diagram of LPTR electrical distribution systems.

REFERENCE DRAWINGS —

- LEA72-G168-11 SINGLE LINE DIAGRAM
- LEA72-G168-21 PANEL SCHEDULES
- LEA72-G168-31 POWER SYSTEMS LAYOUT

NOTES —

1. THIS DRAWING PORTRAYS THE DISTRIBUTION SYSTEM THROUGH THE UNIT SUBSTATION, MOTOR CONTROL CENTER (MCC), AND DISTRIBUTION PANELS AND LOAD CENTERS.
2. FEEDERS 1 AND 2 ARE COORDINATED SO ONLY ONE FEEDER IS IN SERVICE AT ANY TIME, SEE LEA72-G168-11.
3. THE NUMERICAL DESIGNATION FOR EMERGENCY PANELS AND LOAD CENTERS IS PRECEDED BY THE LETTER "E".

2

Fig. 8.3-1. LPTR onsite power system.

Panel 160A-5 in turn energizes distribution panels E160A5-5, 160A5-7 and 160A5-23 along with other auxiliary 480-V loads. For the electrical sequence of the 480-V and emergency power systems see Figs. 8.3-1 and 8.3-2.

8.3.1.1.2 Emergency Power System – The emergency power system is required only for monitoring the reactor facility upon loss of normal (PG&E) power. Emergency power is not required for the safe shutdown of the reactor or for the dissipation of decay heat. This system is served from emergency panel E160A5-5. Under normal conditions this panel is energized from 480-V Panel 160A5. Through an automatic transfer switch, emergency panel E160A5-5 is fed from the 50-kW emergency generator during periods of interruption to normal (PG&E) power.

Emergency panel E160A5-5 (480 V) feeds the reactor primary circulating pump, the foam generator, and Transformer Bank T-162. The secondary side of this transformer bank energizes 208/120-V panel E162A. This panel feeds three distribution panels; E162A4, E162A10 and E162A15; and four load centers; E162A3A, E162A3B, E162A9 and E162A16 which make up the 208/120-V emergency power system for the reactor. Panel E 162A-4 in the control room (1355) energizes the entire reactor control system.

The various 208/120-V loads supplied from these seven panels and load centers are listed on the panel schedules on Fig. 8.3-3.

The arrangement and locations of the components for the 480-V and emergency power systems at building 281 can be seen in Fig. 8.3-4.

8.3.1.2 Analysis

8.3.1.2.1 A-C Systems Fault Protection – The two main feeders which energize area Substation No. 5 from Substation No. 1 are equipped with instantaneous and inverse time overcurrent relays which operate on phase-to-phase or phase-to-ground faults to open the oil circuit breakers and clear the circuits. The feeder between area Substation No. 5 and Transformer Bank T-160 at Building 281 is protected by three 50-A fuses, one in each phase.

The 480-V bus in the unit substation and the underground feeder which energizes distribution panel 160A5 are supplied through drawout type air circuit breakers in the unit substation. These breakers are also equipped with instantaneous and inverse time overcurrent tripping attachments.

All circuits, 480-V and 208/120-V energized from distribution panels and load centers, are supplied from molded-case circuit breakers equipped with thermal overload tripping devices.

8.3.1.2.2 Loss of Power to the Reactor – Loss of normal (PG&E) a-c power at the reactor building results in a reactor scram and a transfer to the emergency power source. Loss of power de-energizes the safety amplifiers and the magnet power supplies, allowing the four shim rods to fall into the reactor core, thereby scrambling the reactor. As a backup the loss of power also interrupts the interlock chain, which must be complete for reactor operation. This interruption alone would scram the reactor.

The loss of normal power also actuates an undervoltage relay which operates the automatic transfer switch and starts the emergency 50-kW generator.

8.3.1.2.3 A-C Systems Component Capacities and Ratings – The buses, high voltage conductor sizes, circuit breaker ratings and interrupting capacities, and low voltage feeder cable sizes are of sufficient electrical rating for reliable operation of the a-c electrical power systems.

The primary 13.8/12-kV system cables from Substation No. 1 to area Substation No. 5 and on to Transformer Bank T-160 are 15-kV varnish cambric lead-covered cables run

Fig. 8.3-2. Single-line diagram of LPTR 480 V and emergency power systems.

Fig. 8.3-3. Panel schedules for LPTR emergency power system.

Fig. 9.3-4. Layout for LPTR 480 V and emergency systems.

in underground ducts. The secondary 480-V and 208/120-V system cables run in conduits, some underground and others overhead within the building. These duct and conduit runs protect the various electrical system cables from physical and mechanical damage. They isolate the various voltage levels from intersystem electrical faults and protect operating personnel from electrical hazards.

The a-c electrical systems components and installations are in accordance with that edition of the National Electrical Code which was current during the design and construction of the reactor and facility.

8.3.2 D-C Electrical Systems

8.3.2.1 Description – Twenty 6-V storage batteries in series, housed in the battery building north of the reactor containment building, provide emergency power for the criticality alarm and the building evacuation systems. See Fig. 8.3-5. This battery system furnishes 120 V and 48 V (from a tap after the eighth battery). Three conductors (+120 V, +48 V, and -common) run in conduit to the control room. Two separate battery chargers located behind the control room console, one at 120 V and the other at 48 V, keep the battery system at full charge and rated voltage.

8.3.2.1.1 120-V System – The 20-battery system exists only to power four Klaxon horns. Each horn draws approximately 1 A at 120 V d-c. They are audible throughout Building 281 and serve as an evacuation warning.

8.3.2.1.2 48-V System – The 48-V section of the battery system supplies control power both for the criticality alarm system and for the building evacuation horns.

In addition, the criticality alarm system includes an inverter that provides 115-V a-c output. This powers two criticality monitoring sensors in the reactor building and three radiation monitoring sensors in the hallway of Building 281.

8.3.2.1.3 24-V System – The 24-V battery, located in the machinery room (1306) supplies power for the fire alarm and building equipment alarm systems, see Fig. 8.3-5. The fire alarm system and the battery charger are also located in room 1306, with the battery. The fire alarm system operates on 24 V d-c and responds to detectors in four zones; the west wing, central building, reactor building, and two small buildings north of the reactor.

The building equipment alarm panel, located in the hallway near the west entrance, operates on 24 V d-c. The d-c conductors run in conduit from the battery to the panel in the hallway. This system has four channels; reactor pool water level and coolant

Fig. 8.3-5. D-C power for LPTR fire, building equipment, and criticality alarm systems.

outlet temperature, containment building air pressure, reactor radiation level, and the criticality alarm.

8.3.2.2 Analysis — Figure 8.3-5 indicates the measures that protect the two battery systems from electric faults. The 120/48-V system is protected by a 20-A molded-case circuit breaker, and the 24-V system is fused through 6-A fuses.

The 48-V portion of 120-V system, which feeds the inverter in the control room (1325), comes into service upon loss of normal and emergency power. The Klaxon warning horns may be energized from 120 V d-c at any time.

Loss of emergency power deenergizes the two chargers that maintain the 120-V battery. This battery bank will continue to operate the criticality alarm system under these emergency conditions for approximately 8 hr.

Loss of emergency a-c power deenergizes only the battery charger on the 24-V system. The battery itself will maintain the fire alarm and equipment alarm system under emergency conditions for 8 hr.

The d-c systems component and installations are in accordance with those editions of the National Electrical Code that were current during their installation.

CHAPTER 9.0 – AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

New fuel elements are stored in the laboratory's central control vault facility in accordance with Materials Management procedures. This facility is specifically designed for the safety and physical security of nuclear materials. A new fuel element is brought to the reactor by Materials Management personnel and is thereafter hand-carried to the reactor top for insertion into the reactor tank. New fuel elements can be moved without special handling equipment once radiological surveys confirm freedom from alpha contamination.

9.1.2 Spent Fuel Storage

Spent fuel is stored in two locations, primarily in the spent fuel storage rack inside the reactor pool tank (Sec. 5.4.4), and occasionally in the floor storage pit. Figure 9.1-1 shows the spent fuel storage rack located 2.6 m above the tank bottom within the reactor tank. Shielding is adequate for all operating conditions.

The floor storage holes (3.3 m deep) in the storage pit, Fig. 9.1-2, can be used for spent or hot fuel element storage in case the reactor tank requires draining. From experience, each floor storage hole (with 3 m of water) can accommodate two irradiated fuel elements. Adjacent storage holes must be filled with water to minimize radiation streaming.

9.1.3 Spent Fuel Fuel Cooling and Cleanup System

(not applicable)

9.2 Water Systems

9.2.1 Station Service Water System

(not applicable)

9.2.2 Cooling System for Reactor Auxiliaries

(not applicable.)

Fig. 9.1-1 Cross section of reactor showing the spent fuel storage rack.

Fig. 9.1-2 Fuel element floor storage pits, plan and cross section.

9.2.3 Demineralized Water Makeup

(See 9.3.4.2)

9.2.4 Potable and Sanitary Water Systems

Normal (potable) plant water plus low conductivity water (LCW) are piped into the building. Primary usage of plant water is for filling shielding tanks, equipment cooling, and for makeup water to the cooling tower.

9.2.5 Ultimate Heat Sink

The ultimate heat sink for normal power and shutdown operations is the atmosphere, through the mechanism of the secondary coolant system and the cooling tower.

During normal shutdown condition the primary coolant may continue to flow at $67 \text{ dm}^3/\text{s}$. The automatic secondary valve continues to allow flow until the primary outlet temperature drops below 41°C . (Normally the secondary pump is shut down manually at scram to prevent overcooling due to the 180 s delay tank in the primary loop.) During reactor shutdown the secondary pump automatically holds the primary outlet temperature at $\approx 34^\circ\text{C}$ by kicking on whenever the primary temperature reaches that level. The primary coolant pump will cause a rise in pool temperature (after most decay heat is gone) due to the work input necessary to pump against the frictional head.

9.2.6 Condensate Storage Facilities

(not applicable)

9.3 Process Auxiliaries

9.3.1 Compressed Air System

Compressed air is plant supplied at 630-700 kPa. An alarm sounds on the control room panel if the pressure drops below 530 kPa. Plant air is used for various instruments, control of secondary coolant automatic valve (V-97) and bypass valve (V-155), and for the building seal system. With the exception of those listed below, there is no emergency air system to the building. In normal outages, a portable compressor can be connected outside the building to supply air.

The building seal system reserve tank contains enough air for five seals and has a backup nitrogen bottle. The tank has a check valve in the event of unplanned outages. In the event of a complete loss of air to the building seal system (including the reserve), the building seal dampers will close.

Plant air is filtered for contaminants (such as oil) and is dried after leaving the compressor. Filtration is as low as 1 micron, depending on the usage. Plant air supplied to the shop and lab areas is filtered and regulated according to need.

9.3.2 Process Sampling System

Pool water (P-W), experimental facilities water (S-1 through 4), and secondary water (S-W) are sampled regularly and tested for radioactivity. The S-1 samples read ≈ 15 mR/h maximum at contact. Typical results are displayed in Table 9.3-1.

Table 9.3-1 Typical radioactive contamination levels in LPTR water samples.

	Sample volume cm^3	Alpha $\mu\text{Ci}/\text{m}^3$	Beta $\mu\text{Ci}/\text{m}^3$
S-1	125	0.65	7 000
S-2	125	0.27	4 800
S-3	125	<0.20	500
S-4	125	<0.20	1 700
S-W	500	<0.006	0.03
P-W	125	<0.20	40

9.3.3 Equipment and Floor Drainage System

The drainage system, Sec. 11.2, for the reactor building consists of four floor drains, two west thermal column drains, an east thermal column drain, a magnet pit drain, and a sink drain. These all flow to either of two (5.7 m^3) waste tanks in the equipment pit. All beam port vestibules, through-tube vestibules, S-tube vestibules, the reactor overflow, and the gutter around the equipment pit have drains that flow to the sump in the equipment pit. The sump holds 170 dm^3 and has a high level alarm. The fluid in the sump can be pumped to the waste tanks in the pit or to waste tanks outside. The equipment pit waste tanks must be pumped to outside tanks. Each of these tanks has a high level alarm.

On the reactor balcony and on the floor around the reactor 0.15 m from shielding face is a small dam. Any water loss in these areas flows to one of the floor drains. All drains in the reactor building flow to either the sump or to waste tanks. No drainage flows outside the reactor building. Certain laboratory rooms (Chemistry, Room 1311, target room 1323, and rooms 1184, 1187) have hot sinks which drain to two retention tanks north of the lab building. All other drains in the labs drain to the plant system.

The outside retention tanks alarm when the tanks are full (3.8 m^3). Secondary water can be drained directly to the storm sewer at the cooling tower.

9.3.4 Chemical and Volume Control System

9.3.4.1 Primary Coolant - Chemical checks of the primary coolant are made on a demand basis. Most recent chemical analysis shows Ag, Al, B, Be, Ca, Cr, Cu, Fe, Mg, Mn, Ni, Pb, Si, and Te present in amounts varying from 4 to 1000 mg/m³ (4 to 1000 parts/billion). No pH control of the primary exists. The primary coolant is filtered (1 μ m) continually at ≈ 0.6 dm³/s. It is typically run through an ion exchange system (one mixed bed, anion and cation and one single anion level) every weekend (starting 24 h after shutdown to allow for decay) at 6 dm³/s. Water purity is kept at < 1 Ω conductivity. In a typical week, water will start at ≈ 0.25 Ω on Monday and rise only 0.10 to 0.15 Ω by shutdown (≈ 100 -105 h). Weekend cleanup will bring it back down to approximately previous starting level. Previous experience shows that little or no corrosion problem exists in aluminum piping when demineralized water is kept within these limits.

9.3.4.2 Demineralized Water Make-Up System - Water make-up consists of manual filling using plant supplied demineralized water (D. W.). No automatic fill system is used due to small amount of losses (normally 2-3 mm³/s or 6-8 mm of pool level each day). With present piping, D. W. can be added to the pool at the maximum rate of 0.95 dm³/s. D. W. is supplied to the building through PVC piping in order to keep conductivity at ≈ 0.4 Ω .

9.3.4.3 Secondary Coolant - Secondary coolant is monitored for pH and is chemically treated for both piping and tower (redwood) preservation. Sulphuric acid (93% technical grade) is added automatically to keep the pH between 6.8-7.1.

Nonoxidizing microbicides for micro-organisms and corrosion inhibitor and antifoulant are also added to the secondary on a demand basis.

Secondary water makeup is automatic and is recorded daily (except weekends) by the water group of Maintenance Machinist Department. Small losses are impossible to detect as normal evaporation and losses due to water being blown out by the draft fan vary from 0.01 to 1.7 dm³/s.

9.3.4.4 Potential Water Losses (Primary) - Primary water losses outside the containment building can come from three sources. These are as follows:

- The main coolant pipe from the reactor core to the 11 m³ hold up tank and from there to the equipment pit. In the event of a leak in this part of the system, the system would have to be drained. Piping in this area is 8 in. Sch. 40 aluminum 5052 (see Ch. 5.0).

- The storage tank (TK-1) is also outside the containment (below reactor floor). If this tank leaks, the water can be drained to the equipment pit sump and pumped to waste tanks outside the building. Leak detection from this tank is difficult; the primary indicator of level is a manometer in the control room, which is recorded weekly. Losses of 300-400 dm³ could be detected. The tank normally has less than 6 m³ in it, and this usually is fresh D. W. (contaminated only from activity left by previous fluids).
- The other possible loss would be heat exchanger tube leakage, allowing primary water to pass into the secondary. Losses would be noted by a rise in the secondary activity. If the leak were large it would be indicated on the control panel alarm; if it were small it would show as a rise in the weekly secondary water sample.

Primary water losses inside containment are detected primarily by pool level changes. They are also checked by manometers in the pit and control room.

9.3.5 Failed Fuel Detection System

Failed fuel is detected principally by two means. A sensor is provided which detects shorter half life radioiodines, such as ¹³⁴I and discriminates out long-lived radioiodines, such as ¹³¹I. Long-lived iodines are present as a result of diffusion through the clad. ¹³⁴I, on the other hand, is too short-lived to escape by diffusion. A second sensor detects neutron emitters in the primary coolant. The neutrons (primarily from high iodines and bromines) represent delayed neutron decay of short-lived fission products in the coolant. An overabundance of either class of short-lived emitters suggests a clad failure.

9.4 Air-Conditioning, Heating, Cooling, and Ventilation Systems

9.4.1 Control Room

The control room is isolated from the reactor building by the air-lock doors and is heated and cooled by the air-conditioning unit located in the east mechanical equipment room, 1328, of the laboratory building. The air conditioner and ducts are described in Sec. 9.4.2. The control room receives heated or cooled air through five outlets. Figure 9.4-1 shows plan and elevation views of the control room and the air conditioning. Since many of the instruments in the console may be affected by temperature variations, the control room is provided with a partition that separates the front of the console from the back. The back of the console is kept at 21°C year-round while the front and area where the operator works is maintained at 22 ± 1°C year-round. The maximum relative humidity is approximately 50%.

D.A.V. = discharge air vent

R.A. = return air

Fig. 9.4-1 LPTR control room and air-conditioning layout.

9.4.2 Laboratory Building

The air-conditioning system for the laboratory portions of the building is designed to meet general comfort and air flow requirements under the following conditions:

- Winter extreme (outside) -4°C (db) with 6.7 m/s average wind velocity.
- Summer extreme (outside) 41°C (db) and 22°C (wb).
- A heating load of 1-1/2 air changes per hour for infiltration.

Conditioned or heated air for the east wing of the building is supplied to each room, except the restrooms, through a duct system with diffusers for each room. (Air for the restrooms is supplied through door louvers, and exhausted through roof ventilators.) Return air from most rooms ($26\text{ m}^3/\text{s}$) is exhausted through door louvers to the corridor, and from the corridor through a wall louver into the east mechanical equipment room, Room 1328. This recirculated air is replenished with $3\text{ m}^3/\text{s}$ fresh air drawn through an exterior wall louver on the south side of the mechanical equipment room.

The doors of each air-conditioned room, except the chemistry laboratory, are provided with door louvers to permit air to return to the corridor. Conditioned air supplied to the chemistry laboratory, Room 1311, is not permitted to recirculate since it might become radioactively contaminated. Air supplied to this laboratory is exhausted through the roof by fume hoods.

The laboratory building is heated by re-heat coils. Heaters are two-row return bend types in the main duct lines in the corridor so that air is heated before entering any of the rooms. Cooling is accomplished by a self-contained 15 ton air-conditioning unit. The total heat flow to be extracted by the air-conditioning unit is 440 kW in order to maintain an ambient temperature of 24°C db when outside temperature is 41°C db. Under these design conditions the unit is capable of extracting 492 kW.

The self-contained air-conditioning unit for the east portion of the laboratory building is located in the east mechanical equipment room. It consists of a flat type cleanable filter section; 0.61 by 1.5 m - five rows deep, 3 m/s velocity direct expansion coil with a final temperature of 13°C db and 11°C wb, 27°C entering water temperature and is good for 14 tons, or a total load of 492 kW. Approximately $3\text{ dm}^3/\text{s}$ of cooling water is required. Two forwardly curved, cleanable fans deliver $28\text{ m}^3/\text{s}$ at 0.25 kPa external static pressure. The fans are mounted on a common shaft driven by a 2.2 kW motor operating on 440 V, 3-phase, 60 Hz current. A 11 kW, 29 rev/s, 440 V, 3-phase 60 Hz motor is used to operate the cooling unit compressor.

Air conditioning and heating to the west portion of the laboratory building are shown in Figs. 9.4-2 and 9.4-3. The air-conditioning equipment located in the west mechanical equipment room (Room 1200) supplies chilled air to the offices and laboratory spaces. Heated water supplied to heater coils in each room maintains the temperature at approximately 22°C for winter and summer conditions. Air is returned to the central air conditioner via ducts in the ceiling of each room. Thermostats are provided in each room to control the air temperature.

9.4.3 Radwaste Area

As radioactive waste is generated at the reactor site it is transferred to laboratory facilities elsewhere. Potentially contaminated liquid waste is collected in tanks, sampled, and then transferred to the treatment plant at Building 514. Dry waste, e.g. paper, glass, small equipment, gloves, etc., is collected in processing drums and transferred to the Dry Waste Processing Facility at Building 612. These are closed systems not requiring auxiliary systems support.

9.4.4 Turbine Building

(not applicable)

9.5 Other Auxiliary Systems

9.5.1 Fire Protection and Alarm System

9.5.1.1 Fire Protection – In the reactor containment building there is a manually operated high expansion foam system, located in the top of the dome. This system is capable of filling the reactor building with foam in 540 s. This system uses a detergent-water mixture sprayed onto a screen and formed into foam by a 12 m³/s fan. Tests have shown that such systems are capable of smothering a fire quickly without injury to personnel or equipment. However, this system would not be used unless a fire could not be controlled by other means.

There are fire extinguishers of various types placed throughout the reactor and laboratory buildings. There are also Scott airpacks and fire hoses in both buildings. This equipment is checked by the fire department and kept in a ready condition.

Fire trucks carry additional fire protection, communications, and radiological safety equipment to be used as necessary.

9.5.1.2 Alarms – Radiation and Fire Protection – The fire house alarm system from Building 281 consists of three different types of alarms; fire alarm, building equipment alarm, and criticality alarm.

Fig. 9.4-2 Air conditioning floor plan, north half of Building 281.

The fire alarm panel's power supply and battery charger are located in the west equipment room (Room 1306). The fire alarm system works on 24 V d-c. The building is divided into four zones. Zone 1 is the west wing, Zone 2 is the central part of the building, Zone 3 is the reactor building, and Zone 4 includes the two small metal buildings north of the reactor containment building.

The fire alarm works on a closed, supervised system. All detectors are closed, and open only on a rise of temperature. Any kind of trouble, such as a broken conductor, bad relay, or loss of power, will broadcast a trouble alarm to the fire house, Building 323, and a buzzer will sound in the annunciator panel in the hallway. An amber light will light up at the same time indicating the affected zone. In case of power failure no light or buzzer will operate in the annunciator panel, as this panel gets its power from the same source as the fire alarm panel. An alarm will sound in the fire house, however, and this alarm will stay on until the trouble is corrected. In case of fire a red light on the annunciator shows which zone has the fire, and the fire alarm bells will ring.

The building equipment alarm panel which includes the fire alarm panel is located in the hallway, near the west entrance. It operated on 24 V d-c (Sec. 8.3.2), and is powered from the fire alarm batteries. The panel has four zones. Zone 1 is low water level in reactor, Zone 2 is air overpressure in the reactor building, Zone 3 is radiation in reactor and Zone 4 is criticality alarm. The panel also acts as an annunciator for the fire alarm panel.

The building alarms operate on a supervised open alarm system. In case of trouble, a contact associated with protected equipment will close and the alarm will sound as explained for fire. The buzzer in the annunciator sounds for trouble and also for reset. In case of circuit failure the amber light will come on and the buzzer will sound. After the circuit is restored to normal, the circuit affected will have to be reset. After the circuit has been reset, the panel will have to be reset as explained for trouble.

The criticality alarm is for building evacuation in case of high radiation (see Fig. 8.3-5). The main panel and power supply is located in the metal building north of the reactor building.

The main power supply consists of a 120 V battery bank, an inverter to change 48 V d-c to 120 V a-c and two battery chargers, one 120 V and one 48 V. (See Sec. 8.3.2.)

The 120 V section of the power supply operates the three Klaxon horns on a supervised system. Trouble on any of the horn circuits will send a trouble alarm to the fire

house. The 48 V section of the power supply operates the supervised alarm panel that controls the Klaxon horns, and also supplies power to the hallway RAMS panel through the inverter in case of building power failure.

The power supply for the RAMS console in the west hallway entrance is from building power through the main control panel. If a power failure should occur the RAMS will transfer to battery power through the inverter.

The three Klaxon alarm horns are connected in series with a supervising relay and a potentiometer to adjust flow of current through the relay. If any horn circuit becomes disconnected, horn breaks down, or fuse blows, an amber indicating light will turn on and a buzzer will sound.

The criticality alarm may be sounded by pushing the manual alarm switch and stopped by pushing manual alarm silence switch. When an alarm is triggered by the RAMS detector the horns cannot be shut off as long as high radiation exists except by a key-operated silence switch. When key-operated switch is turned off, a red light will turn on to show alarm is out of operation. After an alarm has sounded, and if the RAMS unit has reset itself, the horns can be silenced by pushing manual alarm silence. A key-operated switch installed in the main control room will operate the same as the key switch on the panel by the west entrance door.

Two RAMS detectors located in the reactor building are the only detectors that will trigger the evacuation alarm. Both detectors must operate to trigger alarm.

Three detectors in the hallway sound a local radiation alarm chime in the west end of the hallway near the panel only. The three hallway RAMS detectors and the two reactor building RAMS detectors can be individually read out or tested from the hallway panel.

9.5.2 Communications Systems

The communication systems for Building 281 includes paging, intercoms, closed circuit TV, and telephones.

9.5.2.1 Intercoms - The intercom system consists of three 60-station teletalk intercommunication master stations and 36 individual sender-receivers. Two master units are located in Room 1173, the Main Office, and the other is mounted in the Control Room Console, Room 1325. Each master station consists of a cabinet containing an amplifier, a speaker-microphone, and the switches necessary for selecting the desired individual receiver. The speaker locations for the intercom system are shown on Fig. 9.5-1.

Fig. 9.5-1 Intercom and paging system speaker locations in Building 281.

9.5.2.2 Laboratory-Wide Page System — The laboratory-wide page system for Building 281 is shown in Fig. 9.5-2. The system is initiated from the Police Station or Fire House and uses telephone lines to Building 281. The signal is sent directly to the priority unit which overrides any signal input from the teletalk master stations in Building 281 and the microphone in Building 151. From the priority unit, the evacuation page signal goes out to the Altec amplifiers and then to the speakers. Testing of the evacuation page system in Building 281 is accomplished by 400 cycle tone, tape recorder, and microphone from the Fire Department. An annunciator flag on the Fire Department annunciator panel indicates when a good signal transmission for Building 281 is accomplished by any of the three methods of testing.

9.5.2.3 TV Monitoring — The closed circuit television system for Building 281 consists of three television cameras in the Reactor Room 1350 and two television monitors in the Control Room 1325. The three television cameras in the Reactor Room are located as follows: One on the northwest crane structure 9.8 m above the floor; one on the southwest crane structure at the same height, and one on top of the reactor looking down into the pool at the core. The television monitors and remote controls are located in console racks in the Control Room. The crane structure

NOTE:
 P1-600 Ω, 1 PAD
 R2-50 Ω, 1 PAD
 T1 AND T2-UTC MODEL CG137

Fig. 9.5-2 Paging and evacuation page system, Building 281.

mounted TV cameras are selected for monitoring by the operator activating a switch on the reactor control room panel. This panel provides pan, tilt, zoom, and focus by remote control. The pool TV camera is never switched.

9.5.2.4 Telephones - The distribution of telephone instruments for Building 281 is shown in Fig. 9.5-3. There are eight incoming lines into Building 281.

Fig. 9.5-3 Telephone distribution in Building 281. Numbers in parenthesis refer to numbers of telephones.

9.5.3 Lighting Systems

The Reactor Building is adequately lighted by 28 mercury vapor lamps (500 W each) hung high in the dome. These lights are not supplied with backup power. When normal power to Building 281 fails the emergency 50 kW generator comes on, lighting five 300-W incandescent lamps in the Reactor Room. Two of these are suspended from the horizontal crane truss, north of the Reactor, and are controlled by breaker

E16A10-2. Two more are suspended from the horizontal crane truss south of the Reactor, and one, suspended from the dome, is located above the Reactor pool. These three are controlled by breaker E162A10-1.

If the emergency generator should fail to start, an auxiliary lamp unit with self-contained battery will come on. It is located on the east side of the SW I beam support for the overhead crane system.

Figure 9.5-4 shows the locations of all these emergency lights.

Fig. 9.5-4 Emergency lighting layout, reactor room.

9.5.4 Gasoline Generator

A 50 kW gasoline generator is started upon loss of normal power, Sec. 8.3.1.1.2. The generator, located just south of Room 1328 outside the auxiliary laboratory building, is fully self-contained. Gasoline is stored in a tank, which is part of its mounting

chassis. The generator is not required for the safe shutdown of the Reactor, but merely supplies backup power for monitoring and certain nonnuclear emergency systems.

CHAPTER 10.0 - EXPERIMENTAL SYSTEMS

10.1 Introduction

The LPTR is designed to provide a variety of neutron and gamma radiations to experimental apparatus or substances placed either in the core, near the core, or outside the shield. This chapter is specifically intended to provide a more detailed description of services and certain scientific experiments at the reactor.

Chapter 1.0 of the SAR provides a brief description of the overall facility and of the general industrial equipment available in the complex.

10.2 Irradiation Facilities

10.2.1 In-Core

The Reactor core consists of a 5×5 array of plate-type fuel elements with beryllium reflectors along two sides and with the central element replaced by a round tube. This latter item is the core central irradiation facility. There are also general irradiation facilities in two corner beryllium reflectors. The fuel elements may be moved so as to accommodate an in-core experiment or to enhance the neutron flux in a particular location; for example, a beam tube. Figure 10.2-1 shows a plan view of the core irradiation facilities. Two pneumatic rabbit systems go into the reactor core. One system, driven by positive pressure, enters a beryllium reflector element. The other, using a negative pressure, may go into either the west thermal column (WTC) or a beryllium reflector element.

10.2.2 In-Tank

Six horizontal beam tubes are installed in the tank about even with the central plane of the core. S tubes curve down into the tank, stopping above the core center plane just outside the beryllium reflector elements. Figures 10.2-2 and 10.2-3 show the relative locations of the in-tank irradiation facilities.

Each of these beam tubes and the through tube terminate at the reactor face (exterior of biological shield). Various experimental apparatus and shielding can be installed in or outside the beam tube. Each beam, with exception of the through-tube beam, can be turned off by closing a powered shutter.

Fig. 10.2-1. LPTR in-core irradiation facilities.

10.2.3 Thermal Columns

Two thermal columns (graphite piles) provide a variety of neutron energy spectra. Figure 10.2-4 shows a plan view as well as an outer face sectional view. The east thermal column (ETC) contains the fast neutron irradiation facility (FNIF) (Sec. 10.3.2).

A void in the ETC permits a thermal neutron beam to be extracted from the ETC door.

10.2.4 Pneumatic Transfer Systems (Rabbit Systems)

A pressure driven fast rabbit system is available that can push a capsule from the reactor core to the experiment receiving station in less than 3 s. This system is very useful for experiments utilizing exact timing and or very short half-life isotopes. The capsules for this system can be as large as 63 mm long by 25 mm diam. The terminal location in the core is the A-1 position. Neutron flux values for this position are listed in Sec. 10.3.

Notes: All dimensions in inches. One in. = 25.4 mm.

Fig. 10.2-3. Core irradiation facilities and beam ports.

Fig. 10.2-4. LPTR thermal columns.

A slow pneumatic transfer is provided to two core locations; position A-7 and the ETC, position R-2. The system utilizes a vacuum for pulling the capsules into and out of the core. Transfer times between the core and the experiment receiving area average 10 s. The capsule is roughly 110 mm long and 34 mm diam.

Temporary pneumatic transfer systems can be set up when required so long as there are no unresolved reactor safety issues.

10.3 Dosimetry

10.3.1 Thermal Neutron Fluxes

Although neutron fluxes and gamma dose rate are generally known, their exact value changes from time to time depending on nuclear poisons in the core and/or experimental apparatus present. Table 10.3-1 lists the approximate values of thermal neutron

Table 10.3-1. LPTR 3 MW thermal fluxes.

Facility	Thermal flux 10^{12}	Cadmium ratio
C-4	71	2.84
A-1 (rabbit) ^a	19	2.5
C-1-S	34	1.4
E-1	19	2.5
A-7 (rabbit) ^b	19	2.0
R-2 (rabbit) ^b	1.6	4.4
S-1	5.1	6.2
S-2	4.1	3.8
S-3	4.5	3.4
S-4	7.3	5.3
WTC	$3 \times 10^{10} - 7 \times 10^{12}$	1-1000
D ₂ O tube	$2 \times 10^{10} - 8 \times 10^{12}$	4-10,000

} See Figs. 10.3-1
and 10.3-2.

^aRabbit - pneumatic transfer facility fast.

^bRabbit - pneumatic transfer facility slow.

fluxes. Figures 10.3-1 and 10.3-2 display the thermal neutron flux and cadmium ratio for various distances into the WTC. The ETC has similar geometry to the WTC, and flux values may be similar. A listing of flux values vs distance will not be shown, however, since a detailed flux map has not been completed since the major overhaul of 1971.

Fig. 10.3-1. Thermal neutron fluxes and cadmium ratios in the WTC at 3 MW. Stringer position 13 cm north and 13 cm below core centerline.

neutron flux and cadmium ratios for various distances from the front of the thermal column nose.

10.3.2 Fast Neutron Flux

The fast neutron flux listing is broken down according to energy. Table 10.3-2 provides two listings, showing total fluxes greater than 1 MeV and greater than 3 MeV for certain irradiation facilities.

Fig. 10.3-2. Thermal fluxes and cadmium ratios in the D₂O tube at 3 MW.

The D₂O tube located within the WTC graphite and protruding from the thermal column door allows a sample to be inserted to any desired position and removed while the reactor is at power.

Figure 10.3-2 shows the thermal neu-

Table 10.3-2. Fast flux in LPTR experimental facilities.

Facility	3 MW fast flux >3 MeV 10^{10}	3 MW fast flux >1 MeV 10^{10}
C-4	860	2800
E-1	220	720
S-1	15	51
S-2	3.2	11
S-3	4.7	15
S-4	12	38
A-1 rabbit	220	720
R-2	0.62	2.0

Fig. 10.3-3. Fast neutron flux ($E_n \geq 1$ MeV) in fast neutron irradiation facility (FNIF) at 3 MW.

A fission neutron energy spectrum is provided by irradiating two plates containing ^{235}U in the FNIF of the ETC.¹ Figure 10.3-3 shows the fast flux value as a function of distance from the plate.

10.3.3 Gamma Dose Rates

Gamma dose rates for 3 MW operations appear in Table 10.3-3 for certain of the in-tank irradiation facilities. Figure 10.3-4 shows gamma dose rate with respect to distance from the front of the WTC. Some data exist for the ETC facility.

Table 10.3-3. Gamma dose rates at 3 MW.

Facility	MR/h
C-4	2400
A-1	200
E-1	210
S-1	23
S-2	7.5
S-3	9.3
S-4	30
R-1	23

10.4 Summary Descriptions of Research Programs

A large variety of research programs have been pursued at the LPTR since its beginning. As one of LLL's major research facilities it contributes significantly to laboratory programs through beam port experiments, by production of radioisotopes, by irradiation of samples for neutron activation analysis, and by neutron radiography. Educationally, the

1. G. E. Cummings, G. V. Pefley, and A. W. Medcalf, Characteristics of the LPTR Fast Neutron Irradiation Facility, Lawrence Livermore Laboratory, Rept. UCRL-7513 Rev. 1 (1969).

Fig. 10.3-4. Gamma dose rates in west thermal column (WTC) at 3 MW.

the thermal neutron capture by a separated isotope. A Ge(Li) spectrometer system located at beam port B-4 is utilized for this work. It consists of two Ge(Li) diodes that can be used either in a three-parameter coincidence experiment (gamma-gamma-time) or separately for single event measurements. One detector is surrounded by an anti-Compton detector and is used for low energy (50 to 4000 keV) measurements; the other forms part of a three-crystal (pair) spectrometer that is used for high-energy (2-10 MeV) gamma ray measurements.

Among the nuclear level schemes that have been investigated are ^{181}Ta , ^{182}Ta , ^{156}Gd , ^{158}Gd , and several Ca isotopes. Future efforts will include investigation of level schemes for ^{160}Tb and ^{242}Am .

Equipment on hand includes several Ge(Li) detectors, several NaI(Tl) scintillator detectors, amplifiers, logic modules, a single parameter pulse height analyzer, and a four-parameter pulse height analyzer. A computer-based data acquisition system for multiparameter analysis is available. There is also a bent crystal spectrometer that is being developed and used for capture γ -ray studies.

10.4.2 Photodisintegration of ^9Be and D

Investigations of the nuclear structure of beryllium and deuterium have been made using gamma rays emitted by radioactive sources prepared in the center of the LPTR

LPTR has contributed much to the scientific community through the graduate students who have done their thesis work here. Eight doctorate and eight master's theses have been completed.

The continuing program at the LPTR involves basic and applied research in nuclear physics and chemistry using the beam ports and thermal columns; in-core and S-tube irradiations are a smaller but substantial part of the total activity. The more important programs are outlined below.

10.4.1 Capture γ -ray Experiments

An important technique used in nuclear structure research is the measurement of prompt gamma rays emitted following

for photodisintegration measurements. Total cross sections, angular distribution, and polarization of photoneutrons have been measured. Source strength required for these studies (2 kCi of ^{24}Na) makes it convenient to do this work in close proximity to the reactor.

10.4.3 Neutron Diffraction

Neutron diffraction techniques, using spectrometers associated with two of the reactor beam ports, provide information about magnetism and crystal structure. One is an automated powder diffraction spectrometer whose sample temperature can be varied from 2 to 773 K. The other is for the study of single crystals of material. The effects of crystal structure on the magnetic properties of several antiferromagnetic materials, e.g. FeS, NiS, and CrS, have been studied at various temperatures using these spectrometers.

10.4.4 Neutron Optics

Measurements of the coherent scattering amplitude of various nuclei have been made using a liquid mirror. A ribbon beam of subthermal neutrons is formed by filtering and collimating neutrons in one of the reactor beam ports. This beam is reflected from a liquid surface at very small angles and the reflected intensity measured. The relative intensity of the reflected beam is an accurate measure of the scattering amplitude of the nuclei making up the liquid mirror. This technique was used to measure the n-p coherent scattering amplitude which is fundamental to studies on two-nucleon systems. It was also used to measure the amplitude of strong absorbers like boron whose scattering properties are difficult to measure by other means.

10.4.5 Radiochemistry

Measurements have been carried out on one of the reactor beam ports to determine the charge states and chemical identity of the primary fission fragments of ^{235}U . Fission fragments from a thin ^{235}U foil at the front of the beam port are magnetically analyzed as they travel to the collector and then chemically analyzed after collection.

In a related experiment transuranic elements are fissioned in one of the thermal columns and retracted into a counter to study the short-lived beta emitters.

Many samples are irradiated as standards for other radiochemistry projects. One such set of calibration standards has been flown to various Air Force counters around the world.

10.4.6 Fission

The energy (and mass) distribution of fragments from neutron induced fission was studied by an experimental arrangement situated in the WTC. A strong component of symmetric fission in $^{257}\text{Fm} + n$ and $^{255}\text{Fm} + n$ was discovered in these experiments. A similar experiment is planned on ^{236}Np .

10.4.7 Biomedical

A resident group from the LLL Biomedical Department has set up a very sensitive analytic system using neutron activation analysis. The system identifies trace impurities found in various substances from biological and environmental samples.

10.4.8 General Irradiations

Numerous radioisotopes have been produced for various LLL groups for use in their particular programs. Also radiation damage studies have been carried out in support of the LLL nuclear propulsion and weapons programs. Detector calibration work has also been done for these programs. The FNIF located in the ETC has proved particularly useful for these studies.

10.4.9 Neutron Radiography

A neutron radiography facility is being developed in ETC of the LPTR. A hole in the graphite 1.52 m long accommodates a convergent-divergent collimator and terminates in an 0.51 m hole at the filming area. The hole is tangent to the reactor core, thus reducing the problem of gamma exposure to the film.

L/D ratios of 100, 250, and 500 are available and can be interchanged upon request. The flux at the film plane is $7 \times 10^6 \text{ n/cm}^2 \cdot \text{s}$. The film area is $0.43 \times 0.5 \text{ m}$, unusually large for radiographic purposes.

The system is fully automated with shutters and interlocks for personal radiation safety.

10.4.10 LPTR Fast Chemistry Experiments

The experiments to date have been gamma-ray spectroscopy of the short lived fission product isotopes of Sn and Sb. The fissions were produced by irradiating ^{235}U in the LPTR using the fast rabbit facility for target transport.

The Sn and Sb were distilled out as SnH_4 and SbH_3 , collected in HCl-Br_2 , and separated from each other by ion exchange techniques.

The rabbit transport and the liquid transfer systems were controlled by microcomputers.

10.5 Experimental Administrative Controls

Simple irradiations as well as operation of complex experiments are governed by the Reactor Supervisor. When an irradiation service is requested, technical information is furnished to the Reactor Supervisor, who is responsible for safety evaluations, reviews, and scheduling. Information regarding technical details, services available, regulations, and procedures can readily be obtained by calling the Reactor Supervisor. Detailed procedures are referenced in OSP-201.

It is the policy of the LPTN to accommodate any irradiation request as long as it can be done within the context of safety and meets the Laboratory policies regarding AEC contractual regulations.

A multidisciplined advisory group (Reactor Safety Committee, Sec. 13.4.2) is available on request to render advice on technical matters and safety problems regarding reactor operation.

CHAPTER 11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Source Terms

Radioactive wastes can arise in many ways:

1. Liquid waste - contaminated sink drains, floor drains, primary coolant water, resin regeneration discharges and liquid targets.
2. Gaseous waste - ^{41}Ar in natural air emitted via air exhaust system (historical maximum 37 Ci/day at full power and approximately zero for reactor scram periods).
3. Solid waste - small equipment, gloves, poly, spent targets, target containments, spent resins, etc. (Spent fuel elements, since they are returned for reprocessing, are not considered as solid waste and consequently are discussed in Sec. 9.1. 2.)

11.2 Liquid Waste Systems

Potentially contaminated liquid waste is collected in waste retention tanks within the reactor building for reactor systems or outside the laboratory building for laboratory building drains. This liquid waste is sampled, then transferred via tank trailer to the treatment plant at Building 514. The waste water is then processed and disposed of in accordance with laboratory regulations. Known highly contaminated liquid waste is collected in carboys and separately transferred to the treatment plant.

11.2.1 Design Objectives

A liquid waste system is designed to capture all potentially radioactive waste water that may be generated within the reactor containment building. This includes liquids from the sink, floor drains, and process systems such as the primary coolant system or primary purification system. The waste tanks are located in the equipment pit below ground level. Special valving and pumps are required to pump the waste water from the waste tanks to portable auxiliary outdoor tanks. No radioactive liquids can escape to the environment despite failure of any part of the system.

A second liquid waste system is designed to collect potentially radioactive liquids from certain of the laboratory sink drains. This system utilizes special pumps and valving to dispose of the liquid waste so that malfunction in the worst case would allow radioactive liquids to leak into the sump only.

11.2.2 System Description

11.2.2.1 Reactor Building—Potentially contaminated liquids in the containment building are funneled from process piping, floor drains, and sink drains into two 1500 gal hold tanks located in the equipment pit. Figure 11.2-1 illustrates the system

Fig. 11.2-1. Waste water system schematic.

layout. A manometer and high level alarm for each tank reads out in the control room. Waste water flows directly from four floor drains, magnet pit drain, sink drain, and demineralizer drain into the waste tanks. The primary system drain is described in detail in Sec. 5.1.3. Waste water from the primary or secondary system is first drained into the sump then pumped into the waste tanks. The sump pump has an automatic on switch. A sump level alarm reads out in the control room. The equipment pit is large enough and the sensitive equipment is high enough above the floor to be out of danger even if the entire primary system containing approximately 45 m³ (12,000 gal) of water were to drain into the pit.

11.2.2.2 Laboratory Building—Sink drains in rooms 1311, 1184, and 1323 collect in two outdoor 7.5 m³ (2000 gal) retention tanks located in a tank pit just north of the laboratory building. High liquid level triggers an audible alarm in room 1107 and a visual alarm at the top of the pit. Level indicators are provided at the tank pit. No alarm signals go to the reactor control room.

11.3 Gaseous Waste Systems

11.3.1 Design Objectives

The gaseous waste system (duct and stack) is designed to prevent the accumulation of radioactive gas (in particular activated Argon) in the containment building. It removes stagnant air masses from the various beam ports, pneumatic rabbit exhausts, thermal columns, through tube, and pool top. The design includes filtering and monitoring requirements to prevent hazardous releases to the environment. The duct monitors can trigger a building seal approximately 4-1/2 s before the radioactive gas reaches the stack (and hence environment). See Sec. 11.4.2. The emergency filter/scrubber system for educting radioactive gasses is discussed under *Containment Air Purification*, Sec. 6.2.3.

11.3.2 System Description

11.3.2.1 Duct System—From the mouth or vestibule of each of the beam tubes and through tubes, the inner face of each thermal column, and the pool top just below the deck plates, 2-in. piping passes downward through the concrete shielding to an 8-in. header underneath the reactor shield, Fig. 11.3-1. The 8-in. header passes into the equipment pit where it enters a large filter chamber consisting of a Cambridge absolute filter and a pre-filter. Both the pre-filter and absolute filter have manometers to indicate the corresponding pressure drop. The filters exhaust into a centrifugal exhaust blower rated at 0.252 m³/s (535 cfm) against a static head of 750 Pa. The filtered exhaust is forced from the equipment pit through 8 in. ducting to the cupola on top of the containment building where it is diluted with building air and expelled by the stack effluent system. The system is secured automatically by a building seal condition or manually with a switch in the control room. A negative pressure gauge in the control room indicates that the system is functioning.

11.3.2.2 Stack Effluent System—The stack system is located in the cupola atop the reactor dome. Air enters the cupola via the duct piping and a large grating on

Fig. 11.3-1. Gaseous waste duct system schematic.

the bottom side, Fig. 11.3-2. Duct air along with reactor room air is dumped from the stack to the atmosphere by an axial flow blower rated at $1 \text{ m}^3/\text{s}$. Air is continuously sampled just prior to passing through the blower. The waste gases, after being emitted from the top of the reactor dome, diffuse downwind away from the reactor building. The stack blower is secured automatically by a building seal or manually by a switch in the control room.

11.3.3 Operating Procedures

The gaseous waste system is operated full time whether the reactor is at power or shut down. The system is secured manually as desired, or automatically in case of a building seal condition. If the system is secured for more than brief periods while the reactor is at power, radiation levels can build up to undesirable levels within the containment.

The duct system filters are changed when a significant pressure drop (approximately 30%) builds up across them. Start-up check lists require routine weekly tests to insure that the radioactivity monitoring system is operational, and that it trips the building seal system at desired set points.

Fig. 11.3-2. Gaseous waste stack system.

Procedures for control of the gaseous waste monitoring system (Sec. 12.2.4) are discussed in Sec. 11.4.2. Limits on stack effluent gaseous radioactivity emissions are listed in Ch. 16.0. The limit is such that the diluted gaseous activity at the site boundary does not cause exposures in excess of those listed in AECM 0524.

11.3.4 Performance Tests

Refer to Sec. 11.4 for discussions of performance test.

11.3.5 Estimated Releases

Argon-41 is the principal isotope emitted from the containment structure to the environment via the gaseous waste system. Releases may vary depending on the configuration of experimental systems around the reactor. Typical daily emission rates for various operational schedules are shown in Table 11.3-1.

Table 11.3-1. Stack emissions (^{41}Ar) at 3 MW under various operating conditions.

Condition	Ci/hr ^a	Ci/day (2 shift)	Ci/day (3 shift)
Lead cooling blower on ^b	1.54	25	37
Same, av over a year ^c		15	22
Lead cooling blower off	1.13	18	27
Same, av over a year ^c		10.6	16

^aCuries/hr = Counts/min from stack monitor $\times 60 \times 2200 \text{ cfm} \times 0.028317 \text{ m}^3/\text{ft}^3 \times k$, where $k = 20 \text{ nCi} \cdot \text{min}/\text{m}^3 \cdot \text{count}$.

^bHistorical maximum.

^cThe averaging factor of 0.59 is based on experience. It allows historically 105 hr of operation out of a possible 168/wk, and 49 out of a possible 52 wk/yr.

11.3.6 Release Points

The single release point of routine gaseous wastes is the stack cupola atop the containment building, Fig. 11.3-2.

11.3.7 Dilution Factors

The specific dilution factors which will be used for determining yearly dose rates at various site boundaries are those listed in Table 11.3-2.*

The dilution factors associated with the population center at 1609 m (1 mi) to the west are shown in Table 11.3-3.

*Reproduced from Sec. 2.3.

Table 11.3-2. Average dilution factors χ/Q (s/m^3) at site boundaries.^a

	Annual	Dry Season	Wet Season
Critical azimuth ^b	NE	NE	E
Dilution factor	5×10^{-6}	8×10^{-6}	2×10^{-6}
Closest site boundary ^c	W	W	W
Dilution factor	1×10^{-6}	6×10^{-7}	2×10^{-6}

^aSector averaged per Sec. 2.3.5.

^bThe critical azimuth is the direction to the point on the site boundary having the largest χ/Q ; distance is one of the variables.

^cClosest site boundary is 440 m west.

Table 11.3-3. Dilution factors χ/Q (s/m^3) for population center - 1.6 km to west.^a

Dry season arith. av	1×10^{-7}
Dry season 5% ^b	0
Wet season arith. av	4×10^{-7}
Wet season 5%	3×10^{-6}

^aSector averaged per Sec. 2.3.5.

^bThe dilution factor that would be exceeded at most 5% of the time. During the dry season winds to the west are very rare ($<5\%$). All the exposure 1.6 km to the west occurs during these rare winds. Therefore at least 95% of the time the dilution factor is zero.

11.3.8 Estimated Doses

The dose to individuals downwind at any distance from the reactor during normal operation is due primarily to Argon-41 which is generated in and around the various experimental facilities by neutron activation of natural air. No other isotopes have routinely been measured in the effluent system or found in the environment, Sec. 11.6.2.

11.3.8.1 Short Term Close-in Dose Rates—Hourly dose rates immediately down wind from the reactor within the site boundary are calculated using those parameters listed below.

Stack Height

15.8 m (52 ft)

Output

Historical maximum - 37 Ci/day
(Table 11.3-1)

Atmospheric Conditions *

Day hours (0800 - 1900)

wind speed - 5 m/s average

Pasquill - D conditions

Night hours (1900 - 0800)

wind speed - 3 m/s average

Pasquill - E conditions

Dispersion Coefficients

Values of horizontal and vertical dispersion coefficients, σ_y and σ_z respectively, were established in Figs. 3-2 and 3-3, Workbook of Atmospheric Dispersion Estimates, D. Bruce Turner, U.S. Dept. of Health, Education, and Welfare, 1967.

Conversion Constant

$0.33 \frac{\text{rem}}{\text{S}}$ per Ci/m^3 of Ar-41

for whole body immersion dose.

The dose rates shown in Fig. 11.3-3 should be used only for estimating short term exposures for plume centerline locations. Total doses for exposure periods greater than approximately 8 hr should be determined from Sec. 11.3. 8.2 below. These short term dose rates agree favorably with Hazards Control Department calculations¹ which made use of earlier LPTR data.

11.3.8.2 Long Term Dose Rates—The dilution factors listed in Table 11.3-2, when used in conjunction with the ⁴¹Ar stack emissions shown in Table 11.3-1, produce total yearly whole-body doses shown in Tables 11.3-4 and 11.3-5.

Table 11.3-4. Yearly estimated dose at site boundary along critical azimuth, mrem.

⁴¹ Ar output averaged over year	Dry season dose (6 mo)	Wet season dose (6 mo)	Total (yr)
Historical maximum daily output, Pb cooling blower <u>on</u> ; 22 Ci at (3 shifts/day).	10.7	2.7	13.4
Present daily output, Pb cooling blower <u>off</u> ; 10.6 Ci at (2 shifts/day).	5.2	1.3	6.5

* Atmospheric parameters recommended by K. Peterson, Lawrence Livermore Laboratory, also used in Sec. 2.3.

Table 11.3-5. Yearly estimated dose rates at closest site boundary (440 m to west) mrem.

⁴¹ Ar Output averaged over year	Dry season dose (6 mo)	Wet season dose (6 mo)	Total (yr)
Historical maximum daily output, Pb cooling blower <u>on</u> ;			
22 Ci at (3 shifts/day).	0.8	2.7	~3.5
Present daily output, Pb cooling blower <u>off</u> ;			
10.6 Ci at (2 shifts/day).	0.4	1.3	~1.7

The annual whole-body dose at the west site boundary (440 m) is about 1/3 of that shown for the critical azimuth because the percentage of westbound winds is very small. This is evident from inspection of the dilution factors shown in Table 11.3-2 for the closest site boundary.

The whole-body doses at the nearest population center, one mile to the west, are shown in Table 11.3-6. Note that almost the entire dose occurs during the wet season. Arithmetic averages of dilution factors (Table 11.3-3) were used for these calculations as opposed to dilution factors that would be exceeded only 5% of the time.

Incidentally, the lead cooling blower has been idle since November, 1970. It has been left installed, in case it again becomes necessary to cool the thermal column lead nose piece.

Fig. 11.3-3. Immersion dose at surface on plume centerline, variation with distance.

11.3.8.3 Doses From Other Isotopes—No evidence has been found for isotopes such as tritium, iodine, xenon, and others that occur in the effluent from larger reactors. The amounts of radionuclides involved in the forage-cow-milk chain and the pasture-meat chain are insufficient to warrant further analysis.

Table 11.3-6. Estimated yearly dose at the nearest population center 1.6 km to the west, mrem.

Curie output averaged over year	Dry season dose (6 mo)	Wet season dose (6 mo)	Total (yr)
Historical maximum output, Pb cooling blower <u>on</u> ;			
22 Ci at (3 shifts/day).	0.13	0.53	0.66
Present status;			
10.6 Ci at (2 shifts/day).	0.06	0.26	0.32

11.4 Process and Effluent Radiological Monitoring Systems

11.4.1 Design Objectives

The monitoring system at the LPTR is designed to define the rate of radioactive material leaving the facility on a full-time basis. Stack monitor operation is a prerequisite for reactor operation. In the event of an accident situation in which levels outside the facility might approach MPC guidelines of AECM 0524, the ventilation system, Sec. 12.2.2, automatically shuts down to prevent effluent escape.

11.4.2 Continuous Monitoring

Gaseous waste is the only continuous effluent disposed of at the reactor. Consequently it is the only effluent with continuous monitoring.

Section 12.2.4 discusses in detail the overall airborne monitoring instrumentation and its functions. One sample (stack) is drawn from the building exhaust just ahead of the exhaust fan to determine release levels, Fig. 11.4-1. Another sample (duct) is taken from a duct drawing air from the experiment areas before dilution with building air to give an early shutdown warning in the event of a problem.

The effluent contains about $200 \mu\text{Ci}/\text{m}^3$ ^{41}Ar . No other radioactive material above background is normally found in the exhaust.

The stack monitor consists of a Nuclear Measurements Corporation (NMC) model AM-2A air monitor in series with a Tracerlab gas monitoring system unit. The NMC utilizes a Whatman 41 filter and a charcoal trap to capture particulate activity. This is viewed by an end window (5.6 mg/cm² thick) GM counter with 10.3% efficiency for beta and 0.5% efficiency for gamma radiation. The Tracerlab unit has a 2.83 dm³ sample volume shielded by 75 mm of lead, housing a 50 mm NaI(Tl) crystal. Sensitivity of the unit is 26.5 nCi/m³ · cpm, and the range is 0 to 200 000 cpm.

Fig. 11.4-1. Airborne radioactivity monitoring in ducts and stack, schematic.

The duct monitor is an LLL designed gamma-compensated gas monitor. It has a 1-dm³ ionization chamber with a sensitivity of 200 MCi/m³·A. The Keithley micro-microammeter readout ranges from 0 to 300 pA.

The emergency generator at the facility supplies backup power to the monitor units when normal commercial power fails.

The stack particulate monitor records at the unit and all monitor readouts are recorded automatically in the reactor control room. Each monitor has a meter readout at the unit and the stack gas and duct monitors have parallel meters in the control room.

At 40% of full scale, an adjustable optical meter relay on the duct monitor seals the building and initiates a warning light and bell on the control panel and sounds an evacuation horn in the dome. The same type of relay on the stack monitor seals the building and initiates a warning light in the control room at 30% of full scale, a release rate of $6 \times 10^{-4} \mu\text{Ci}/\text{cm}^3$ or $620 \mu\text{Ci}/\text{s}$. This would result in a maximum offsite concentration of $9.0 \times 10^{-8} \mu\text{Ci}/\text{cm}^3$, <50% of the AECM 0524 guide lines, Sec. 11.6.2.

The stack particulate monitor has an audible and light alarm on the unit. The stack gaseous and duct monitors trigger a light and bell in the control room on alarm and seal the reactor dome area. A Klaxon horn is sounded following the building seal, to signal evacuation of the dome area. In each alarm situation, the operator acknowledges the alarm, investigates the cause, and then scrams the reactor or attempts to reset the instrument from the control room, depending on the situation.

11.4.3 Sampling

Waste water is sampled when the waste tanks are emptied into the portable transfer tanks outside the Reactor building. This occurs about once a quarter and involves approximately 9.5 m^3 (2500 gal). The primary radioactivity is ^{24}Na , with 0.5-8 μCi gross activity normally present. The sample is analyzed for gross activity in a low background counting laboratory and the portable tank transferred to waste processing elsewhere on site. This waste is primarily composed of pool water runoff. The pool water is sampled once a week for gross activity and monitored continuously in the coolant loop for fission product activity in support programs other than effluent monitoring.

11.4.4 Calibration and Maintenance

The stack gas and duct monitor are calibrated against a standard ion chamber annually, using reactor effluent. All monitors are checked for proper operation weekly with a standard check source. Maintenance is performed by staff technicians as needed. Stack particulate samples are counted in a low background counting laboratory in which alpha and beta standards are run each day to check calibration.

The entire calibration procedure is reviewed periodically by the supporting Health Physicist.

11.5 Solid Waste System

Solid waste is generated in insufficient quantities to require a specific solid waste system at the reactor.

Dry waste (paper, glass, small equipment, gloves, etc.) is collected in 55 gallon processing drums and transferred to the Dry Waste Processing Facility at Building 612. These are closed systems without auxiliary systems support.

11.6 Off Site Radiological Monitoring Program

The Hazard Control Department carries on a continuing environmental surveillance program both within and beyond the laboratory boundary. Even though LLL places maximum emphasis on controlling the effluents at the source, it maintains an environmental surveillance program to ascertain the effectiveness of the control measures and to determine whether radiological changes in the environment are the result of Laboratory operations. This surveillance program utilizes complex analytical techniques capable of detecting the activity of the numerous radionuclides in the environment at natural background levels. At present the radionuclides of concern to LLL, because of their potential for escaping into the environment from Laboratory operations, include the transuranic elements, fission and neutron activation products, and tritium.

11.6.1 Expected Background

For several years, radiation levels at the site boundary due to ^{41}Ar stack emissions from the LPTR were undetectable. More recently instruments have been developed to measure extremely low radiation doses. The overall LLL off site results are submitted to the AEC on an annual basis.² The data is also contained in the Environmental Protection Agency publication, "Radiological Health Data and Reports."

The program includes the analysis of various types of environmental samples that have been collected within the Livermore and Amador Valleys, Site 300, and the San Joaquin Valley. Changes are expected from time to time as the specific operations carried out are changed and as knowledge in specific areas increases.

11.6.2 Critical Pathways

The only pathway for off-site radiation due to regular LPTR operations is via the ^{41}Ar gaseous activity ($<1 \text{ fCi/m}^3$ gross alpha and $<3.3 \text{ fCi/m}^3$ gross beta). The primary water system is cooled via a secondary coolant system that is regularly monitored for radioactivity. Because the secondary water is routinely found to be free of radioactivity, the critical pathway discussion will be limited to ^{41}Ar emissions.

Calculation of site boundary concentrations, and hence of dose rates, use the x/Q equations developed in Sec. 2,3,4 and .5. Q values and x/Q values are shown in Sec. 11,3,8. The following calculation, although more conservative than the assumptions used in Sec. 11,3,8,1, will illustrate the general method for calculating gaseous activity concentrations.

Assumptions:

- a) The eastern site boundary (1100 m) has the highest percentage of winds.
- b) Pasquill Type F atmospheric conditions exist; i.e. maximum stability.
- c) Annual average wind speed μ is 6 m/s (13 mph).
- d) Historical maximum stack emission rate Q is 37 Ci/day (430 μ Ci/s).
- e) Emission released at ground level, a conservative simplifying assumption.
- f) No building wake factors.

Then from "Meteorology and Atomic Energy, 1968" we obtain

$$x_{(\text{ground})} = \frac{Q}{\pi \sigma_y \sigma_z \mu},$$

where

$$\sigma_y = 50 \text{ m and } \sigma_z = 18 \text{ m.}$$

Substituting we find

$$x_{(\text{ground})} = \left(\frac{430 \mu\text{Ci/s}}{\pi \cdot 50 \text{ m} \cdot 18 \text{ m} \cdot 6 \text{ m/s}} \right) = 25.3 \text{ nCi/m}^3.$$

This value is <0.1% of the limit, 40 $\mu\text{Ci/m}^3$ (4×10^{-8} mCi/ml), as listed in AECM 0524. Even disregarding reductions due to reactor scram periods, atmospheric instabilities, higher wind speeds, and stack height, the ^{41}Ar gaseous emissions are clearly within acceptable limits.

11.6.3 Sampling Media, Locations and Frequency

11.6.3.1 Atmospheric Particulate Sampling—The air monitoring program consists of 15 continuously operating sampling stations situated within approximately 16 km of the Livermore Laboratory. The filters are removed weekly and are subjected to gross alpha, gross beta, and gamma spectral analysis. Some of these filters also undergo radiochemical analyses for ^{238}Pu , ^{239}Pu , ^{235}U , and ^{238}U .

11.6.3.2 Water Sampling—The water sampling program is designed to ascertain that radioactivity from the Laboratory's operations is not entering the domestic water supplies of the surrounding communities. The surface drain water from the Livermore Valley flows westward through the valley in various arroyos that enter Alameda Creek,

a source of water for the City of Fremont. During the rainy season, this drain water is sampled at various locations within the Livermore Valley. A significant portion of the water used by the cities of Livermore and Pleasanton, as well as the surrounding ranches, comes from local wells. Some of these sources of water are sampled on a monthly basis for gross alpha, gross beta and tritium determinations.

11.6.3.3 Sewer Effluent Sampling—The Laboratory's sewer effluent is processed by the Livermore Sewage Disposal Plant. As the effluent is discharged from the Laboratory, it is continuously sampled and daily analyses for gross alpha and gross beta activities are performed. ^{239}Pu , ^{90}Sr , ^{226}Ra , and ^{228}Ra , and mixed iodine analyses are performed on a monthly basis. In addition, the effluent is monitored continuously, both by a FIDLER detector employing a 13 cm (5-in.) diam $\times$ 3.2 mm (1/8-in.) thick NaI (Tl) crystal and also by a 50 $\times$ 50-mm (2 $\times$ 2-in.) NaI (Tl) crystal. The FIDLER is capable of detecting the low energy x- and gamma-rays emitted by the heavy elements while the higher energy radiations emitted by fission products and neutron activation products are detected by the 2 $\times$ 2-in. NaI crystal. When the effluent enters the Livermore Sewage Disposal Plant, the liquid and sludge are separated, and the sludge passes into one of two digesters where it is broken down by bacterial action. Methane gas is evolved and burned, and the remaining sludge is released to large sludge lagoons and/or drying beds and retained for subsequent possible use as a soil conditioner. Samples are collected weekly from the aeration tanks, each digester, and the effluent to the Arroyo. All of these samples undergo gross alpha, gross beta, and tritium determinations with some undergoing ^{239}Pu and ^{90}Sr analyses.

11.6.3.4 Soil Sampling—In order to measure the surface distribution of various radionuclides that have been deposited within the Livermore Valley as a result of global fallout from atmospheric weapons tests as well as from possible Laboratory effluent discharges, the Environmental Surveillance Program includes an extensive soil sampling effort. The samples are analyzed for a variety of radionuclides of plutonium, uranium, americium and fission products.

11.6.3.5 Vegetation and Milk Sampling—Vegetation samples are collected monthly from various locations within the Livermore Valley. These samples are subject to tritium and gamma spectral analyses. Milk samples are collected monthly from a dairy. These samples are condensed and analyzed for tritium and gamma emitters.

11.6.3.6 Background Radiation Measurements—LLL presently has approximately 50 fixed stations for measuring the radiation background in the near vicinity of the Laboratory by means of thermoluminescent dosimeters. The dosimeters are read every 3 months. From these results, we could detect any increases in radiation levels due to Laboratory operations.

11.6.3.7 Other Measurements—Gamma spectral measurements have been made in the Livermore Valley with a mobile in-site Ge(Li) gamma detector. This instrument may be employed to detect gamma emitting radionuclides deposited on the soil surface from global fallout or from Laboratory operations.

11.6.4 Analytical Sensitivity—(See Sec. 11.6.6 below)

11.6.5 Data Analysis and Presentation—(See Sec. 11.6.6 below)

11.6.6 Program Statistical Sensitivity

Various instruments and techniques used for off site monitoring, their sensitivities and data presentation are fully described in the annual report on environmental levels of radioactivity in the vicinity of the LLL.²

REFERENCES

1. R. J. Kloepping, ^{41}Ar Concentrations at Varying Distances from the LPTR, Lawrence Livermore Laboratory, Rept. UCID-16039 (1972).
2. P. H. Gudiksen, C. L. Lindeken, J. W. Meadows, and K. O. Hamby, Environmental Levels of Radioactivity in the Vicinity of the Lawrence Livermore Laboratory, Lawrence Livermore Laboratory, Ann. Rept. UCRL-51333 (1973).

CHAPTER 12.0 RADIATION PROTECTION

12.1 Shielding

12.1.1 Design Objectives

The principal purpose of the reactor shielding is to keep laboratory radiation levels within the safety limits shown in Table 12.1-1. A secondary objective is to prevent scattered radiations from affecting adjacent experiments during normal full power operations.

Table 12.1-1. Maximum radiation dose rates.

Area	Radiations	Hourly mrem	Quarterly rem	Yearly rem
General work areas	All	2.5		
Whole body ^a	Penetrating		3	5 ^b
Whole skin	β , γ , and x		10	30
Hands, feet ^c	β , γ , x , and n		25	75

^aIncludes dosage to head and trunk, bone marrow, gonads, or eye lenses.

^bMay be prorated if total accumulated dose $5(N-18)$ rem, where N is employee's age in years at next birthday.

^cIncludes forearms and ankles.

Exposures for anticipated operational occurrences are determined in detail by both the LLL Health and Safety Manual¹ and the LPTR procedures.² Compliance with LLL radiation limits meets the requirements of 10 CFR part 20 and AECM 0524.

12.1.2 Design Description

The biological shielding for the reactor consists of about 0.6 m of water and 1.8 m of heavy concrete* in the horizontal plane, and 6 m of water in the vertical direction. The concrete forms an octagonal structure 5.6 m across its flats, with an additional irradiation facility extending 3.1 m from the east face. Heavy concrete 1.8 m thick surrounds the lower 3.4 m of the reactor tank, tapering to 0.9 m at the 4.3 m level. From this point to the top of the tank (8 m elevation), 0.9 m of ordinary concrete 4.3 Mg/m³ provides a sufficient shield. The step in the shield at the 3.4 m elevation forms a convenient "balcony" providing access to several irradiation ports and beam shutter mechanisms. Thermal columns are shielded by 0.9-m thick heavy concrete doors lined on the inside with 6 mm ($\frac{1}{4}$ in.) boral. See Figs. 12.1-1, 2, and 3.

*Magnetite concrete 5.8 Mg/m³.

Fig. 12.1-1. East-West sectional view through centerline of reactor looking North.

● On floor working level of reactor room outside primary shield	0.3
● 1.5 m (5 ft) from those regions of core face where beam tubes exit or ducting traverses the shield	3.0
● Over equipment pit and fuel element storage area	3.0
● On irradiation port balcony and above present water surface	10.0
● With 0.6 m central graphite region removed from thermal column.	7.5

Fig. 12.1-3. Reactor equipment pit.

Two problems arise with the use of heavy concrete as the primary shield material; high-energy gammas such as the aluminum capture gamma (7.7 MeV) are not strongly attenuated by the concrete and radiation heating occurs in the concrete. Both problems are readily solved by surrounding the aluminum tank in the region of the core with a 64-mm lead thermal shield. The lead is bonded to the outer surface of the tank, thus permitting pool water to carry away radiation heating.

The reactor room has various arrangements of $0.6 \times 0.6 \times 1.2$ -m water-filled aluminum shielding tanks. Such shielding can be moved wherever it is needed.

A wall of $0.9 \times 1.5 \times 2.7$ m concrete blocks shields the control room from external radiation; airborne contamination is isolated in the reactor containment building, minimizing any chances for control room contamination.

The control room meets AEC General Design Criterion 19 for all types of accidents except during the initial phases of the case of complete core melting and fission product dispersal as discussed in Sec. 15.4., Design Base Accident (DBA). The quantities of direct line of sight radiation, scattered radiation, and radioactive emissions from the containment building suggest that the control room may be too near the reactor containment building to be used as the initial control point for the DBA. Experimental data discloses, however, that a loss of coolant accident (Sec. 15.2) would produce radiation levels in the control room less than 100 mrem/hr, a tolerable level for solving most problems. For those problems requiring a longer stay time the operation could be moved to the front office where there is additional communications equipment. Refer to Fig. 12.1-4 for control room layout and shielding. The location of the reactor with respect to the laboratory site, areas surrounding the Laboratory, and transportation routes is discussed in Sec. 2.1.2.2.

12.1.3 Source Terms (contained)

Radioactivity is produced directly in the core by fission and secondarily in the coolant, shielding, and structural components by activation. Radiation from the core is, by several orders of magnitude, the most intense source.

Activants in the coolant include ^{16}N (gamma, 7.4 s), ^{17}N (neutron, 4.1 s), and ^{19}O (gamma, 29.4 s). They occur in such small quantities and have such short half lives that they become negligible after a short delay in the coolant system (delay tank). The 6-7 MeV gamma radiation from ^{16}N , penetrating through the concrete deck above the equipment pit (Fig. 12.1-3), is less than $0.6 \mu\text{rem/hr}$. Even above the delay tank it is less than 1 mrem/hr. The ^{16}N activity⁴ leaving the core at 3 MW is about 85 Ci/m^3 .

Fig. 12.1-4. Control room shielding layout.

Another significant source of radiation is the ^{24}Na (gamma, 15 hr) activity associated with aluminum components and water impurities. The $^{27}\text{Al} (n, \alpha) ^{24}\text{Na}$ and the $^{23}\text{Na} (n, \gamma) ^{24}\text{Na}$ reactions account for this source. Although it is difficult to calculate the quantities of ^{24}Na activity in the primary coolant, measurements have been made on the radiation levels near the demineralizer beds and at various other locations within the equipment pit. Radiation levels for 3 MW, 3 shift/day and continuous demineralizer running are shown in Fig. 12.1-5 for various locations in the equipment pit. These radiation levels are considered as the maximum values that will ever be observed for that schedule of operation. The radiation levels are due chiefly to ^{24}Na for routine operations. Fission products present in the primary coolant, however, can increase radiation levels in the pool and equipment pit enough to significantly affect the stay time. The reactor operations schedule and the demineralizer operations schedule can significantly alter the radiation levels in the demineralizer due to ^{24}Na , consequently altering the background radiation within the equipment pit.

Fig. 12.1-5. Radiation in reactor equipment pit from ^{24}Na following startup. Assumes 3 MW continuous operation with demineralizer on line.

Reactors with high flow rates and stainless steel piping systems build up radioactive deposits of ^{60}Co -containing and other compounds. We encounter no such deposits at the LPTR.

12.1.4 Area Monitoring

Remote area monitoring systems (RAMS) detectors are placed around the reactor, in certain work spaces, and in the main hallways. Their function is to alert the occupants and operations staff to potentially hazardous radiation levels. Figure 12.1-6 shows typical locations of RAMS equipment.

Table 12.1-2. Continued.

Station No.	Location	Normal reading mrem/hr	Alarm set mrem/hr
<u>Shops or Special Work Areas</u>			
1a	Control room	0.1	2
2a	Electronic shop (2)	0.1 - 0.5	2
3a	Containment structure experiment areas (2)	Varies with location	5 above background

Pneumatic Rabbit Systems

A RAMS unit is placed near the receiving end of each tube. Such units may have a variety of placements and ranges as appropriate to the experimental situation.

Additional RAMS units are placed for detecting and monitoring abnormal accident conditions independent of the control room units. Readouts for these units are in the hall near the main doorway of the laboratory building (Bldg. 281). Table 12.1-3 shows locations and typical readouts of these units.

Table 12.1-3. RAMS units having readouts separate from the control room.

Station No.	Location	Normal reading mrem/hr	Alarm set mrem/hr
<u>Hall Monitors</u>			
1b	Hall, at main door	0	20
2b	Hall, middle	0	20
3b	Hall, near air lock	0	20
<u>Criticality Accident Monitor</u>			
1	Inside cupola	0	10^4
2	Top of foam generator	0	10^4

Table 12.1-4 lists the specifications for the above RAMS units.

Table 12.1-4. RAMS equipment specifications.

<u>Range:</u>			
Probe:	TA-6-10R	$10^{-2} - 10^4$	mrem/hr
	TA-6-100R	$10^{-1} - 10^5$	
Master meter:			
	TA-1-10R	$10^{-2} - 10^4$	mrem/hr
	TA-1-100R	$10^{-1} - 10^5$	
	TA-10R/100R	$10^{-2} - 10^5$	

Table 12.1-4. Continued.

Station indicators:			
	TA-3-100mR	$10^{-2} - 10^2$	mrem/hr
	TA-3-10R	$10^{-2} - 10^4$	
	TA-3-100R	$10^{-1} - 10^5$	
Plug-In meter and alarm light:			
	AX-2-10R	$10^{-2} - 10^4$	
	AX-2-100R	$10^{-1} - 10^5$	
Manufacturer:	Tracerlab		
Adjustment Over-All:	<ul style="list-style-type: none"> a) H. V. potentiometer, chassis mounted, not on panel. b) True calibration: Two sealed miniature potentiometers mounted on probe base. 		
Response Time:	A maximum of 5 s to reach 90% of new equilibrium reading (time constant: 1.8 s).		
Accuracy:	<ul style="list-style-type: none"> ±10%, over-all, typical. Items: <ul style="list-style-type: none"> a) Line voltage: <0.015% per 1% change in line voltage. b) Temperature: <0.11% per °C. c) Calibration drift: <0.2% per day. d) Photon energy: <±20% for 100 keV to 2.5 MeV. 		

12.1.4.2 Maintenance and Calibration — The units are serviced when the detectors fail to calibrate or fail to function.

The typical calibration program is as follows:

- a) Containment structure, hall monitors, and room monitors — Once every four months a crew from Hazards Control calibrates the RAMS using a special source. Every week the operators make a calibration check using other sources.
- b) Criticality accident monitor — Calibrated every 4 months same as the RAMS. Every week Hazards Control makes a calibration check using the internal source installed in the probe.

A typical calibration includes placing a calibrated source on the probe and observing the master meter. If it cannot be calibrated by the potentiometers on the probe base the system is repaired.

12.1.4.3 Recording Devices

- a) RAMS stations 1 thru 10 are continuously recorded on either north or south radiation recorders on the reactor console.

- b) RAMS room monitors and criticality accident monitors are checked periodically by the operator when the reactor is running and all radiation levels are noted in the daily log.
- c) RAMS hall monitors are not recorded in normal operation.

12.1.4.4 Typical System and Operator Actions – The alarm and most probable reaction for each is shown in Table 12.1-5. Certain combinations of alarms produce automatic reactor containment seal. These combinations are listed in Sec. 7.3.1.

Table 12.1-5. Alarms vs actions.

Unit	Type of alarm	Alarm location	Action
10 RAMS in containment structure	Bell and light	Control room annunciator panel. Read out meter on control panel.	Check problem. Reset. No system action.
1a, 2a, 3a, RAMS	Light and tone	Read out meter on control panel.	Check problem. Reset. No system action.
3 Hall monitors	Bell and light	Read out panel at west end of hallway.	Check problem. Reset. No system action.
2 Criticality accident monitors	Klaxon horns and light	Locations throughout the bldg. Read out panel at west end of hallway. Signal to firehouse.	Emergency services respond. Operators initiate disaster control procedures.
Pneumatic rabbit system RAMS	Light	Control room	Check problem. Reset.

12.1.4.5 Power Sources – The RAMS normally operates on standard building a-c power (Sec. 8.3) backed up by power from the emergency generator (Sec. 8.3.1.1.2). In addition the three hall monitors and the two criticality accident monitors can operate on the battery system (Sec. 8.3.2).

12.1.5 Operating Procedures

All areas in which radiation levels above 2.5 mrem/hr occur are off limits to all personnel. They are blocked off with either portable shielding or rope and clearly marked as to the nature of the hazard. Entrance into these areas is controlled by the operations and the hazards control staffs.

Each beam facility has a flashing light attached that warns personnel in the area when the shutter is open. For lower level beams we interlock entry into the beam area with a photo cell relay. Interrupting the light path will activate a positive alarm and close the shutter. For intense beams we use a gate that can only be unlocked when the beam shutter is closed.

Much of the shielding around the reactor is portable and is arranged so that continuous work areas receive no more than 100 mrem/wk for an average 40-hr work week. Radiation surveys are made weekly and high radiation areas are noted with signs and in some cases flashing lights. The movement of portable shielding is strictly controlled and is done only by the operations staff with the reactor normally shutdown. Shielding effectiveness is carefully checked during the subsequent reactor startup.

All operations personnel, experimenters and visitors are required to wear thermoluminescent dosimeter (TLD) badges to monitor their exposure, Sec. 12.3.3. The duct system (Sec. 11.3) is continually monitored by an ionization chamber and the effluent air leaving the building is monitored at the point of exhaust in the stack by the stack monitor, Sec. 12.2.4. The primary water is monitored with a fission product water monitor. The building is automatically sealed if either the stack or the duct monitors exceed a pre-set value, Sec. 11.4. The building will also be automatically sealed if readings for either the reactor top or primary coolant plus one other room location exceed the pre-set value as described in Sec. 7.3.1.

Radiation monitors (β, γ) producing a hallway entrance alarm are located along the hallway of the laboratory building with readouts at the front entrance. The criticality monitors located over the reactor tank alarm at the hallway entrance and the LLL firehouse if they detect a high radiation field. (Sec. 9.5.1.)

12.1.6 Estimates of Exposures

The peak whole body dose rate to personnel in the reactor containment building is approximately 1 mrem/hr gamma and 0.3 mrem/hr neutron. The dose rate in the control room is 0.15 mrem/hr and there is no measurable dose in the rest of the laboratory building or other locations outside the reactor building. The yearly man-rem exposures for reactor operation personnel are 0.73 rem, based on personnel dosimetry averaged over the last eight years and excluding the 1971 shutdown.

In the operating history of this reactor, no personnel exposures exceeding AEC radiation limits, (AECM 0524) 10 CFR 20 have occurred.

Figures 12.1-7 through 11 are radiation surveys inside the reactor containment building showing gamma and neutron levels for the shielding arrangement shown. Radiation levels at the site boundary due to line-of-site radiation from the reactor are effectively zero. Radiation levels at the site boundary produced by ^{41}Ar emitted by the stock system are presented in Sec. 11.3.8.

Fig. 12.1-7. Radiation survey at LPTR floor level. Reactor operating at 3 MW. Instruments: ionization chamber, G-M meter, and Rem meter (neutrons) all read at waist height. All readings in mrem/hr.

Fig. 12.1-8. Radiation survey around and on top of LPTR shield. All readings in mrem/hr.

Fig. 12.1-9. Gamma radiation survey of LPTR basement floor. All readings in mrem/hr.

Fig. 12.1-10. Gamma radiation survey of LPTR basement elevation. All readings in mrem/hr.

Fig. 12.1-11. Radiation survey — East face of LPTR. Gamma and neutron readings in mrem/hr.

12.2 Ventilation

12.2.1 Objectives

Ventilation systems in the containment building include the 1) air conditioning system, 2) emergency decontamination system, and the 3) filter/scrubber system. These systems individually or collectively:

- a) Recirculate air and control temperature for comfort (air conditioning system, Sec. 6.2.3).
- b) Control and remove gaseous radioactive waste, (duct and stack system, Sec. 11.3.2), keeping airborne radioactivity in general work areas below limits of AECM 0524.
- c) Scavenge particulate airborne radioactivity from containment air under sealed building conditions, preventing all but radionoble gases from escaping to the environment (scrubber/filter system, and emergency decontamination system, Sec. 6.2.3).

12.2.2 Design Description

12.2.2.1 Air Conditioning System – The air conditioning unit is capable of maintaining summer temperatures 11°C below ambient, with a maximum of 40% relative humidity at 40°C and supplies enough fresh air to completely change the air at breathing level (below 4.6 m) twice every hour. This equipment is located in room 1328 of the adjacent lab building, with ducts penetrating the reactor building near the air lock. Air, cooled in summer and heated in winter, is delivered by two fans ($3.8 \text{ m}^3/\text{s}$) through equally spaced registers in a duct running the circumference of the wall. Air is warmed by hot water coil sections installed in the ducts within room 1328. Recirculating air leaves the building through a duct leading to the inlet of the air conditioning system where it is replenished with about 30% fresh air. In addition, building air is exhausted via the stack system on top of the reactor building by means of a blower rated at $1.0 \text{ m}^3/\text{s}$ (2200 cfm).

The stack system, the main release point for containment building gasses, discharges at a nominal rate of $1 \text{ m}^3/\text{s}$. This quantity includes duct system gasses at a rate of less than $0.2 \text{ m}^3/\text{s}$ and general containment structure air at a rate of about $0.8 \text{ m}^3/\text{s}$.

12.2.2.2 Decontamination System – A normal air conditioning system of ducts and dampers permits air to be recirculated within the sealed building when one flapper valve is closed, Fig. 12.2-1. The air flow is out through the air conditioning duct on the north wall of the building, through a bank of four Cambridge absolute filters just outside the reactor air lock and in through the duct on the south wall. Thus cleanup air sweeps air through the building from south to north. This system will filter $2.6 \text{ m}^3/\text{s}$,

Fig. 12.2-1. Decontamination system and filter/scrubber systems. Black arrows represent emergency decontamination air flow, South to North.

or the equivalent of the entire building volume in 540 s. The filters are 99.5% efficient for particles larger than $0.3 \mu\text{m}$. The equipment for this recirculation system is located just outside the air lock where the filter is housed in a unit specially designed to facilitate replacement of contaminated filters. The cleanup operation may be performed continuously until it reduces the activity to the desired level.

12.2.2.3 Air Cleaning System (Scrubber/Filter System) – In the event of a building seal, the three butterfly valves close, one of the air lock door gaskets inflates to seal

the building and the divider flapper valve closes (the emergency escape and truck door gaskets are normally inflated at all times). If the building overpressure exceeds 500 Pa (2 in. H₂O) positive pressure, both emergency filter-scrubber systems start to exhaust air from the building. (The system can also be started manually.) This air, after passing through Cambridge absolute filters, activated charcoal filters, and KOH scrubbers, is exhausted through 12-m stacks into the atmosphere essentially free of particulate fission products and radioiodines (see Fig. 12.2-1). The charcoal filters act as additional filters but specifically remove any methyl iodide formed. Measurements show them to be 99.9% efficient (dry weather), Sec. 6.2.3.2.

The use of two separate systems, each with its own filters, assures the continuous availability of the air cleaning system. Each unit alone is capable of maintaining the building pressure below 500 Pa during the most severe diurnal temperature changes, Sec. 3.8.2.2. Each system is also connected to the emergency power system.

12.2.3 Source Terms (airborne radioactivity)

Neutron activation of natural air generates ⁴¹Ar ($t_{1/2} = 110$ min) in amounts shown in Sec. 11.3.5. Should the containment structure become sealed while running at full power, ⁴¹Ar radiation levels would rise slowly inside the containment until shutdown or resumption of stack system flow.

12.2.4 Airborne Radioactivity Monitoring

Airborne radioactivity monitoring is done from five fixed locations as shown in Fig. 12.1-6 and from portable units. The five fixed locations are:

- Duct system (gaseous)
- Stack gas system (gaseous)
- Balcony NMC* (particulate)
- Fast neutron irradiation facility (FNIF) NMC (particulate)
- Stack NMC (particulate)

12.2.4.1 Monitoring System Specifications – The above air monitoring systems consist of three basic types of unit, the LLL duct monitor, the Tracerlab gas monitor and the NMC units. Specifications for the LLL duct monitor and the Tracerlab gas monitor are shown in Table 12.2-1. Specifications for the radioactive particulate monitors (NMCs) are shown in Table 12.2-2.

*NMC – Nuclear Measurements Corp.

Table 12.2-1. Specifications - gaseous monitors.

<u>LLL Gamma Compensated Gas Monitor, Duct</u>	
Sample volume	1 dm ³
Air flow	0.5 dm ³ /s (2 cfm)
Detector	Ionization chamber
Sensitivity (⁴¹ Ar)	255 MCI/m ³ ·A
Range	Normal 0 - 30 pA adjustable Keithley picoammeter
Display	L & N Speedomax recorder and 50 μA meter
Warning system	Adjustable optical meter relay seals building and initiates warning light at 40% of full scale. Normal reading is 12% full scale.
<u>Tracerlab Gas Monitoring System, Stack</u>	
Sample volume	2.83 dm ³
Shielding	75 mm (3 in.) lead
Air flow	2 - 3 dm ³ /s (4 - 6 cfm)
Detector	5.08 cm NaI (TI) crystal
Sensitivity (⁴¹ Ar)	20 pCi/m ³ ·(cpm)
Range	0-200,000 cpm
Display	L & N Speedomax recorder and 50 μA meter
Warning system	Adjustable optical meter relay seals building and initiates warning light at 30% full scale. Normal reading is 6% full scale.

Table 12.2-2. Specifications - airborne particulate monitoring system.

NMC Air Monitor, Model AM2A, Nuclear Measurements Corp.

Air flow	2 dm ³ /s (4 cfm)
Collector	Filter paper
Area	20 cm ²
Detector	End-window GM counter
Window diam	48 mm (1-7/8 in.)
Window thickness	56 g/m ² (5.6 mg/cm ²)
Efficiency	
α	0%
β	10.3%
γ	0.5%
Operating voltage	850 V

Sensitivity: Since this monitor accumulates activity on a filter paper, its readout must be interpreted in terms of the difference in counting rate at two different times. Its sensitivity therefore relates to the change in counting rate with time.

Table 12.2-2. Continued.

The overall sensitivity of a fixed filter air monitor, S, is the increase in counting rate per hour of operation. This in turn depends on the counting efficiency, the air flow rate, and the half lives of the activities collected, i.e.

$$S = \frac{C_2 - C_1}{t_2 - t_1} = 3600 t_{\frac{1}{2}} YRA$$

where: $t_{\frac{1}{2}}$ is half life in s
Y is counting efficiency in %
R is air flow rate in m³/s
A is the activity in Ci/m³.

12.2.4.2 Calibration of Airborne Monitors — The FNIF-NMC is checked before each use. The other airborne monitors are checked daily by the reactor staff.

Calibration of each of the airborne monitors is done under the cognizance of the Hazards Control Department Health Physicist as follows:

- a) Stack (Tracerlab) — A standard ion chamber is calibrated against standard ⁴¹Ar gas in a laboratory elsewhere on site.
- b) Duct (LLL Design) — γ spectrum analyses of gas samples have shown the only normal radioactive effluent to be ⁴¹Ar. A standard ionization chamber is used once a year in series with the normal monitoring instrument to calibrate.
- c) Balcony, FNIF, and Stack NMCs — Operational internal electronic calibration.

12.2.4.3 Recording of Airborne Sampling System Outputs — Each of the monitoring systems produces a signal on the north radiation multipoint recorder in the control room in direct proportion to the meter movement on the actual instrument. The multipoint recorder samples each signal in turn, completing the cycle in about 45 s. Separate scales on the recorder provide for direct readout of each instrument.

In addition the duct and stack gas monitors each read out on an A.P.I. meter with a settable relay. This relay generates the alarm signal for these monitors.

12.2.4.4 Set Points — Set points are conservatively established to ensure that the monitors will alarm and alert the operators to any abnormal situation. Typical set points are as shown in Table 12.2-3.

Table 12.2-3. Typical set points for airborne monitors.

Unit	Usual base activity		Alarm set	
	% of full scale	cpm	% of full scale	cpm
a) Stack				
1) Gas	6		30	
2) Particulate		2000		5000
b) Duct	13		40	
c) FNIF		2000		5000
d) Balcony		2000		5000

12.2.4.5 Alarm vs Action – Each airborne radiation monitor alarm (bell and light) calls forth some automatic or operator action. An alarm on one of the NMC particulate monitors, since it represents a local concentration, requires only that the operator determine the cause of the alarm, take corrective action, and reset the alarm. An alarm on one of the gas monitors (stack or duct), since it might indicate a more general malfunction, automatically seals the containment building, i.e. inflates the air lock gaskets, stops the stack blower, and closes all the butterfly valves.

12.2.4.6 Maintenance and Calibration Schedules – There is no regular maintenance program, apart from regular calibration. The units are serviced whenever they malfunction or fail to calibrate. Section 12.2.4.2 briefly describes the calibration technique. The typical calibration schedule follows.

a) Stack

Gas – The stack gas monitor is calibrated annually. Once a week a channel test is performed, measuring the response to a known check source.

Particulate – The stack NMC is calibrated once a week, using a built-in source at response settings of about 4000 and 20 000 cpm.

b) Duct – The duct monitor is calibrated annually. Once a week a source is used to perform a calibration check.

c) FNIF & Balcony NMC – Same as for stack NMC.

All air sampling data records are saved in permanent storage.

12.2.5 Operating Procedures

During operation, certain gases will be formed in the cooling water. The concentration of these, notably ^{16}N , and the products of water dissociation, hydrogen and oxygen, will be very small. These gases may diffuse to the surface; the majority are carried down out of the core with the primary flow. If they do rise to the surface they will be

continuously drawn off from the space between the water and enclosing top structure by air exhaust system.

The exhaust facility continuously draws air from all of the experimental facilities and the reactor top, preventing accumulation of the ^{41}Ar produced by neutron absorption in the ^{40}Ar found in normal air. Since each beam port has an air duct near each reactor face leading into the exhaust system, withdrawal of a beam plug will not release significant quantities of irradiated air. Similarly, experiments (e.g. scattering experiments at a beam facility) resulting in the neutron irradiation of large quantities of air can be separately enclosed, with the air from the enclosure exhausted into the air exhaust system. To further minimize the inhalation of airborne radioactivity, air monitors about the reactor room (especially those on the air exhaust system) are set to alarm at very low gas concentration.

Laboratory procedure requires bioassay sampling whenever there is a possibility of radioactive airborne uptake, Sec. 12.3.3.

12.2.6 Estimates of Inhalation Doses

Since the start up of the LPTR there have been no significant inhalation doses to the operating personnel of the reactor. This is true because ^{41}Ar , the only airborne activity prevalent at the reactor, is continuously swept from all irradiation facilities by the duct system and exhausted to the outside via the stack system, Sec. 11.3. In the event of other airborne activity wholebody counting and bioassay would be used for estimating inhalation doses.

12.3 Health Physics Program

12.3.1 Program Objective

Although programmatic supervisors have the direct responsibility of providing safe equipment and implementing safe procedures, the Hazards Control Department helps to guide and assist employees in preventing injury from ionizing radiation to themselves, to the general public, and to the environment. Every effort will be made to keep radiation exposure of personnel at an absolute minimum. Only in emergencies will authorization be given to exceed the permissible levels set forth in the U.S. AECM 0524, "Permissible Levels of Radiation Exposure."

The Hazards Control Department provides the reactor facility with technicians who have training in each of the safety disciplines. In addition Health Physics related problems requiring detailed analysis in a particular field are referred to professionals who support and review the Laboratory-wide effort in that area.

12.3.2 Facilities and Equipment

A Remote Area Monitoring System (RAMS) has been included in the facility to monitor the gamma radiation level. This system, Fig. 12.1-5 includes numerous Tracerlab remote area monitors, located at strategic points throughout the facility, with associated readout meters in the reactor control room or the building entrance lobby. Two additional Tracerlab units are provided to trip a criticality accident alarm system. Normal and alarm values for each unit are listed in Sec. 12.1.4.1. Airborne radioactivity is monitored using that equipment discussed in Sec. 12.2.4. Typical normal and alarm values for this equipment are listed in Table 12.2-3. Additional air monitoring is provided by an NMC constant air monitor at the reactor balcony, and one extra unit for special job requirements.

Three nuclear accident dosimeter (NAD) systems at fixed locations in the facility provide neutron spectrum and high-range dose information in the event of a criticality accident. Table 12.3-1 shows the activation material contained in each dosimeter.

Table 12.3-1. Neutron activation materials in the nuclear accident dosimeter (NAD) system.

Material	Quantity g	Activation threshold
^{197}Au (gold)	0.25	Thermal
$^{197}\text{Au} + \text{Cd}$ (gold covered with cadmium)	0.25	0.4 eV
$^{239}\text{Pu} + ^{10}\text{B}$ (plutonium covered with boron)	1.0	10 keV
^{237}Np (neptunium)	0.01	0.6 MeV
^{238}U (uranium)	5.0	1.5 MeV
^{32}S (sulphur)	} 16-21	3.0 MeV
^{34}S (sulphur)		6.0 MeV

LLL maintains a team and equipment to evaluate the activation foils in the NAD system. A high-range chemical dosimeter package provides a measure of the gamma dose.

In addition to the fixed detectors in the above systems, the Hazards Control organization provides portable monitoring equipment including:

- Three high-range ionization chambers
- Three low-range ionization chambers
- Three G-M beta-gamma meters
- Three alpha meters

- Two LLL neutron counters (rem meter)
- Continuous air monitors as required.

Other systems include a fission product water monitor, and two hand and foot counters to aid in contamination control. Each portable radiation meter is checked at least once a week and at each use; malfunctioning or doubtful instruments are repaired and calibrated. Fixed instrumentation is checked at the start of each week and calibrated routinely.

The Laboratory maintains an active effort to develop and improve monitoring instrumentation.

12.3.3 Personnel Dosimetry

A record of the occupational radiation dose of each LLL employee is obtained by means of a continuous personnel monitoring program. Non-LLL employees in areas under direct laboratory control are monitored in the same manner as LLL employees. Calibration and development facilities are maintained by the laboratory in the Hazards Control Department.

12.3.3.1 External Radiation Monitoring - Personnel exposure to external beta rays, gamma rays, x rays, and neutrons are monitored with thermoluminescent dosimeters (TLD). Supplementary systems including finger dosimeters are used whenever necessary. The normal frequency for changing the TLD is once a month for reactor operations personnel. The minimum detectable dose is 10 mrem with the dosimeter system presently used. Each member of the reactor operations staff also wears a nuclear criticality accident dosimeter.

Each exposure that exceeds an average of 100 mrem/wk is reported to the employee's supervisor. Each exposure that exceeds U. S. AECM 0524 is reported to the supervisor and to the employee. The record of the employee's occupational radiation dose will be provided to him on request. Reports of accumulated dose are normally issued to the employee at each badge change. Each employee is responsible for protecting his dosimeter from damage, loss, and non-occupational exposure.

12.3.3.2 Internal Radiation Dosimetry - Employees who work frequently with unencapsulated radioactive material are monitored to determine possible internal deposition of radioactive material. The frequency and method are determined by the Radiation Safety Group. In all cases of known exposure to internal radiation by ingestion, inhalation, or puncture wound, the absorbed dose is to be estimated by the most appropriate methods available. Elimination rates are to be followed as long as they are relevant to the dose determination. Whole-body-counting equipment and bioassay

laboratory facilities are maintained by the Laboratory to support this effort. The reactor operations staff is checked at least annually.

REFERENCES

1. Lawrence Livermore Laboratory Health and Safety Manual, Lawrence Livermore Laboratory, Rept. M-010 Rev. 2-28-73 (1973) Sec. 33.
2. Building 281 Operational Safety Procedure, Lawrence Livermore Laboratory (1973).
3. J. L. Mitchell, Jr., Shielding Calculations for the LPTR, Foster Wheeler Corp. N.Y. (1956).
4. J. W. Bolstad, G. E. Cummings, J. L. Eagan, and H. F. Finn, Safety Analysis Report for the Livermore Pool Type Reactor, Lawrence Livermore Laboratory, Rept. UCRL-50147 (1966).

CHAPTER 13.0 CONDUCT OF OPERATIONS

13.1 Organizational Structure

13.1.1 Laboratory Organization

The Lawrence Livermore Laboratory consists of several major divisions, each reporting to an appropriate Associate Director. The Associate Directors report to the Laboratory Director. The LPTR is administered by the Chemistry Division. The organization chart, Fig. 13.1-1, describes the channels of responsibility for the LPTR. One channel of responsibility includes audits, reviews, and area surveillance. The other channel

Fig. 13.1-1 Laboratory organization chart.

of responsibility covers programmatic interest. Various support groups within the Laboratory assign personnel to reactor operations as necessary. These individuals report to the Reactor Supervisor on all technical matters.

13.1.2 Operating Organization

At least one reactor shift supervisor and one reactor operator must be on duty, with one of them in attendance at the console, whenever the reactor is operated.

13.1.2.1 The Radiochemistry Division Leader — This person is responsible to the LLL Director through the Head of the Chemistry Department and the Associate Director for Chemistry for the overall operational and experimental program related to the LPTR.

13.1.2.2 The Physics Section Leader — The Physics Section Leader is responsible to the Radiochemistry Division Leader for general reactor operations and experiments at the LPTR. In the absence of the Radiochemistry Division Leader, the Physics Section Leader may act in his behalf.

13.1.2.3 The Reactor Supervisor — This individual is directly responsible to the Physics Section Leader for the safe and efficient operation of the entire reactor facility. It is his responsibility to insure that reactor operation stays within the envelope set forth in this Safety Analysis Report. In the absence of the Reactor Supervisor, the Assistant Reactor Supervisor may act in his behalf.

13.1.2.4 The Shift Supervisor — Each Shift Supervisor is directly responsible for the operation and safety of the reactor facility on his assigned shift. Shift Supervisors also assist the Reactor Supervisor in matters such as scheduling of irradiations, burn-up calculations, and calculation of core change effects. In addition, they advise and assist experimenters in the use of the reactor facility.

13.1.2.5 The Supervisory Reactor Operator — This operator is responsible for certain specialized functions concerning operation of the reactor. In particular, the supervisory operator handles the administrative supervision of reactor operators regarding work schedules and nontechnical scheduling of reactor operations. He is also responsible for the proper maintenance of all control room records, check lists, and logs. The supervisory operator functions as a reactor operator when required.

13.1.2.6 The Reactor Operators — The specific function of these individuals is to operate the reactor console controls and perform those daily and weekly checks required for safe reactor operations. They also perform other duties, such as sample insertion and withdrawal, minor shielding changes, routine electronic maintenance, and routine mechanical maintenance.

13.1.2.7 The Mechanical Engineer — The Mechanical Engineer advises the reactor staff on matters pertaining to the mechanical systems used in the reactor. He is responsible for the design, procurement, and fabrication of mechanical equipment and performs the required maintenance functions to keep the facility in proper operating condition. He is responsible for the quality assurance program for all mechanical systems.

13.1.2.8 The Electronic Engineer — This engineer advises the Reactor Staff in matters pertaining to the electronics system of the reactor facility. He is responsible for the design, procurement, and fabrication of electronic equipment and performs the required functions to maintain the facility in proper operating condition. He is responsible for the quality assurance program for all electronic systems.

13.1.2.9 Hazards Control — The Hazards Control Department is responsible to the LLL Director through the Associate Director for Support for all LPTR health and safety matters. The Hazards Control Department is responsible for detecting and evaluating all industrial and radiological hazards associated with reactor operations and communicating these observations and results to the Reactor Supervisor. Hazards Control personnel also perform the function of periodically surveying the facility for radiation and contamination levels and advise the operations staff on safety matters pertaining to the work environment.

13.1.3 Qualification Requirements

13.1.3.1 Reactor Shift Supervisors — These supervisors are required to have had college level work in nuclear or reactor engineering courses and to have passed qualification studies involving undergraduate college level engineering problems. They are required to be knowledgeable of radiological, industrial and reactor safety practices. The potential Shift Supervisor is also required to become proficient with the reactor controls and safety systems and to pass operational exams. A typical training period is in excess of two months. The written examination covering all aspects of reactor operations and safety is given by the reactor staff and reviewed by an independent examiner. The operational examination is given by an independent examiner with a qualified Shift Supervisor or Operator in attendance at the console.

The physical condition and general health of each Reactor Supervisor shall be such that they can perform their assigned job tasks without undue risk to health and safety. Each Reactor Supervisor will have a physical examination at least bi-annually.

13.1.3.2 Reactor Operators — The nuclear and reactor theory study schedule required of Reactor Operators is similar to that for Shift Supervisors except that the math and physics used is at the high school rather than college undergraduate level. The

training program is again of several months duration and includes both operating practice on the reactor console as an "operator in training" and seminars or study sessions covering reactor fundamentals, radiological safety and industrial safety.

The physical condition and general health of each Reactor Operator shall be such that they can perform their assigned job tasks without undue risk to health and safety. Each Reactor Operator will have a physical examination at least bi-annually.

Written and operational examinations are given and reviewed in the same manner as for Shift Supervisors.

13.1.3.3 The Support Staff — The engineers, technicians, and safety personnel of the Support Staff meet laboratory qualification requirements in their specific area of assignment. The resident support staff must be acquainted with the procedures and hazards associated with the reactor. Nonresident support personnel are required to demonstrate a strong safety attitude and to be proficient in the use of the reactor evacuation systems.

13.2 Training Program

13.2.1 Program Description

The training program is conducted by the supervisory staff, and takes advantage when possible of the laboratory's instructional television programs and various on-the-job educational programs available from other departments of the laboratory.

Candidates for positions as reactor operators should be high school graduates, and are expected to have a basic understanding of high school level algebra and physics. The detailed training for each operator will vary depending upon the individual educational background and his work experience and previous training. The overall program has as its objectives the following goals and areas of coverage:

- (1) Review of high school level mathematics and physics;
- (2) An introduction to basic nuclear physics, reactor physics, and reactor kinetics;
- (3) Sufficient training to obtain a working knowledge of and experience with radiation safety and shielding principles;
- (4) Experience with the use of ancillary equipment such as handling tools, cranes, sample preparation, shipping casks, etc. . . ,
- (5) Complete familiarization with local operational policies, procedures, and safety limits,
- (6) Sufficient startups and checkouts to insure both ability to perform and understanding of the principles of operations.

The training program for shift supervisor candidates follows much the same procedure as for an operator, but with increased depth of coverage. Since all shift supervisor candidates are expected to have the equivalent of a masters degree in physics or nuclear engineering, primary emphasis in their training is to relate their academic training and previous work experience to LPTR operations. A significant portion of the shift supervisor's training program is to emphasize reactor safety and to develop an attitude directed toward safe operating practices.

The programs outlined above consist of both "textbook studies" and on-the-job training, lasting typically 3 to 6 months.

Candidate testing consists of a written exam prepared by the reactor supervisor in coordination with an independent examiner appointed by the Director's Office. Following successful completion of the written exam, the candidate is given an oral and operational examination by the examiner, on any material that the examiner deems appropriate.

After completion of testing, the candidate's qualifications are reviewed and certification is issued jointly by the Reactor Supervisor, Physics Section Leader, and the Director's Office.

Records of examinations and certification are maintained in a personal training file for each operator and supervisor. A current list of persons certified to operate and/or supervise operation is posted in the reactor control room at all times.

13.2.2 Retraining Program

The retraining program is designed to maintain a level of familiarization commensurate with safe operating practices. The retraining program is primarily an "in-house" program. It is administered by the reactor supervisor and conducted by the staff.

The retraining program is divided into two categories: (1) those areas requiring annual recertification, and (2) those areas which require biennial recertification. The areas in each of these categories are listed in Table 13.2-1. The retraining for operators and supervisors is carried on concurrently, and only the depth of coverage is different. The times are approximate and can be modified to accommodate special needs or unusual problems that may arise.

When each area of training is completed an examination is given and grades and/or examination papers are made a part of the individual's training folder and maintained on file. Yearly recertification notices are prepared and filed.

13.2.3 Replacement Training

The replacement training program is the same as described in paragraph 13.2.1 above.

Table 13.2-1. Subjects included in the LLL retraining program.

I. Annual Subjects

- A. LLL Disaster Control Plan
- B. LPTR Disaster Control Plan
- C. LPTR Operations Manual (OSP 281)
- D. LPTR Operating Limits
- E. LPTR Safety Analysis Report
- F. LPTR Emergency Systems (maintenance, operation, and testing)
- G. Reactor Console (proper and safe manipulation of controls, operation and maintenance procedures, checklists and instruments)
- H. Auxiliary Equipment (handling tools, scrubber, demineralizer, etc. . .)

II. Biennial Subjects

- A. Basic Math and Physics
 - B. Nuclear Physics
 - C. Reactor Physics and Kinetics
 - D. Instrumentation and Control Systems
 - E. Radiological Health and Safety
 - F. LPTR Facilities
-

13.2.4 Records

All records pertaining to the qualification, experience, training, and retraining are maintained in an individual personnel folder for each member of the operations staff.

There is no current program, other than testing, to determine the effectiveness of the training program.

13.3 Emergency Planning

13.3.1 General

Emergency plans exist at the local level of reactor operations and at the laboratory-wide level. Description of functional details of both the LLL Disaster Control Plan¹ and the LPTR Disaster Control Plan² are omitted from this chapter because they are subject to constant surveillance and updating. However, Sections 13.3.2 and 13.3.3 provide an overview of these control plans.

13.3.2 LLL Disaster Control Plan

The LLL Disaster Control Plan documents the approach toward controlling major and minor accidents. Basic authorization for this plan is set forth in Section 3.00, Accident Control, of the LLL Health and Safety Manual.³

This plan, which has been approved by the California Disaster Office, establishes a Disaster Control Organization, Fig. 13.3-1. All members of the Disaster Control Organization are knowledgeable disaster workers and are given the authority set forth in the California Disaster Act of 1943, as amended, and Chapter 6 of the Alameda County Administrative Code.

Disasters are serious accidents that cause injuries or extensive damage and threaten to cause additional injuries and damage. Generally, accidents that can cause serious off-site effects and create public relations problems with the community are classified as disasters. The entire Disaster Control Organization is available during disasters to control the extent of injuries and property damage.

The LLL Disaster Control Plan outlines the organization, the actions and responsibilities of groups and key individuals. In addition, the emergency communications system, the equipment available during emergencies, and the fallout shelter plan are described briefly. More detailed descriptions of the communications systems, equipment available for use during emergencies, and detailed emergency plans are contained in a number of supplements to the plan. These supplements, when combined with the plan, comprise the total Disaster Control Plan.

The Disaster Control Organization as represented in Fig. 13.3-1 is divided into an Administrative Group and a Field Group. The Field Group is responsible for performing the control operations necessary to alleviate the effects of a disaster. The Administrative Group develops policy guidelines governing operations of the Disaster Control Organization. In the event of a disaster, the Administrative Group also provides overall control and administrative support and notifies concerned individuals and off-site agencies including the press.

13.3.3 LPTR Disaster Control Plan

The actions of laboratory personnel in controlling an accident involving Building 281 are outlined in Fig. 13.3-2. This figure summarizes the overall accident control activities as described in the LPTR Disaster Control Plan. Specific actions shown in Fig. 13.3-2 are not necessarily taken in the order given.

Fig. 13.3-1 Disaster Control organization chart.

Fig. 13.3-2 Plan of operation in case of accident.

Certain general features in the LPTR accident recovery operations are made part of the LLL Disaster Control Plan. For example the Emergency Dispatcher position appears in both Figs. 13.3-1 and 13.3-2. The entire LPTR Disaster Control Organization, Fig. 13.3-3, functions under the control of the Emergency Control Coordinator.

Fig. 13.3-3 Disaster Control organization for LPTR.

13.4 Review and Audit

13.4.1 Audit Committee

An annual Audit committee⁴ is appointed by the Director's Office to review all aspects of reactor operations and report its findings to the Director's Office.

classified according to their level of potential hazard and complexity as outlined in Table 13.5-1.

Table 13.5-1 Classification of Operation and Maintenance Procedures

A. Class I

1. Kind of operation
 - a. Routine startup or operation of the reactor with the loading and configuration unchanged from previous runs.
 - b. Startup after a minor scram.^a
 - c. Loading of samples in an S-tube, rabbit, or reflector position, or loading of samples in a core facility with the reactor shut down.
 - d. Maintenance performed by operators or technicians that involves no major changes to the reactor control system or console.
2. Administrative controls
 - a. Requires the prior approval of the Reactor Supervisor or the Shift Supervisor on duty.
 - b. Requires the presence of Reactor Supervisor or Shift Supervisor in the Building 281 area (within immediate communication).

B. Class II

1. Kind of operation
 - a. Operation of the reactor after a minor core change.
 - b. Startup after a major scram.^b
 - c. Maintenance operations such as changing magnets or replacing clutch switches.
 - d. Minor shielding changes and loading or unloading of collimators or beam tube plugs.
 - e. Removal of large sources from the reactor tank requiring the use of a coffin.
 - f. Routine fuel cycles.
2. Administrative controls
 - a. Requires the prior approval of the Reactor Supervisor.
 - b. The work is to be performed under the direct supervision of either the Reactor Supervisor or a Shift Supervisor.

C. Class III

1. Kind of operation
 - a. Nonroutine operation or maintenance involving a major change in the core configuration or major shielding change.
 - b. Startup of the reactor after an extended shutdown period.
 - c. Spent fuel shipments.
2. Administrative controls
 - a. Requires prior approval of Reactor Supervisor.
 - b. The work is to be performed under the direct supervision of the Reactor Supervisor.

D. Class IV

1. Kind of operation
 - a. Special tests or major operations of an unusual or potentially hazardous nature as determined by the Reactor Supervisor or Radiochemistry Division Leader.
2. Administrative controls
 - a. Requires the prior approval of the Radiochemistry Division Leader.
 - b. The work is to be performed under the direct supervision of the Reactor Supervisor.

^aCause is understood and introduces no new hazardous condition.

^bAn incident or accident has occurred or scram's cause is in doubt and due to a potentially hazardous condition.

The list of OMP below represents the currently used procedures for normal reactor maintenance. The list may be enlarged or shortened as necessary to carry out the research reactor functions so long as the requirements of Sect. 13.4 are met.

- OMP #1 – Core Disassembly – Class II
- OMP #2 – Core Reassembly – Class II
- OMP #3 – Shim Rod Replacement – Class III
- OMP #4 – Regulating Rod Replacement – Class II
- OMP #5 – Magnet Removal, Replacement or Maintenance – Class II
- OMP #6 – Shim or Regulating Rod Actuator Maintenance or Replacement – Class II
- OMP #7 – Spent Fuel Element Removal from Pool – Class II
- OMP #8 – Spent Fuel Element Removal to Shipping Cask – Class III (includes "Pre-Fuel Shipment Checklist and Equipment Inventory")
- OMP #9 – Fuel Cycle (Standard Elements) – Class II
- OMP #10 – Fuel Cycle (Special Element) – Class II
- OMP #11 – Fission or Ion Chamber Maintenance or Replacement in LPTR – Class I
- OMP #12 – Addition of Water to Reactor Pool – Class I
- OMP #13 – Vacuuming Top of Fuel Elements – Class I
- OMP #14 – Underwater Source Transfer from Shielded Container to LPTR Pool – Class II
- OMP #15 – Large Source Removal from Reactor Pool – Class II
- OMP #16 – Shim Rod Gaging Procedure – Class II
- OMP #17 – Rod Travel Measurements – Class II
- OMP #18 – Fast Neutron Irradiation Facility Operation – Class I

13.5.2 Operational Safety Procedures

Operational Safety Procedures (OSP) are issued to assign responsibility, permit review processes, and identify hazards and counter measures for special situations generally related to experimental setups interfaced with the reactor. They constitute a supplementary set of procedures to the Reactor Building Procedure, OSP 281. A list of these procedures would be meaningless; they are all temporary and each expires on an assigned date.

The Hazards Control Department advises on safety matters related to any experiment and establishes the level of approval for each OSP in accordance with the LLL Health and Safety Manual.³

13.6 LPTR Record Retention

Records of significant reactor related events are retained at the LPTR to provide a review of data in evaluating events and to provide a basis for audit. Reactor records are broken down into three major categories and are retained according to AECM 0230. Disposition and retention periods are listed for specific types of records in the Disposition Schedule of the Central Files Index.

13.6.1 Process Records

Recorder outputs in the form of paper rolls or circular charts are retained for one year. Such recorder outputs include:

- Lead temperatures
- Coolant system temperatures
- Radiation levels (two systems)
- Log count rate recorder
- Period recorder
- Linear level recorder
- Log N recorder
- Coolant flow rate recorder.

Certain of these records may be retained indefinitely if they relate to a peculiar reactor problem or experiment.

13.6.2 Operational Records

Operational logs are retained for varying periods depending on the value of the information they contain. Those retained for one year include:

- Startup Checklist
- Weekly Checklist
- Shutdown Checklist
- Amplifier Checklists
- Off Shift Maintenance Checklist.

Those retained permanently include:

- Daily Log
- Reactivity Survey (weekly)
- Power Survey (weekly)
- Fuel Movement Forms
- Operation and Maintenance Procedures
- Maintenance Card Files.

13.6.3 Administrative Records

Administrative records, i.e. all other classes of recorded information include:

- Audit reports – retained permanently
- Occurrence reports – retained permanently
- Personnel Qualification records – retained permanently
- Safety Committee Proceedings – retained permanently
- Material Irradiation requests – retained for five years
- Irradiation Schedules – retained one year
- Operations Work Schedules – retained one week
- Weekly Operational Schedules – retained one week
- Irradiation charges – retained permanently
- Experiment reviews – retained for ten years
- Quality Assurance related information – retained permanently or until disposal of related equipment as applicable.

All records are retained in Building 281 in Room 1325 (Control Room) or 1173 (Main Office) with appropriate identification as indicated in the LPTR Files Index.

13.7 Industrial Security

13.7.1 Personnel and Physical Security

Personnel selection processes for the Laboratory as a whole screen out potentially maladjusted persons who may wish to damage the reactor complex because of malice or espionage. In general, persons working at the Laboratory must qualify for Q clearance. The reactor has only minimal physical security barriers, therefore, whereas the Laboratory site boundary surrounding the reactor is well secured and well monitored by the Security Department.

13.7.2 Security Plan

A badge check by the security officers at the Laboratory entrance allows a properly badged person to enter the complex surrounding the reactor area. No individual, even though he may be assigned work space in Building 281, is allowed into the reactor room by the operations staff without an escort unless he has been properly checked out on safety procedures. After indoctrination an individual may enter and leave the reactor room (1350) on his own.

Whenever the operations staff leaves the reactor unattended, they energize a circuit that signals the Security Department if a containment door is opened. Individuals previously authorized to enter, as indicated by a list held by the Security Department, must

phone in before entering and upon leaving. Security officers immediately check out any alarm not preceded by a call from someone authorized to enter.

New fuel is stored in the central vault under the custody of Materials Management. New fuel elements are sent to the reactor operations staff one at a time, as needed, at intervals of 3 weeks or more. In the rare cases when a fuel element arrives before it is needed, the operations staff stores it inside the reactor complex where it is safe from accident or tampering.

Two TV cameras in the reactor room allow the operations staff to observe any emergency situation. In general these two monitors are used for safety purposes and not personnel surveillance.

For accident conditions involving only LLL personnel, complete security checks exist for each person as indicated by his badge. If outside help is obtained then only preliminary security information may be available regarding those individuals.

13.7.3 Security Related Surveillance

In addition to routine surveillance carried on by the operations staff, maintenance machinists routinely enter and exit the reactor building or the containment structure in accordance with 13.7.2. These people can call the operations staff, Security Department or Hazards Control Department to respond to a given situation. In addition the Security Department Officers regularly go through the laboratory portion of the building on off shifts checking for abnormal situations.

REFERENCES

1. Disaster Control Plan, Lawrence Livermore Laboratory, Manual M-014, Rev. 2, (January 15, 1973).
2. G. Cummings, Livermore Pool Type Reactor Disaster Control Plan, Lawrence Livermore Laboratory Document O.S.P. 281, App. A., rev. by J. Eagan and R. Koopman, (August 1, 1970).
3. Health and Safety Manual, Lawrence Livermore Laboratory, Manual M-010, Rev. 6 (February 28, 1973).
4. Health and Safety Manual, op. cit., Section 1.11.

CHAPTER 14.0 INITIAL TESTS AND OPERATION

Tests and inspections¹ on the LPTR were carried out in 1957 and 1958 leading up to initial criticality on December 13, 1958. These tests included a) pre-neutron testing of process and control systems, b) fuel element handling, c) monitoring and test instruments. Post-neutron testing included nuclear calibrations and radiation surveys of the shielding. The test program was carried out with Foster Wheeler Corp. personnel and Laboratory employees from Plant Engineering and N-Division.

Additional tests were carried out by Laboratory personnel for major technical changes to the facility or power increases, Sec. 1.1.

14.1 Test Program

The following is a brief outline of the initial testing done at the LPTR.

- 1 Pre-Neutron testing
 - a. Water circulation and cleaning systems
 - b. Air cooling and building seal systems
 - c. Process and nuclear instrumentation
 - d. Pneumatic system
 - e. Control rod system
- 2 Fuel element handling
- 3 Monitoring and test instruments
- 4 Post-Neutron testing
 - a. Determination of neutron flux distribution
 - b. Calibration of LPTR control rods
 - c. Temperature and power coefficients of reactivity
 - d. Radiation survey

References

1. R. L. Ditterman, General Outline for Testing Livermore Pool Type Reactor, Foster Wheeler Corp., New York, N. Y. (1957).

15.0 ACCIDENT ANALYSIS

15.1 General

The LPTR contains no high pressure or high temperature process systems. Consequently only industrial hazards are directly associated with the process systems. However, the radiological hazard associated with misuse of the reactor or a loss of control of the fissile material can be severe enough to effect the health and safety of the site and community. For normal reactor operations, the laboratory's health and safety program as described in Chapters 11 and 12 is sufficient to limit the consequences of radiological hazards to laboratory personnel. Consequently, accidental misuse of a neutron beam or radioactive system will not be discussed here. Table 15.1-1 summarizes the potential accidents at the LPTR, the postulated worst case consequences, and the protective measures that mitigate the consequences of failure. Paragraphs a) through g) provide additional discussions of certain failures that could lead to potential radioactivity releases from the fuel:

- a) Loss of flow at power. This accident is not considered a serious threat to fuel plate integrity. Loss of flow initiates a scram signal. The fission heating in the core will decrease promptly and then eventually decay with an 80-s period. The fission product heating will initially be about 6.2% of the steady state power and decay as: $P = 0.062 P_0 t^{-1.2834}$ ($t > 10$ s). Heat from these two sources will have to be dissipated by natural circulation of the cooling water in the reactor tank. The power of the reactor due to delayed neutrons immediately after shutdown is given by:

$$\frac{P}{P_0} = \frac{\beta}{\beta + \rho},$$

where ρ is

$$\frac{K_{eff} - 1}{K_{eff}}$$

and P is the reactor power level immediately after shutdown from power level P_0 , by introduction of a step change in reactivity ρ (%). β is the effective delayed neutron fraction of the core. As a pessimistic case, a maximum value of 0.01 for β and a minimum shutdown reactivity of 0.03 will give $P = 0.25 P_0$. Actual measurement on the LPTR confirm that this is overestimated. If the maximum decay heat of the fission products immediately after scram is $0.07 P_0$, the total decay heat due to both fission-product and delayed-neutron heating may be conservatively estimated as $0.32 P_0$. For a steady state power level of 3 MW, the maximum power after shutdown will then be 0.96 MW.

Table 15.1-1. LPTR accident analysis.

Postulated Failure	Class	Reactor power MW	Postulated worst case consequences	Protective measures
1. <u>COOLANT SYSTEMS</u>				
a) Loss of primary flow	1	3	1. Boiling in core 2. Reactivity oscillations 3. No fuel element damage	1. Low coolant flow scram 2. Period scram 3. Power level scram
b) Partial core blockage	1	3	1. Boiling in core 2. Potential reactivity oscillations 3. Some fuel element melting 4. Some fission product release to primary coolant.	1. Filters and screens in primary system 2. Administrative procedures preventing work over core at power 3. Core T. V. 4. Rod guide tubes limit size of blockage
c) Loss of coolant (LOCA)	1	3	1. Potential fuel melt not expected 2. Fuel temperature increase to 85% of melt. 3. Intense radiation fields.	1. Pool low level scram 2. Low flow scram 3. Loss of moderator shutdown 4. Emergency core spray system. 5. Radiation alarms 6. Fire hoses
d) Loss of secondary coolant	1	3	1. Primary coolant boiling 2. Reactivity oscillations	1. Primary coolant alarm at 120°F 2. Primary coolant scram at 125°F 3. Power-flow alarm 4. Period rundown, scram
2. <u>FUEL ELEMENT CLAD</u>	1	3	1. Fission products in primary coolant	1. Fission products water monitor 2. Radiation alarms 3. Routine sampling of water
3. <u>REACTIVITY EXCURSIONS</u>				
a) Shim rod withdrawal during refueling	1	0	1. Uncontrolled criticality.	1. Procedural controls 2. Shutdown margin
b) Shim rod withdrawal from shutdown "start up accident"	1	0	1. Reactivity excursion 2. Power excursion 3. Some boiling	1. Period rundown 2. Period scram, slow and fast. 3. Power level scrams 4. Procedural controls
c) Shim rod withdrawal at power	1	3	1. Power excursion 2. Some boiling	1. Period rundown 2. Period scram—slow and fast 3. Power level scrams.

Table 15.1-1 (Continued)

Postulated Failure	Class	Reactor power MW	Postulated worst case consequences	Protective measures
d) Regulating rod withdrawal	1	0-3	1. Power ramp 2. Routine scram	1. Period rundown 2. Period scram (slow)
e) Experiment failure in tank with fissile material	1	3	1. Power transient, up or down 2. Fission products in primary coolant.	1. Administrative limits 2. Reactor scram instrumentation 3. Procedural controls 4. Experiment safety reviews
f) Failure of beam tube (flooding)	1	3	1. Power excursion 2. Possible boiling 3. No fuel damage	1. Administrative controls on heavy objects 2. Reactor scram instrumentation 3. Procedural controls 4. Shutdown margin 5. Total reactivity of all flooded beam tubes insufficient to cause melting
g) Fuel loading error (fresh element in center void)	1	0	1. Uncontrolled criticality	1. Administrative controls (procedures) 2. Round dummy present 3. Shutdown margin
4. <u>MECHANICALLY INDUCED DAMAGE</u>				
a) Tool or target mishandling	1	0	1. Fuel element clad break	1. Administrative controls 2. Tool design
b) Fuel mishandling	1	0	1. Clad damage	1. Operator proficiency
5. <u>INADVERTENT CRITICALITY</u>				
(Out of core)	1	0	1. Uncontrolled criticality	1. Cannot have criticality with plate type elements except in moderator 2. Limit of 7 elements in shipping configuration (subcritical) 3. Administrative controls
6. <u>ALUMINUM—WATER REACTION</u>				
	1	3	1. Fuel element destruction with significant energy release; fission products released to water	1. No way to get aluminum hot enough (1100°C) to start reaction 2. SPERT 1-D and SW-1 indicate low probability of such a reaction

Table 15.1-1 (Continued)

Postulated failure	Class	Reactor power MW	Postulated worst case consequences	Protective measures
7. <u>CONTROL ROD FAILURES (SHIMS)</u>				
a) Rod falls out of core	1	0-3	1. Uncontrolled reactivity insertion	1. Rod is restrained by geometry of "special" fuel elements 2. Hold downs prevent "special" element movement
b) Rod rupture	1	0-3	1. Uncontrolled reactivity insertion	1. Stainless steel sheathing and reinforcing of rod prevent swelling or rupture 2. Routine swelling checks 3. Reactivity limit requires one-rod-stuck-out shutdown capability
8. <u>EARTHQUAKE</u>				
	1	3	1. Physical damage to entire facility 2. Loss of water 3. Damage to containment structure 4. Loss of services 5. Potential core damage	1. Earthquake could not topple reactor pool tank 2. Loss of water will not cause fuel melt 3. Loss of containment would not cause D.B.A.
9. <u>TORNADO</u>				
	1	3	1. Physical damage to support building, containment, and services	1. Reactor would scram on loss of a-c power 2. Loss of primary flow creates no boiling in core 3. Reactor pool tank would not be damaged
10. <u>LOSS OF A-C POWER</u>				
	1	3	1. Loss of primary flow 2. Reactor shutdown	1. Automatic reactor scram on loss of a-c power 2. Key instrumentation and emergency systems are on emergency generator 3. Loss-of-a-c-power scram precludes boiling in core
11. <u>DESIGN BASIS ACCIDENT (DBA)</u>				
	3	3	1. Gross fission product leakage to containment 2. Some leakage to environment	1. Safety circuits 2. Safety systems 3. Engineered safeguard features 4. Administrative controls

It has been shown¹ that the LTR, which is similar to the LPTR, could be operated without boiling at powers above 1 MW without forced convection cooling. Therefore, by analogy, a shutdown power of 0.96 MW for the LPTR constitutes no hazard. The above reference also shows that if the low flow scram should fail, the water would eventually boil and shut down the reactor by a high power level or period scram. The data also show that no fuel damage would result.

- b) Overheating of fuel element plates due to partial flow stoppage. This has happened on several reactors resulting in gross fission product releases to the coolant. Reactivity monitors as well as water monitoring equipment detected the damage and shut the reactors down. At the LPTR no rigid object could block flow to more than 5 elements (1 row) at one time because the control rods and guide tubes protrude from the reactor top side. Assuming 6 elements were blocked at once, then conceivably 24% of the core could be subject to fuel element damage, and could release fission products to the coolant. This is highly unlikely accident which creates a markedly less severe situation than that analyzed in Sec. 15.4.
- c) Loss of coolant following a scram. The fuel element temperature following a scram would climb to an equilibrium value dependent on time since scram and environmental conditions. A detailed analysis in Sec. 15.2 indicates that the peak fuel element temperature would be at least 55 K below melting.
- d) Loss of coolant at power. This accident is the worst case for temperature build-up from decay heat. Loss of coolant from the core would stop the fission power production. Nuclear instruments, coolant system scrams, and radiation alarms conservatively prevent this accident from occurring. The fuel elements are not expected to reach the aluminum melt point under these circumstances as indicated in Subsection 15.2.1.
- e) Cladding failure due to manufacturing error. Historically, clad failures have been detected and faulty fuel elements removed without releasing fission products to the containment atmosphere.
- f) Power excursion due to uncontrolled reactivity insertion. This accident is specifically analyzed in Subsection 15.2.2.
- g) Reactivity insertion due to experiment failures. The reactivity value of movable experiments (target within pool) is administratively limited; uncontrolled movement cannot produce an asymptotic period of less than 20 s. Sudden flooding of all beam tubes on the north side of the core is estimated to be worth less than 3.8^2 which would not produce fuel melting as shown in Subsection 15.2.2.

The consequences following such accidents fall into three general classes:

- 1) No radioactivity release to the containment, Class 1.
- 2) Moderate radioactivity release to the containment, Class 2.
- 3) Design basis accident (DBA) - Gross radioactivity release to containment and specified leakage to the environment, Class 3.

A reactor with minimal decay heat and located in a pool only slightly above ambient temperature lacks the usual power reactor mechanisms for emitting significant quantities of radioactivity to the containment or the environment. The reactor pool water has no stored energy and consequently no major coolant removal force; it acts as a tremendous barrier in preventing fission product release.

15.2 Events Resulting in No Radioactivity Release to the Containment, Class 1

15.2.1 Loss of Coolant (LOCA)

In large power reactors loss of coolant is a serious concern. Decay heat buildup following such an accident is enough to cause fuel melting, if the emergency core cooling system failed, releasing fission products to the containment building. For the LPTR, loss of coolant is of lesser concern. Fuel melting would not occur and fission products would therefore not be released.

15.2.1.1 Identification of Causes and Controls—The LPTR core's centerline is 1.2 m (4 ft) above the tank floor. The core is centered within an aluminum tank 10 mm thick, 2 m in diam, and 8 m deep. The tank is filled with water to the 7.3-m (24-ft) level (nominally). An 8-in. outlet line leads to the heat exchanger. Several beam ports project radially from the core through the heavy concrete walls surrounding the tank.

A loss of coolant accident thus could occur in any of three ways:

- a) rupturing of the primary coolant inlet or outlet pipes.
- b) severing of any beamport (refer to Chapter 4.0)
- c) cracking of the aluminum tank.

Portions of the coolant inlet and outlet lines outside of the equipment pit are buried underground and the aluminum tank is surrounded by concrete. This would impede the flow from any break in these locations. A perhaps more likely event is the severing of a beam port. This could occur by accidentally dropping a spent-fuel-element cask on a beam port. Such an accident would happen only during a fuel or target reloading, however, i.e. during a reactor shutdown.

15.2.1.2 Low Pool Level Scram—When the pool level drops below 1.2 m from the top of the tank, a float-actuated device scrams the reactor and shuts off the primary pump motor.

15.2.1.3 Emergency Core Cooling System (ECCS)—Refer to Sec. 6.3.

15.2.1.4 Analysis of Effects and Consequences—To find the maximum fuel temperature following a loss of coolant, we consider:

1. Case 1. Outlet or inlet pipe breaks, completely draining tank.
2. Case 2. Beam port B-1 ruptures, (worst case), leaving core partly submerged.

Both of the above cases will be considered with and without the ECCS working. In case 1, the core is cooled entirely by air passing down through the outer rows of elements and up through the inner elements. For case 2, however, the flow areas for circulating air are blocked. The following analysis will show that case 2 is the worst case (provided the ECCS is inoperable). Note that a break of the inlet pipe has a similar effect as a break in the outlet pipe as long as both legs are allowed to drain. Since all valves in the primary loop are open during operation this would be the expected case. Valving off the outlet line manually after the break could produce a situation similar to case 2.

15.2.1.4.1 Tank is Entirely Drained—Case 1—The original analysis performed in the 1966 SAR Reference showed that the maximum fuel temperature following a loss of coolant is 767 K (921°F) assuming a 10-s drain time.³ This analysis included natural convection to 311 K (100°F) ambient air from an infinite supply below the plenum and did not include heat conduction to the grid plate. A more realistic model should include the effects of heating the air as it circulates down through the outer fuel elements. Also heat conduction and the effect of air flow resistance on the heat transfer coefficient should be considered. In essence, Omega West at Los Alamos included these effects in their LOCA analysis.⁴ Omega West has nearly the same plate-type elements as the LPTR. Thus the revised analysis of the LPTR will closely parallel the OWR analysis with some slight modifications.

15.2.1.4.1.1 The Model—The analysis in essence requires the solution of two coupled differential equations. The first differential equation describes a heat balance between the hottest fuel element (location C-3 hereafter referred to as element C-3) and the air which flows up through it. A decay heat source term is included. The model also allows for heat conduction between the fuel element and the grid plate.

The second differential equation sets up a heat balance between the air and the tank wall (assumed to be an infinite heat sink). The source term for this equation is the convective heat transfer term of equation one.

15.2.1.4.1.2 Heat Balance of Hottest Fuel Element (C-3)—The first differential equation may symbolically be written

$$q_s - q_{cd}(f \rightarrow b) - q_{cv}(f \rightarrow a) - MC_p \frac{dT_f}{dt} = 0 \quad (1)$$

where

q_s = decay heat generation rate of element C-3.

q_{cd} = heat conduction rate between fuel element and grid plate.

$\frac{(MC_p)dT_f}{dt}$ = net heat absorption rate.

q_{cv} = convective heat transfer rate from fuel to air.

a) Source term—The source term for the afterheat generation rate in a fuel element is taken to be⁵

$$q_s = \left\{ 0.13 \left[t^{-0.2834} - (t + T_0)^{-0.2834} \right] P_e + P_r \right\} \times F \quad (\text{MW})$$

t = time after shutdown (s)

T_0 = reactor operating time (s)

P_e = element operating power level, element C-3

F = fraction of total $\beta + \gamma$ decay heat deposition = 0.79 (in fuel element)

P_r = residual heat generation

Normal operation for the LPTR may include 5 days of continuous operation followed by 2 days of shutdown. For the worst possible situation the reactor operating time prior to shutdown, T_0 , will be assumed to be infinite.

$P_e = 0.201 \text{ MW}$ (6.7% of total power)

$P_r = 0$

$T_0 = \infty$

- b) Conductive heat transfer— $q_{cd}(f \rightarrow b) = H_{cd}(T_f - T_b)$, where H_{cd} is the coefficient for heat conduction from the fuel to the lower grid to the base casting. The value of this coefficient measured⁶ by OWR was $H_{cd} = 3.91 \times 10^{-7}$ MW/K (2.17×10^{-7} MW/°F) per fuel element. This is in good agreement with 2.2×10^{-7} MW/°F per fuel element as reported by ASTR.⁷ In these calculations we assume $T_b = 366$ K (200°F).
- c) Convective heat transfer— $q_{cv}(f \rightarrow a) = H_{cv}(T_f - T_a)$, where H_{cv} is the convective heat transfer coefficient. Our value is based on measurements of the maximum temperature attained by OWR fuel elements removed from cooling water and suspended in air. After correcting for radiative heat transfer, the convective heat transfer coefficient was determined⁸ within $\pm 30\%$ confidence to be

$$H_{cv}(f \rightarrow a) = 10^{-7} (40.9 + 1.5(T_f - T_a)^{0.55}) \text{ (MW/K)}$$

where

T_f = fuel temperature, K

T_a = air temperature, K.

The above result is approximately 3/4 of the heat transfer coefficient of the vertical plate model used in the 1966 SAR; the decrease in the above heat transfer coefficient is due to air flow resistance in the narrow fuel channels.

- d) Heat capacity of the fuel elements—the mass of the fuel elements in the calculation of the heat capacity⁹ is taken conservatively to be 4.4 kg. This includes only the masses of the fuel plates and side plates but not the end boxes.

$$MC_p = 10^{-6} (3830 + 1.23 T_f) \left(\frac{\text{MW-s}}{\text{K}} \right)$$

Substituting the terms into Eq. (1) and rearranging we obtain

$$q_s - H_{cd}(T_f - 366) - H_{cv}(f \rightarrow a)(T_f - T_a) = MC_p \frac{dT_f}{dt} \quad (1')$$

Heat Balance of Air in Tank—The temperature of the circulating air in the tank obeys:

$$\frac{P_t}{P_e} q_{cv}(f \rightarrow a) - q_{cv}(a \rightarrow w) - V \rho_a C_p \frac{dT_a}{dt} = 0 \quad (2)$$

where

$$P_e = 0.201 \text{ MW (C-3)}$$

$$P_t = \text{total operating power, 3 MW}$$

$$V = \text{volume of air in tank}$$

$$\frac{P_t}{P_e} q_{cv}(f \rightarrow a) = \text{convective heat rate from all the fuel elements to the air}$$

$$q_{cv}(a \rightarrow w) = \text{convective heat loss from air to tank wall}$$

$$V \rho_a C_p \frac{dT_a}{dt} = \text{net heat absorption rate of air.}$$

Equation (2) neglects the effect of circulation of outside air. We conservatively assume zero net displacement of the air in the tank.

- a) Convective heat transfer from air to tank walls $= q_{cv}(a \rightarrow w) = H_{cv}(a \rightarrow w)(T_a - T_w) \times A_w$, where H_{cv} is the convective heat transfer coefficient:

$$H_{cv}(a \rightarrow w) = 1.08 \times 10^{-6} (T_a - T_w)^{1/3} \frac{\text{MW}}{\text{m}^2 \cdot \text{K}}$$

- b) Volumetric heat capacity of air, the product of the specific heat and density of air at low altitudes, is given by

$$\begin{aligned} \rho_a C_p &= 10^{-6} (1240 - 1.12 T_a) \frac{\text{MW} \cdot \text{s}}{\text{m}^3 \cdot \text{K}} \\ &= (1.24 \times 10^{-3} - 1.12 \times 10^{-6} T_a) \end{aligned}$$

- c) Other LPTR parameters include:

$$\text{Area of tank wall (including wall and floor)} = A_w = 42.6 \text{ m}^2$$

$$\text{Volume of water above grid plate} = \text{volume of air} = V_a = 19.8 \text{ m}^3$$

Substituting the required terms into Eq. (2) and rearranging produces

$$\frac{P_t}{P_e} H_{cv}(f \rightarrow a)(T_f - T_a) - H_{cv}(a \rightarrow w)(T_a - T_w)A_w = V \rho_a C_p \frac{dT_a}{dt} \quad (2')$$

Substituting empirical values for the terms in Eqs. (1') and (2') gives us

$$0.0207 t^{-0.2834} - 3.91 \times 10^{-7} (T_f - 366) = 10^{-7} [40.9 + 1.5(T_f - T_a)^{0.55}]$$

$$\times (T_f - T_a) = 3.83 \times 10^{-3} + 1.23 \times 10^{-6} \times T_f \frac{dT_f}{dt} \quad (1'')$$

and

$$10^{-8} \left\{ 49.3 + 1.89(T_f - T_a)^{0.55} \right\} (T_f - T_a) - 5.58 \times 10^{-5} (T_a - T_w)^{4/3} \\ = \left\{ 2.45 \times 10^{-2} - 2.22 \times 10^{-5} T_a \right\} \frac{dT_a}{dt}. \quad (2'')$$

With the initial conditions

$$T_f = 366 \text{ K (fuel temperature)}$$

$$T_a = 311 \text{ K (air temperature)}$$

$$T_w = 311 \text{ K (wall temperature)}$$

15.2.1.4.1.4 Time to Drain—For an upper bound on the fuel plate temperature we assume that the pool drains instantaneously. A more realistic accident would include the rupture of the 8-in.-i.d. outlet pipe which corresponds to a drainage time of 68 s, assuming frictionless flow.

The two coupled nonlinear differential equations were solved using a CDC 6600 computer and the Runge-Kutta technique. The subroutine used (INTS, for INTEgrate a System of ordinary differential equations) employs Hamming's fifth order, stable, predictor method with adaptive integration steps.

15.2.1.4.1.5 Results—The maximum fuel temperature in case of an instantaneous loss of coolant is 505°C (see Fig. 15.2-1). Using a 2-σ value of 85 K¹⁰ (as calculated in the OWR analysis) one can assert with a 95% degree of confidence that the fuel temperature will not exceed 860 K, 55 K lower than the melting point of aluminum. This is in agreement with the LITR reference analysis¹¹ showing that the large margin of safety inherent in 3 MW operation makes the emergency core spray system unnecessary in the event of a total loss of coolant.

Figure 15.2-1 also reveals that even if the ECCS were inoperative the maximum fuel plate temperature eventually reached would be below the softening temperature of aluminum by a 2-σ confidence level.

Figure 15.2-2 shows that the time to reach the maximum fuel temperature as a function of time after shutdown exhibits a broad maximum with the time to uncover core as a parameter. For the particular case of a drainage time of 68 s (corresponding to the rupturing of the 8 in. i.d. outlet) it takes 1500 s after shutdown to reach the maximum fuel temperature.

Fig. 15.2-1. Maximum fuel temperature vs time to drain pool, assuming core spray failure.

The above results are highly conservative. First, a drainage time of 68 s is very pessimistic. The tank lid is closed during normal operation, offering some resistance to free-flow drainage. Thus a more realistic drainage time would be several minutes. If for some reason the tank lid were open during the loss of coolant then air would be free to circulate out of the tank, an effect not included in the overall heat balance. Secondly, we neglected radiative heat transfer. Finally, note that the central dummy element (empty except for an orifice plate the size of a regular element) is right next to the hottest fuel element, C-3. Air flow through the dummy element can efficiently cool fuel element C-3.

15.2.1.4.2 Partial Submersion—Case 2—The partial loss of coolant resulting from the rupturing of beam port B-1 (the worst case since B-6 is not removable) would leave the water level 85 mm above the bottom of the fuel element and thus block internal air convection completely.

The Omega West staff at Los Alamos performed partial immersion tests¹² with fuel plates identical to the LPTR. Their results are:

Fig. 15.2-2. Maximum fuel temperature vs time after shutdown for different times to uncover the core.

"Fuel-plate temperatures of inductively heated simulations of the aluminum MTR-type fuel elements used in the Omega West reactor have been measured as a function of power input and plate immersion depth in water. At immersions ≤ 5 in., the primary cooling mechanism is production and superheating of steam. A much more efficient cooling mechanism, percolation, becomes dominant at immersions ≥ 5 in. Percolation carries liquid water directly up into the cooling channels and cools so effectively that increased input power decreases plate temperature, within the range of input power (≤ 5.1 kW) available." (Five inches ≈ 130 mm.)

Their results are depicted in Fig. 15.2-3.¹³

15.2.1.4.2.1 Analysis—The heat source term for the hottest fuel element C-3 in the LPTR is

$$q_B = 20.1 t^{-0.2834} \text{ kW}$$

where t is in seconds.

It is of interest to find the average power for fuel element C-3 over the time interval of one hour (time at which fuel element reaches its maximum temperature) allowing for a

Fig. 15.2-3. Maximum fuel-plate temperatures vs power input and vs immersion depth for inductively heated elements partially immersed in water.

drainage time of 157 s (frictionless flow through full beam-port aperture). The average power for the fuel element over this time interval ($157 \leq t \leq 3757$ s) is

$$q_s = \frac{20.1 \int_{157}^{3757} t^{-0.2834} dt}{\int_{157}^{3757} dt} = 2.55 \text{ kW}$$

For an average power level of 2.50 kW and an average submersion of 85 mm (3.38 in.) of water the maximum fuel temperature attained is 784 K (965°F), indicated by the solid square in Figs. 15.2-3.

15.2.1.5 Radiation Dose Rates Following LOCA—Radiation levels inside and outside the containment structure following coolant drainage have been measured¹⁴ at 2 MW. These measurements were then extrapolated to 3 MW operation.

After steady state operation at a power level of 2 MW, the reactor was shut down and the pool water was drained into the 37.9-m³ (10,000-gallon) storage tank until it was level with the top of the core. The core itself was not exposed. The water receded about 5.1 mm/s (1 ft/min); the water level dropped to the top of the core in about 1200 s.

Gamma and fast neutron radiation levels were measured in and near the reactor building. Isodose plots of levels outside the building and of representative readings inside the building are shown in Fig. 15.2-4. The radiation levels were corrected to 600 s after reactor shutdown. Radiation levels on top of the reactor and on the reactor building are shown in Fig. 15.2-5. Nearly all the observed radiation outside the building and on the reactor room floor is gamma radiation from skyshine; the neutron contribution is less than 3% of the total dose.

Extrapolating this data to 3 MW, reveals that the maximum reading at the reactor room floor would be about 0.6 rem/hr. The gamma field 6.1 m above the core would be 7125 rem/hr 600 s after shutdown. Thus, the loss of water shielding at the LPTR would cause serious gamma or neutron radiation fields only directly over the core. The dose rates would approximately double if the entire core were exposed.

15.2.2 Reactor Start-Up Accident—Analysis

Continuous withdrawal of the shim bank produces the most hazardous operations-type accident. It is extremely unlikely to occur, since it is contrary to established operating techniques and assumes multiple failure of almost all scrams and rundowns.

The available scrams and rundowns appear in Table 15.2-1. Paragraph 3b of Table 15.1-1 lists other features of this failure.

The accident proceeds under the same assumptions as earlier 1 MW and 2 MW analyses:

- The reactor is initially shut down.
- The shim rods are withdrawn simultaneously at their maximum rate.
- All period and level scrams and rundowns fail except one level safety fast scram at 150% of full power.
- The shim rods continue out for 20 ms (the measured delay time) after this level has been reached and then fall into the core with an acceleration of 5.2 m/s².

NOTES:

Reactor balcony (A), 292 mrem/hr,
top of reactor, 6.1 m above core, 7125 rem/hr

Fig. 15.2-4. Isodose contours and radiation levels in and near reactor building.
Water level down to core top. Readings in mrem/hr, corrected to
600 s after shutdown from 3 MW.

Fig. 15.2-5. Radiation levels at building surface in rem/hr at 600 s after shutdown from 3 MW. Water level down to core top.

Table 15.2-1. Scrams and rundowns available to prevent a reactor start-up accident.

<u>Scrams, period</u>		<u>Rundowns, period</u>	
	s		s
Log count rate	5	Log count rate	20
Log N	5	Log N	20
Period safety amplifier, slow	3.5		
Period safety amplifier, fast	1		
<u>Scrams, power level</u>		<u>Rundowns, power level</u>	
	MW		MW
Level safety amplifier, slow (2 each required)	4.2		
Level safety amplifier, fast (2 each required)	4.5		
Linear level recorder, slow	4.5	Linear level recorder	3.9
Log N, slow	6	Log N	4.5

Four cases were investigated: nominal (500 mW) and minimum (as required by startup instrumentation) (3.53 mW) startup sources coupled with the maximum (\$18) and minimum (\$7.5) possible shim bank worths. Shutdown margins were taken as \$9.1 and \$3.8 respectively.

The above was translated into input for the Atomics International AIREK code.¹⁵ AIREK is a point reactor generalized kinetics code which solves the standard reactor kinetics equations by a fourth-order Runge-Kutta technique. The code accepts up to 40 sets of reactivity/time values and has provision (although unused) for temperature coefficient feedback. The code was run twice for each case, the first time to determine the time of scram. Table 15.2-2 presents the final results.

Table 15.2-2. Powers and periods for start-up accidents under various source and bank worth conditions.

Case	Maximum power (MW)	Integrated power (MW · s)	Minimum period (ms)
Minimum source Minimum bank worth	5.081	3.136	386.7
Minimum source Maximum bank worth	6.232	1.383	138.2
Nominal source Minimum bank worth	4.684	4.345	595.0
Nominal source Maximum bank worth	5.577	1.753	204.7

Since SPERT I-D¹⁶ data indicates that for a core similar to the LPTR it takes periods less than about 6 ms (accompanied by a 1000-MW power peak) to melt fuel elements, and since the above analysis conservatively neglected negative temperature coefficient feedback, one must conclude that the above accident is of minimum hazard.

15.3 Events Resulting in Small to Moderate Radioactive Releases to the Containment. Class 2

15.3.1 Failure of Fueled Experiment

Any number of useful experiments can be conducted by irradiating materials within and around the reactor. The potential hazard depends on the size and physical properties of the target materials. Fueled experiments constitute a major source of radioactivity that can be generated by irradiation in the reactor. Such experiments are very carefully evaluated for individual containment and heat removal mechanisms. Additional protective circuitry may supplement the reactor control system in connection with fueled experiments.

Hazardous materials such as ²³³U, ²³⁵U, or ²³⁹Pu, having serious biological as well as radiological and nuclear consequences if mishandled, are closely governed by the Laboratory's Health and Safety Manual, Section 2.04. The limits specified in that manual are based on overall safety considerations, including individual nuclear criticality. All major experiments are reviewed by the Reactor Safety Committee. Every experiment is regularly monitored by the operations staff and Hazards Control personnel.

The radioisotopic inventory of an experiment and its accidental dispersion can be analyzed in the same manner as the hypothetical core melt covered in Sec. 15.4. The dose rates shown in its tables for 3 MW may be directly scaled down to the isotopic inventory generated by the specific experiment.

15.4 Analysis of Event Leading to Major Radioactive Release to the Containment and the Exclusion Boundary. Class 3

15.4.1 Design Basis Accident

15.4.1.1 Identification of Causes and Controls—A light-water research reactor operated at 3 MW in a pool near ambient conditions lacks the usual mechanisms for an accident leading to total fission product release from the fuel as discussed in the loss-of-coolant accident and the start-up accident (Sec. 15.2). No research or test reactor has ever had fission product release from a loss-of-coolant accident. Consequently, no empirical data exist for evaluating such an accident.

The combination of plate-type aluminum-clad fuel elements in a covered tank, with burn distributed in each element and with emergency cooling protection, suggests that this type of accident would lead to only fractional release of fission products to the containment. The 35% fuel damage in the SPERT tests and the 40% fuel damage during start-up-type accident¹⁷ in the SL-1 reactor support this hypothesis.

The following calculations deal with a total core melt, however. The results may be scaled down for any lesser accident involving fuel element damage in the absence of coolant. We make the following assumptions:

- Cause; additional energy from unknown sources
- Contributing cause; no coolant
- Result: total core melt
- 100% release of noble gases to containment¹⁸
- 25% release of radioiodines to containment¹⁸
- 0.25% per day gaseous leak rate from containment (500 Pa sustained overpressure, Sec. 6.2)
- Other specific assumptions will be presented as required.

15.4.1.2 Fission Products Inventory—Radioiodines responsible for thyroid and whole-body dosage, and the noble radiogases contributing to whole body dosage, will be specifically analyzed. We assume a fission product concentration in the reactor core equal to that obtained after 15% average fuel burnup. If the cold clean mass of the core is 4.2 kg ^{235}U , then 630 g of fission products would be available. This level of burnup assumes continuous reactor operation for 105 days (a number which is used only in determining isotopic build-up factors) consuming ^{235}U at 1.31 g/MW day. Table 15.4-1

Table 15.4-1. Half lives and activities of radioisotopes of interest in a core melt accident.

Isotope	Saturation activity (kCi/MW)	Half-life	Build-up ^a factor	Final activity level, 3 MW for 105 days (kCi)
¹³³ Xe	53.2	5.3d	1.00	159.6
¹³⁵ Xe	59	9.1h	1.00	177
⁸⁵ Kr	2.7	10.7y	0.0184	0.05
¹³¹ I	21.6	8.14d	1.00	64.8
¹³² I	37.7	2.4h	1.00	113
¹³³ I	53.2	20.5h	1.00	159.6
¹³⁴ I ^b	61.8	52m	1.00	185.4
¹³⁵ I	59	6.68h	1.00	177

^aBuild-up factor for 105 days: $(1 - e^{-\lambda_i t})$.

^bData extrapolated from W. E. Draper, Effluent and Filter Monitor System, General Dynamics Corporation, Fort Worth, Texas, Rept. FZK-336 (1967).

shows the resultant curie content for isotopes of primary interest.¹⁹ Other isotopes present as a result of the accident produce only a small fraction of the iodine doses or escape to the containment only in insignificant amounts.

15.4.1.3 Filtered and Unfiltered Emission Rates from the Containment Structure—The following assumptions will be used in determining emissions from the containment building:

- The total containment building volume is approximately 7100 m³ (250,000 cubic feet). Assume 10% is occupied by equipment, leaving a net usable volume of 6400 m³
- Let 40 cfm (18,800 cm³/s) escape through the filter/scrubber system (stack height of 15.8 m) during the 500 Pa overpressure. (Refer to Ch. 6.2 for design description of system.)
- The filter/scrubber system is assumed to capture 100% of all particulates and 95% of all radioiodines including organic iodines within the charcoal and HEPA filter bank. The caustic scrubber will be assumed ineffectual.
- For simplicity we conservatively neglect the effect of stack height on the distribution of radionoble gases assuming them instead to be emitted at ground level only.

- e) The ground-level, uncontrolled, unfiltered leak rate of radioiodines at 500 Pa overpressure is 0.25% per day, extrapolated from measured leakage at 14 kPa. Thus $0.0025/\text{day} \times 6400 \text{ m}^3 = 16 \text{ m}^3/\text{day}$ or $185 \text{ cm}^3/\text{s}$.

Using the values of initial isotope content shown in Table 15.4-1 and the above assumptions, the hypothesized radioactivity leak rates can be determined. Table 15.4-2 displays the result.

Table 15.4-2. Isotopic concentrations and emission rates.

Isotope	Released to containment %	Concentration in containment (Ci/m ³)	Uncontrolled ground level emission (mCi/s)	Filtered stack emission (mCi/s)
¹³³ Xe	100	25.04	475	—
¹³⁵ Xe	100	27.77	526	—
⁸⁵ Kr	100	0.008	0.15	—
¹³¹ I	25	2.54	0.46	2.39
¹³² I	25	4.43	0.81	4.16
¹³³ I	25	6.26	1.15	5.88
¹³⁴ I	25	7.27	1.34	5.8
¹³⁵ I	25	6.94	1.28	6.5

15.4.1.4 Diffusion Models¹⁸—The diffusion model parameters for site boundary and nearest population center dose determinations are conservatively set as follows:

- The wind speed (μ) is assumed to be 1 m/s for the first 24 hr.
- The nearest site boundary is the west boundary at 440 m.
- Pasquill Type F atmospheric conditions exist for the first 24 hr.
- For the first 8 hr following the release:
 - The ground level diffusion equation is:

$$X/Q = \frac{1}{\pi \mu \sigma_y \sigma_z} \text{ s/m}^3$$

- The elevated (stack) diffusion equation is

$$X/Q = \frac{\exp - (h^2/2 \sigma_z^2)}{\pi \mu \sigma_y \sigma_z} \text{ s/m}^3$$

- e) After 8 hr: The ground level diffusion equation, taking into account sector averaging, is

$$\chi/Q = \frac{2.032}{\sigma_z \mu x}.$$

The stack diffusion equation, employing sector averaging, is:

$$\chi/Q = \frac{2.032 \exp - \left(\frac{h^2}{2\sigma_z^2} \right)}{\sigma_z \mu x}.$$

The resulting radioiodine concentrations for specific distances and durations of exposure are shown in Table 15.4-3. Note that the 440 m boundary is the nearest boundary and that concentration is arrived at without consideration of the more probable meteorologic effects, or radioactive decay. Each concentration value is the sum of the unfiltered ground leakage plus a fraction of filtered stack emission.

Table 15.4-3. Concentration of radioiodines.

Isotopes	Concentrations ^a 440 m site boundary; ≤8 hr μCi/m ³	440 m site boundary; ≥8 hr μCi/m ³	1609 m radius, 24 hr μCi/m ³
¹³¹ I	1.77	0.43	0.142
¹³² I	3.11	0.75	0.246
¹³³ I	4.11	1.06	0.356
¹³⁴ I	5.14	1.23	0.414
¹³⁵ I	4.9	1.18	0.39

^aUSAEC Guides 3 and 4 allow sector averaging for times greater than 8 hr.

The resulting radioactive noble gas concentrations for specific distances and duration of exposure are shown in Table 15.4-4. The concentration values were arrived at as if all the emission occurred at ground level, i.e., sum of stack emission and ground level leakage.

Table 15.4-4. Concentrations of radioactive noble gases.

Isotope	Concentrations ^a 440 m boundary ≤8 hr μCi/m ³	440 m boundary 24 hr μCi/m ³	1609 m radius 24 hr μCi/m ³
¹³³ Xe	1040	270	53
¹³⁵ Xe	1160	300	36.8
⁸⁵ Kr	0.33	0.08	0.01

^aConcentrations based on ground level emissions for simplicity and conservatism.

15.4.1.5 Biological Uptake and Dose Rates—The dose to the thyroid gland consists of doses caused by the iodine series of radioisotopes. The related dose conversion constants used for dose calculations will assume a breathing rate of 30 m³/day for the first 24 hr instead of 20 m³/day as used in less conservative situations.

For whole body and thyroid gland doses, the individual is assumed to stand in the plume centerline except in the case of sector averaging. The resultant whole body dose rate is then arrived at as if the radioactive cloud of specified radioisotope concentrations were infinite.¹⁸

Table 15.4-5. Thyroid and whole body concentration conversions to dose constants.

Isotope	Thyroid dose conversion constant ^a $\frac{\text{rad}}{\text{s}} \frac{\text{Ci}}{\text{m}^3}$	Whole body conversion constant $\frac{\text{rem}}{\text{hr}} \frac{\text{Ci}}{\text{m}^3}$
¹³¹ I	514.5	3100
¹³² I	18.6	130
¹³³ I	138.5	620
¹³⁴ I	8.7	50
¹³⁵ I	42.9	350
¹³³ Xe	— ^b	250
¹³⁵ Xe	— ^b	620
⁸⁵ Kr	— ^b	620

^a Assumes 30 m³/day breath rate.

^b Insignificant thyroid dose from noble gas radionuclides.

in excess of two hours. The specific isotopes making the dominant contribution to whole body immersion doses are the radionoble gases.

The conversion constants used for thyroid dose²⁰ and whole body doses²¹ are listed in Table 15.4-5.

15.4.1.6 Resultant Exposures at Specified Distances and Times—Resultant thyroid gland exposures at 440 m (the nearest site boundary) and at 1609 m (1 mile), the nearest higher populated zone, are shown in Table 15.4-6. For times greater than two hours, isotope concentrations are averaged over the duration of exposure.^{*}

Resultant whole-body doses due to passage of the radioactive cloud of calculated concentrations are shown in Table 15.4-7. Appropriate averaging factors were used for exposure durations

$$A_{av} = A_0 \frac{\int_{t_1}^{t_2} e^{-\lambda_1 t} dt}{\int_{t_1}^{t_2} dt}$$

Table 15.4-6. Thyroid doses due to iodine inhalations.

Isotope	Dose ^a at 400 m (rem)		Dose ^b at 24 hr (rem)	
	2-hr exposure	8-hr exposure	440-m boundary	1609-m radius
¹³¹ I	6.56	25.2	12.2	4.02
¹³² I	0.42	0.65	0.11	0.04
¹³³ I	4.10	14.4	5.84	1.96
¹³⁴ I	0.32	0.1	0.01	0.01
¹³⁵ I	1.51	4.1	1.06	0.35
Totals	12.9	44.4	19.22	6.39

^aThe diffusion model listed in paragraph 15.4.4d applies only for exposures ≤ 8 hr.

^bThe diffusion model listed in paragraph 15.4.4e (22-1/2° sector averaging) applies to entire 24 hr period (i.e., not additive to 8 hr exposure).

Table 15.4-7. Whole body doses from immersion in airborne radioactivity.

Isotope	Dose ^a at 440 m (mrem)		Dose ^b at 24 hr (mrem)	
	2 hr	8 hr ^c	440 m	1609 m
¹³³ Xe	520	2034	1520	185
¹³⁵ Xe	1450	4362	2088	265
⁸⁵ Kr	0.5	2	1.2	— ^e
¹³¹ I	11	44	31	11
¹³² I	0.8	3	— ^e	— ^e
¹³³ I	5	20	10	— ^e
¹³⁴ I	0.5	— ^d	— ^e	— ^e
¹³⁵ I	3	14	— ^e	— ^e
Totals	2 rem	6.5 rem	3.7 rem	0.5 rem

^aThe diffusion model listed in paragraph 15.4.4d applies only for exposures ≤ 8 hr.

^bThe diffusion model listed in paragraph 15.4.4e (22-1/2° sector averaging) applies to entire 24-hr period (i.e., not additive to 8-hr exposures).

^cIsotopic decay allowed.

^dLess than 0.5 mrem.

^eLess than 1 mrem.

Resultant thyroid gland doses due to radioiodine and whole body doses due to the radioactive cloud, conservatively meet site boundary criteria specified in 10 CFR 100 for power reactors (25 rem whole body and 300 rem thyroid).

15.4.1.7 Dose Rates Near the Containment Building after DBA—Dose rates at points outside the reactor building were calculated using only those isotopic concentrations in the containment structure shown in Table 15.4-2. The calculations assume a uniform distribution within the containment atmosphere. The containment shell is

Fig. 15.4-1. Initial dose rate as a function of distance from the containment building, DBA.

Table 15.4-8. Dose rates outside containment due to radioactive atmosphere within, immediately following accident.

Isotope	Dose rate (R/hr)		
	3.05 m	30.5 m	305 m
^{133}Xe	0.0065	0.002	—
^{135}Xe	11.5	1.82	0.37
^{85}Kr	—	—	—
^{131}I	11.2	0.92	0.069
^{132}I	49.5	4.18	0.19
	172.2	14.2	0.41
^{133}I	22.7	2.1	0.16
^{134}I	98.0	8.74	0.25
	82.0	7.18	0.13
^{135}I	466.0	38.3	0.84
Total	913.0	77.5	2.5

assumed to offer 9.5 mm steel as the only shielding. The resulting initial dose rates are shown in Table 15.4-8.²² It should be recognized that other fission-product isotopes will be present in the containment structure. Their quantities and distributions depend on fission yield, half-life, and mode of dispersal. Dose rates from these other isotopes are expected to be only fractions of those reported here. Figure 15.4-1 shows a plot of initial dose rate vs distance from the containment structure.

REFERENCES

1. W. M. Breozzeale, T. E. Cole, and J. A. Cox, Preliminary Boiling Experiment in the LITR, Oak Ridge National Laboratory, Oak Ridge, Tenn., Rept. TID 5065 (1953).
2. LPTR Safeguard Report AECD-4258 with Addenda, I, II, III, and IV, Foster Wheeler Corp., LPTR Operations Staff (1956).
3. J. W. Bolstad, G. E. Cummings, J. L. Eagan, and H. F. Finn, Safety Analysis Report for the Livermore Pool Type Reactor, Lawrence Livermore Laboratory, Rept. UCRL-50147 (1966) p. 57.
4. H. T. Williams, O. W. Stopinski, L. Yarnell, A. R. Lyle, C. L. Werner, and H. L. Maine, "Section VI, Calculation of Fuel Temperatures in Loss-of-Coolant Accident," in 1969 Status Report on the Omega West Reactor, with Revised Safety Analysis, Los Alamos Scientific Laboratory, Los Alamos, N. Mex., Rept. LA-4192 (1969).
5. H. T. Williams et al., op. cit., p. VI-8.
6. H. T. Williams et al., op. cit., p. VI-9.
7. R. A. Lawrence, An Investigation of the Thermal Response of the ASTR to Loss of Coolant Accident, General Dynamics, Fort Worth Division, Rept. WL TR-64-137 (1965).
8. H. T. Williams et al., op. cit., p. VI-11.
9. H. T. Williams et al., op. cit., p. VI-13.
10. H. T. Williams et al., op. cit., p. VI-16.

11. F. T. Benford and C. C. Webster, The Low Intensity Testing Reactor—Safety Analysis, Oak Ridge National Laboratory, Oak Ridge, Tenn., Rept. TM-1924 (1968).
12. R. G. Wenzel and G. P. Arnold, Heat Loss from Fuel Elements Partially Immersed in Water, Los Alamos Scientific Laboratory, Los Alamos, N. Mex., Rept. LA-4473 (1970).
13. R. G. Wenzel and G. P. Arnold, op. cit., p. 5.
14. M. Knezevich et al., Loss of Water Experiment at the LPTR, Lawrence Livermore Laboratory, Rept. UCRL-7641 (1964).
15. A. Schwartz, Generalized Reactor Kinetics Code AIREK, Atomics International, Canoga Park, Calif., Repts. NAA-SR Memo 3509 and 4980 (1960).
16. R. W. Miller, A. Sola, and R. K. McCardell, SPERT I Destructive Test Program on an Aluminum Plate-Type Water Moderated Reactor, Phillips Petroleum Co., Idaho Falls, Idaho, Rept. IDO-16883 (1964).
17. T. J. Thompson and J. C. Beckerley, Nuclear Reaction Safety (Massachusetts Institute of Technology Press, Cambridge, Mass., 1964) vol 1.
18. USAEC Safety Guides 3 and 4 (1970).
19. I. Sax, Dangerous Properties of Industrial Materials (Reinhold Publishing Co., New York, 1957), pp.180-184.
20. W. E. Stocum, Los Alamos Scientific Laboratory, Los Alamos, N. Mex., private communication (March 15, 1968).
21. National Bureau of Standards Handbook 69, U.S. Dept. of Commerce, Washington, D.C., (U.S. Government Printing Office, 1959) Table 1.
22. T. Crites, Lawrence Livermore Laboratory, private communication (October 30, 1972).

CHAPTER 16.0 TECHNICAL SPECIFICATIONS, LPTR

16.1 Introduction

These technical specifications set forth operational and design limitations for the LPTR and establish certain administrative parameters relating to surveillance, staffing, and program review. These specifications derive from this Safety Analysis Report, USAEC Guides,^{1,2} and Appendix C, the LPTR Operating Limits (1966).

Where applicable, each specification is preceded by an objective or reason for establishing the specification and is succeeded by a base or reason for choosing the value assigned.

16.1.1 Definitions of Key Terms

- Instrument calibration – An adjustment of the instrument such that its output responds with acceptable accuracy over a given range to known values of the input parameter.
- Instrument check – A comparison of instrument readout with other instruments to ascertain that it is functioning and has a reasonable readout.
- Instrument test – A check of an instrument after a planned signal or event has been put to the instrument.
- Limiting safety system setting (LSSS) – Those limiting values of parameters for which protective action must be initiated. They are so chosen that automatic protective action will correct the most abnormal situation anticipated before a safety limit is exceeded.
- Limiting condition for operation (LCO) – Those limiting conditions placed on systems and parameters such that LSSS values are not intentionally exceeded and such that the status of certain nonoperating systems is established.
- Operable – A component or system is operable when it is capable of performing its intended function in a normal manner.
- Safety limit – A maximum upper bound placed on the actual value of a variable to protect the integrity of the principal physical barrier, fuel clad, preventing the release of radioactivity. Values greater than the safety limit might lead to damage of the fuel clad.
- Actual value – The true value as distinct from the observed value.
- Annually – At intervals of 14 months or less.
- Startup – The act of bringing the reactor from shutdown to final steady state operating power level.
- Scram – Process of inserting the neutron absorbing sections.
- Shutdown – The reactor is shut down when three of the four neutron absorbing sections are fully inserted.

- Operational startup rate reactivity – That reactivity capable of producing a 20-s period.

16.2 Safety Limits and Limiting Safety System Settings

16.2.1 Safety Limits

16.2.1.1 Objectives – This specification provides true values of power, flow, and temperature that if violated could reduce the integrity of the principle physical barrier (fuel clad) for core fission products under forced and natural convection modes of operation.

16.2.1.2 Specifications –

1) Natural convection.

The true value of power shall be below 0.5 MW whenever the primary pump is secured. The pool temperature shall be $\leq 51.6^{\circ}\text{C}$ (125°F). This establishes the safety limit for natural convection.

2) Forced convection cooling.

The maximum actual value for power shall be below 4.9 MW. The primary coolant flow is above $46 \text{ dm}^3/\text{s}$ (735 gpm) and coolant outlet temperature is below 58.3°C (137°F). (Minimum core size, Sec. 16.5.2).

3) Pool water height.

The pool water height shall be at least 6.5 m above tank bottom (or less than 1.5 m from top of reactor tank).

16.2.1.3 Bases –

1) The basis for the limit for natural convection cooling is Table 4.4-4.

2) The basis for safety limits for the forced convection cooling mode is the analysis presented in Sec. 4.4.6 and summarized in Table 4.4-4.

3) The value of 150 kPa at core (water level 6.5 m above tank bottom) is used to determine the saturation pressures for various temperatures in power and flow safety limits, Sec. 4.4.6.

16.2.2 Limiting Safety System Settings (LSSS)

16.2.2.1 Objectives – These LSSS are established to provide settings on safety and control systems monitoring reactor power, primary coolant flow, primary coolant outlet temperature, and pool height. A 2σ error analysis in measurable parameters is incorporated to assure that protective action is initiated before a safety limit can be exceeded.

16.2.2.2 Specifications -

- 1) For natural convection the maximum power level shall be 0.15 MW (150 kW) with $T_{(pool)} \leq 51.6^{\circ}\text{C}$ (125°F).
- 2) For forced convection the maximum power level shall be 4.5 MW, $54 \text{ dm}^3/\text{s}$ (850 gpm) or greater, and $T_{(outlet)} \leq 51.6^{\circ}\text{C}$ (125°F).
- 3) The pool height above tank floor shall be 7.2 m (0.9 m minimum from tank top).

16.2.2.3 Bases - The values listed in this specification are far enough below those of the safety limits so that the maximum expected adverse transient in any one of the variables would not cause a safety limit to be exceeded. The margin of error in determining these values derives from Sec. 4.4.6.2, Table 4.4-4.

16.3 Limiting Conditions for Operations

16.3.1 Reactivity

16.3.1.1 Objectives - The objective of the reactivity limitation is to insure that the reactor core a) can be made subcritical under adverse conditions, and b) cannot be put on a positive period less than that which produces a fast scram, 1 s.

16.3.1.2 Specifications -

- 1) A minimum of four shim rods shall be used and positioned such that the removal of any one rod by itself will not make the reactor critical.
- 2) Minimum ratio of shim bank $\frac{\Delta k}{k} \%$ to core excess $\frac{\Delta k}{k} \%$ shall be two.
- 3) Minimum shutdown margin (shim bank $\frac{\Delta k}{k} \%$ - core excess $\frac{\Delta k}{k} \%$) shall be the worth, in reactivity, of the most reactive fuel element at the core center.
- 4) No movable experiment combined with operation startup rate reactivity shall be allowed that would put the reactor on less than a 1-s asymptotic period.
- 5) The reactivity effect of any experimental equipment or apparatus located in or adjacent to the reactor core shall be strictly controlled by the Reactor Supervisor. Any portion of such equipment or apparatus for which one can postulate a rapid movement mechanism shall be limited such that its reactivity, combined with operational startup rate reactivity, will never reduce the asymptotic period to below 1 s.

16.3.1.3 Bases - The design features of the core as presented in Sec. 4.3.1 must be met. These limits insure protection against a potential startup accident as presented in Sec. 15.2.2, and at the same time permit a large envelope of safe operation. Sec. 15.2.2 describes the analysis of a reactivity type accident. Reactivity value corresponding to a 6 ms period could lead to core damage as evident in the SPERT series of tests. A 1-s minimum period due to the movement of an experiment and rods is a conservative limit.

16.3.2 Control and Safety Systems

16.3.2.1 Objectives – The following systems must be operational as specified to insure adequate monitoring of the reactor and its immediate environment and to ensure that safety features are preserved.

16.3.2.2 Specifications – The reactor will not be started up or will be scrammed as follows.

1) Nuclear instrumentation.

If any set point is greater than LSSS or circuits associated with the following are inoperative or abnormal so as to effect operation:

a. Period scrams.

- (1) Log count rate (only required for startup).
- (2) Log N (only required for startup).
- (3) Period safety slow scram (D amp.).
- (4) Period safety fast scram (D amp.).

b. Power level scrams.

- (1) Log N.
- (2) Linear level.
- (3) Level safety slow scram (2 of 3 required).
- (4) Level safety fast scram (2 of 3 required).
- (5) Log count rate (for startup).

2) Reactivity control system.

If any set point is greater than LSSS or system found in inoperative or abnormal condition so as to affect operation.

- a. Shim rod actuator (4 required).
- b. Shim rod assembly (4 required).
- c. Magnet amplifiers (4 required).

3) Primary coolant monitor system.

If any of the following systems is inoperative or abnormal or if set points are less conservative than LSSS values.

- a. Primary coolant system.
- b. Low flow scram.
- c. High temperature scram (outlet).

4) Radiation monitoring system.

a. If either of the following stations is inoperative:

- (1) Stack monitor.
- (2) Secondary coolant monitor.

b. If two or more of the following stations are inoperative:

- (1) Room monitor stations (5).
- (2) Fission product monitor.
- (3) Primary coolant monitor.
- (4) Duct monitor.
- (5) Reactor top monitor.

16.3.2.3 Bases -

1) Nuclear instrumentation.

The combination of log count rate channel (source range), linear level channel (intermediate and power range), and the log N channel (intermediate and power range) offer double coverage of all power levels (except for source range levels where the log-count-rate channel records the very low count rate). These channels in concert protect against excessive startup rates during approach to power and reduce the probability that power levels in excess of LSSS values, Sec. 16.2.2, can occur.

2) Reactivity controls.

The requirement that safety rod assemblies and actuator assemblies be operable ensures that all reactor startups, power changes, and shutdowns are within expected values.

3) Primary coolant system.

The primary system must be operable to insure that thermal hydraulic parameter, Sec. 4.4, stays within accepted values.

4) Radiation monitoring system.

- a. The stack monitoring system is required to insure that airborne radioactivity is not only monitored, but that the building seal system activates properly when required.
- b. The secondary coolant monitor is required to give warning of any major leakage from the primary to the secondary side of the heat exchanger.
- c. This minimum combination of remote area radiation monitors insures that the radiation levels in the building will be adequately monitored and that the building seal system will be activated automatically as necessary.

16.3.3 Rod Drop Parameters

16.3.3.1 Objectives - The objective of assigning rod drop times and holding currents is to insure that the rate of rod insertion can be sufficiently rapid to avoid potentially damaging overpower situations as described in the startup accident, Sec. 15.2.2.

16.3.3.2 Specifications -

- 1) The fast rod drop shall be completed within 650 ms of initial signal input.
- 2) The slow rod drop shall be completed within 750 ms of initial signal input.

16.3.3.3 Basis - The basis for this specification is directly related to control of a startup accident as described in Sec. 15.2. It also provides an additional test that the nuclear instrument systems, Sec. 7.2, perform within a specified time frame.

16.3.4 Engineered Safety Systems

16.3.4.1 Objectives - This specification provides a minimum condition or level of performance such that these systems will be available to control the consequences of an accident. Such systems are:

- Containment structure.
- Filter/Scrubber system.
- Emergency core cooling system.

16.3.4.2 Specifications - The reactor will not be started up or will be scrammed unless the following conditions are met:

- 1) Containment structure - The containment structure shall have no open penetrations except for the stack system, air conditioner system, and the airlock doors. Each of these systems shall be immediately sealable to achieve a building seal condition.
- 2) Filter/Scrubber system - At least one of these systems shall be capable of filtering particulates and the radioiodines. The HEPA filter and the activated charcoal filter shall be installed and operational.
- 3) Emergency core cooling system - This system shall be configured such that the manual or solenoid valve in the reactor room will supply water to the core.

16.3.4.3 Bases -

- 1) Containment structure - In an emergency and without forewarning, the containment structure shall be sealable in order to meet the requirements of Sec. 15.4, Design Basis Accident.
- 2) Filter/Scrubber system - Although each filter/scrubber system consists of an HEPA filter, an activated charcoal filter, and a KOH scrubber system, only the HEPA and the activated charcoal are required to remove particulates and iodines to the levels analyzed in Sec. 15.4 Design Basis Accident.
- 3) Emergency core cooling system - The emergency core cooling system offers added assurance that, under the LOCA conditions of Sec. 15.2, fuel element temperatures will not reach the softening point of aluminum.

16.3.5 Effluent Radioactivity Levels

16.3.5.1 Objectives - The objective of this specification is to insure that effluents released from the containment are closely controlled with respect to radioactivity. Since secondary coolant may be lost to the atmosphere by transpiration at the cooling tower or may be dumped into sewer systems its radioactivity requires specification.

16.3.5.2 Specifications -

- 1) Maximum stack effluent activity ($560 \mu\text{Ci}/\text{m}^3$ air).
- 2) Maximum secondary coolant activity ($3 \mu\text{Ci}/\text{m}^3$ water).

16.3.5.3 Basis - The basis for the limitation of gaseous activity is presented in Sec. 11.3. This specification takes into account atmospheric dispersion and meteorology for the site.³

16.3.6 Experiments

16.3.6.1 Objective - This specification prescribes the administrative control placed on experiments.

16.3.6.2 Specifications - No experiment will be run in a reactor facility unless approved by the reactor supervisory staff in accordance with those procedures shown in Sec. 13.5.

16.3.6.3 Basis - Approval authority for experiments must be clearly delegated so that all hazards associated with an experiment are reviewed and controlled by knowledgeable people, and so that specification in this document are met.

16.4 Surveillance Requirements

16.4.1 Reactivity Control

16.4.1.1 Objective - Reactivity levels as well as certain control rod parameters are checked periodically to ensure that nuclear parameters have not changed from expected values, Sec. 16.3.1. This specification establishes a minimum interval between those reactivity-related checks.

16.4.1.2 Specifications -

- 1) Scram tests - Weekly except for extended runs.
- 2) Core excess reactivity - Weekly except for extended runs.
- 3) Rod drop times - Alternate fast and slow with every other fuel cycle or major fuel movement.

- 4) Rod travel calibration - Annual.
- 5) Shim rod thickness gaging - Annually.
- 6) Single rod criteria - Change in core configuration.
- 7) Ratio of shim bank to core $\frac{\Delta k}{k}$ - Change in core configuration, or annually.
- 8) Worth of shim bank - Change in core configuration, or annually.
- 9) Shutdown margin - Change in core configuration.
- 10) Experimental worth $\frac{\Delta k}{k}$ - On initial insertion and after any changes to it thereafter.

16.4.1.3 Basis - These surveillance requirements generate a high degree of trust that the specifications in Sec. 16.3.1 are met.

16.4.2 Control and Safety Systems

16.4.2.1 Objectives - Minimum periodic surveillance intervals of nuclear instrumentation, rod control systems, auxiliary and radiation monitoring systems are specified here to minimize component failures during operation.

16.4.2.2 Specifications -

1) Nuclear instrumentation.

- a. Nuclear instruments, Sec. 16.3.2.2, shall be checked daily when the reactor is operating and during the first stable period following a power change.
- b. The nuclear instruments shall be tested weekly, or prior to the next startup in case of extended periods of operation.
- c. The nuclear instruments shall be calibrated annually. Calibration shall include analysis of electronic responses and, with exception of the log-count-rate system, power calibration in the reactor using the primary coolant heat balance. The log-count-rate system (source levels) shall be calibrated with known inputs. Verification of nuclear response in the reactor shall be accomplished by noting count rate changes for chamber movements after the gamma discrimination plateau is well established.

2) Rod control systems.

The shim and regulating rod actuators, rod extensions, and magnet assemblies shall be tested at least annually for mechanical and electrical integrity.

3) Auxiliary and primary coolant systems.

The auxiliary piping systems and primary coolant pumps, valves, and piping shall be visually inspected at least annually. Primary coolant temperature monitors shall be checked at least daily and shall be calibrated annually.

4) Radiation monitoring systems.

The radiation monitoring systems as specified in 16.3.2.3 shall be checked daily and shall be tested at least weekly. These systems shall be calibrated annually.

16.4.2.3 Hazards - The basis for surveillance functions of components or systems is influenced by historical performance, importance within the safety scheme, and susceptibility to false indication.

- 1) Nuclear instrument checks, tests and calibrations will insure that the instruments are operable.
- 2) Rod control system surveillance will reduce the probability that simple mechanical or electrical faults would prevent intended rod insertion when initiated by control and safety circuits.
- 3) Auxiliary and primary control system surveillance will insure that simple component failure will not produce an undesirable event during reactor operation.
- 4) Surveillance of the radiation monitoring equipment will reduce the probability that a radiation hazard could go undetected or that radioactive release to the environment could go unmonitored.

16.4.3 Engineered Safety Features

16.4.3.1 Objective - This specification establishes surveillance steps to be carried out at stated intervals for:

- Containment structure.
- Filter/scrubber system.
- Emergency core cooling system.

16.4.3.2 Specifications -

1) Containment structure

The building seal system shall be checked weekly for operability by tripping a different channel, Sec. 7.3.1, each week. The emergency air decontamination blower and air flow shall be verified as operable each week by operating the system.

In addition, the following tests will be made every three months but not to exceed four months.

- a. Reactor shutdowns associated with sealed doors.
- b. Containment building positive pressure sensor calibration.

The accessible portions of butterfly valves and operating lines, the air supply tank and check valving, and the isolation valve for the emergency air decontamination system shall be visually inspected annually.

2) Filter/Scrubber system.

At least one of these two systems shall be checked each week for:

- Airflow through system noting manometers.
- HEPA and charcoal bed filters present are checked.
- Valve positions and exterior manometer condition.
- Any signs of flow restriction or component deterioration.

3) Emergency core cooling system.

The emergency core cooling system shall be checked annually for operability both from the reactor room and the control room.

16.4.3.3 Basis -- Surveillance requirements for the engineered safety features are based on frequency of usage in connection with other safety related checks, probability and mode of component failure, and operational experiences. Although additional checks and surveillance steps are normally present as part of the operational check lists, this minimum frequency of inspection will establish reasonable assurance that these systems will do their assigned task when called on in an emergency.

16.5 Design Features

Important design features are specified in order to insure appropriate review, should these features require modification.

16.5.1 Reactor Fuel

The fuel element shall be of the MTR type consisting of curved fuel-containing plates as described in Sec. 4.2.1.

16.5.2 Reactor Core

Although the thermal hydraulic parameters were analyzed under the conservative conditions of a core consisting of 16 standard and 5 special elements, the reactor core will normally be operated in a configuration approximating that shown in Sec. 4.3.2.

16.5.3 Control and Safety Systems

The control and safety systems described in Ch. 7.0 shall represent a minimum of equipment necessary for safe operation of the reactor.

16.5.4. Radiation Monitoring System

A minimum radiation monitoring system is required within the reactor room environment to ensure that operators know where potential hazardous areas may exist. Secs. 12.1.4 and 12.2.4 describe such a system.

16.5.5 Fissionable Material Storage

Refer to Sec. 9.1.

16.5.6 Reactor Building

The reactor building shall be considered satisfactory containment as long as a) the building is readily sealable, and b) one of the two filter/scrubber systems (having at least the positive HEPA filter and the charcoal bed) is on line.

16.6 Administration

16.6.1 Reactor Administration

Functions and assignments of responsibility for reactor operations are described in Sec. 13.1.2. The Reactor Supervisor shall be responsible for all aspects of reactor operations. The reactor may be taken critical if, and only if, at least one Reactor Shift Supervisor and one Reactor Operator, who are qualified in accordance with Sec. 13.2, are in the Building 281 area. One of these individuals must be in attendance at the reactor console whenever the magnet power key is inserted into the console.

These individuals shall maintain:

- 1) Familiarity with systems and procedures.
- 2) Proficiency of operation by requalification as discussed in Sec. 13.2.2.

16.6.2 Review and Audit

16.6.2.1 Reactor Operations – Reactor operations shall be open for audit, Sec. 13.4.1. Reactor files and records shall be available on request to audit groups. The subjects to be audited are assigned by the Director's Office.

16.6.2.2 Reactor Safety Committee – A reactor safety committee composed of people from several disciplines shall exist, Sec. 13.4.2, to advise the Reactor Supervisor on operational matters that could effect safety. See Appendix B of Operational Safety Procedure 281, Appendix B of this SAR.

16.6.3 Incidents or Accidents

16.6.3.1 Evaluation of Occurrences – Appropriate corrective action shall be completed for abnormal occurrences to prevent or reduce the probability of recurrence. Abnormal occurrences falling into the following categories, however, shall be reported by the Reactor Supervisor to the Director's Office and the Programmatic Division Leaders.

- 1) Conditions that result in safety system settings less conservative than those listed in Sec. 16.2.2, or that result in violation of a limiting condition for operation established in Sec. 16.3.
- 2) Abnormal degradation of any of the boundaries which are designed to contain radioactive materials, or a release of radioactive materials that indicates potential operating problems or significantly interferes with operations, or any uncontrolled or unplanned release of radioactivity from the reactor building.
- 3) Any uncontrolled or unanticipated changes in reactivity that could significantly or adversely affect safety of operations.
- 4) Conditions of components that prevent or could prevent the intended overall performance of safety, control, or shutdown systems. This is not intended to include instances where such systems are checked regularly and a malfunction is found during such a check.

16.6.3.2 Reporting of Incidents or Accidents – The Reactor Supervisor shall report such an incident or accident immediately by telephone to the Director's Office and the Programmatic Division Leader. A written report must follow within 24 hours.

- 1) The verbal report shall include:
 - a. Incident or accident description.
 - b. Safety implications.
 - c. Actions taken to determine the cause of the incident or accident.
 - d. Corrective action proposed.

Based on this information, a determination will be made by the appropriate programmatic division and the Director's Office as to whether operations can be safely resumed.

- 2) The written report shall include:
 - a. All the information in the verbal report.
 - b. An analysis and evaluation of the safety implications.
 - c. Corrective action proposed to prevent similar occurrences involving similar components or systems.

16.6.4 Operating Procedures

Operational procedures, Sec. 13.5, are used as guides and provide a step-by-step check for certain detailed operations. They are:

- 1) Operation and Maintenance Procedures (OMP),
- 2) Operational Safety Procedures (OSP).

The conduct of reactor operations, policies and emergency actions are set forth in Operation Safety Procedure 281, Appendix B of this SAR.

16.6.5 Operational Records

Operating records shall be maintained in accordance with retention schedules discussed in Sec. 13.6.

REFERENCES

1. Regulatory Guide to Content of Technical Specifications for Nuclear Reactors (U.S. Atomic Energy Commission, Washington, D.C., 1968).
2. Technical Specifications for Research Reactors (American Nuclear Society, Hinsdale, Ill., 1973).
3. "Chapter 0524, Annex A, Note C" in U.S. Atomic Energy Commission Manual (U.S. Atomic Energy Commission, Washington, D.C., 1968).

CHAPTER 17.0 QUALITY ASSURANCE

17.1 Quality Assurance for Reactor Operations

The LPTR includes structures, systems, and components intended to either prevent accidents or to control and mitigate the consequences of accidents. The design and construction of these critical items are required to meet specified standards to assure that, when placed into operation, they will perform their function adequately and reliably.

The quality assurance (QA) process is part of the design, construction, and operation of any system or facility. The incorporation of a QA program does not change the scope of these activities but formalizes and documents the process. Once a facility reaches the routine operations phase it no longer needs to reiterate the complete QA process except for design changes associated with critical items, (17.1.2).

The requirements set forth in this chapter are designed to comply with those general criteria as specified in the LLL Quality Assurance Program,¹ and applicable QA program requirements for Nuclear Power Plants.² Available standards set forth in AEC guides may be used where applicable except where retrofitting may be required.

17.1.1 Organization

17.1.1.1 AEC—The AEC-SAN Office is represented by its Safety Division for all matters dealing with QA. This division acts as the main approval channel for reviewing and approving the LPTR QA Plan and monitoring the LPTR program performance by periodic documentation reviews and audit surveys. Two additional documents related to QA over which AEC-SAN exercises final review and approval are the current LPTR Safety Analysis Report and occurrence reports on events as described in Sec. 16.6.4.

17.1.1.2 Facility Supervisor—The individual responsible for overall application of QA standards to the LPTR is the Reactor Supervisor: refer to Fig. 13.1-1 for organizational structure. He implements the QA plan by assigning the specific

1. J. L. Olsen, Lawrence Livermore Laboratory Quality Assurance Program, Lawrence Livermore Laboratory, Internal Document, June 28, 1972. Readers outside the Laboratory who desire further information on LLL internal documents should address their inquiries to the Technical Information Department, Lawrence Livermore Laboratory, Livermore, California 94550.
2. Quality Assurance Program Requirements for Nuclear Power Plants, American National Standards Institute, ANSI N 45.2 (1971).

responsibilities represented in Fig. 17.1-1. The Reactor Supervisor provides, with assistance from the support groups, specific controlling documents including this Safety Analysis Report with its Technical Specification and the QA implementation documents. He also approves original issue of and revisions to the critical item specifications as originated by the engineering groups.

The Reactor Supervisor reviews and approves all new procurement nonconformance inspection reports and all incident occurrence reports involving critical items.

17.1.1.3 Engineering Departments—The primary engineering support for the facility consists of the Mechanical and Electronics Engineering Departments. The Plant Engineering Department provides engineering functions identical to those described for ME and EE, Sec. 13.1.2, for those critical items associated with buildings and facility support systems. These Engineering Groups are responsible for maintaining surveillance over appropriate segments of the QA Program. Specific assignment of functions is expanded on in Sec. 17.1.3 and 17.1.4.

17.1.1.4 Hazards Control—The Hazards Control staff serves in a safety advisory, surveillance, and review capacity, Sec. 13.1.2.

17.1.1.5 QA Review Group—The QA Review Group consists of those individuals in the above departments who are responsible for reviewing and advising on QA-related activities. These individuals represent a wide range of expertise. They a) conduct periodic reviews of the QA program, b) advise on significant changes to specifications, c) review reports that reflect significant nonconformance with specification requirements, and d) make appropriate recommendations. The QA Review Group may perform these functions either as a committee meeting together or as individuals for each specific engineering discipline.

17.1.2 Quality Assurance Program—

17.1.2.1 Critical Items Subject to Quality Assurance

- Fuel elements (Sec. 4.2.1)
- Primary coolant pressure boundary (Sec. 5.2)
- Containment system (Sec. 6.2)
- Reactivity control system (Sec. 4.2.3)
- Effluent monitoring systems (Sec. 11.4)
- Emergency core cooling system (Sec. 6.3)
- Nuclear safety instrumentation (Sec. 7.2)

— Administrative Supervision
- - - Technical Supervision

Functions in connection with QA and SAR

1. Approve SAR
2. Review and approve QA plan implementation
3. Review and approve occurrence reports, Sec. 16.6.4.1
4. QA plan implementation
5. Coordinate preparation of technical specification
6. Review and approve critical item specifications
7. Coordinate SAR preparation
8. Approve and submit SAR
9. Issue critical item occurrence reports with recommendations
10. Authorize internal audits and QA implementation effort
11. Supervise staff functions
12. Review staff activities and documentation
13. Assist preparation of technical specification
14. Critical item design and specification
15. Assist SAR preparation
16. Coordinate and inspect new procurement (programmatic systems)
17. Issue inspection reports (programmatic systems)
18. Assist preparation of facility specifications
19. Critical item specifications (Bldg. Services)
20. Coordinate and inspect new procurement (Bldg. Services)
21. Review specifications
22. Technical specifications review
23. Critical item specifications review and recommendation
24. Reviews and recommendations
25. Nonconformance inspection reports

2

Fig. 17.1-1. Functions and responsibilities of various groups at LLL in connection with QA and SAR of the LPTR.

17.1.2.2 Mechanisms for Achieving Operational Quality Assurance— The following key documents constitute the operational quality assurance mechanism.

- Engineering safety analyses (Sec. 17.1.3)
- Engineering documents (Sec. 17.1.4)
- Building procedure OSP 281 (Operations Manual, Sec. 13.5.2)
- Special operational safety procedures (Sec. 13.5.2)
- Operation and maintenance procedures (Sec. 13.5.1)
- Checklists—weekly, start up, and shutdown (Sec. 13.6.2)
- Surveys—weekly reactivity and weekly power surveys.
- Daily logs
- Maintenance card file (Sec. 13.6.2)
- Component instruction manuals and shop logs.

17.1.3 Design Control

Based on safety and design analyses, a determination of applicable regulatory requirements, codes and standards is made for the critical item. These applicable regulatory requirements, codes, and standards are incorporated into the design, drawings, procedures and instructions for the critical items. Appropriate quality standards are included in design documents. The Reactor Supervisor reviews and approves the quality standards before they are issued. Deviations from these standards require the Reactor Supervisor's approval.

The selection of materials, parts, equipment and processes for critical items is the responsibility of the cognizant design group. The QA Review Group reviews these selections and the Reactor Supervisor evaluates their recommendations.

The identification and control of design interfaces is the responsibility of the Engineering group having the primary role. This group will coordinate the job and arrange for the review, approval, release, distribution and revision of documents involving design interfaces.

The adequacy of a design is determined by verification or checking by individuals or groups other than those who performed the original design, but who may be from the same organization. Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, it must include suitable qualification testing of a prototype unit under the most adverse design conditions. The checking process includes a design review in accordance with Sect. 13.4.

Design changes, including field changes, are subject to design control measures commensurate with those applied to the original design. All design change recommendations require the Reactor Supervisor's review and approval. It is the Reactor Supervisor's responsibility to determine whether the proposed change is significant. If the change is judged to affect the original design criteria regarding safeguards characteristics,

the QA Review Group and the Reactor Safety Committee will provide review and recommendations.

Finally, no design changes or modifications in the Facility performance characteristics listed in Technical Specifications (Ch. 16.0) will be implemented until authorized by AEC-SAN. These changes may require supplemental safety analysis to substantiate a review and approval by cognizant AEC offices. Such changes will be integrated into this SAR.

17.1.4 Procurement Document Controls

Procurement documentation is initiated by the appropriate support engineering group and conforms to those requirements listed in Sec. 17.1.3. Applicable regulatory requirements, codes, standards, and other requirements that are necessary to assure adequate quality are included or referenced in the documents for procurement of material, equipment, and services. To the extent necessary, procurement documents require vendors or contractors to provide a quality assurance program consistent with the pertinent provisions of this QA Program. The procurement documents are prepared by the cognizant engineering group, reviewed by the QA Review Group and approved by the Reactor Supervisor before issuance.

17.1.5 Instructions, Procedures, and Drawings

Activities affecting quality shall be prescribed by appropriate instructions, procedures, or drawings. These documents must include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Adherence to instructions, procedures, and drawings that are required to assure adequate quality will be the responsibility of the Reactor Supervisor. Detailed steps related to QA may be delegated to the appropriate support group as listed in Sec. 17.1.1.3.

17.1.6 Document Control

Each engineering group controls the issuance of instructions, procedures, and drawings, including changes thereto, that prescribe activities affecting quality in its jurisdiction. Each submits these documents, including changes, to the Reactor Supervisor, who reviews for adequacy and approves for release. Each makes sure the documents are distributed to and used at the location where the work is performed.

17.1.7 Control of Purchased Material, Equipment and Services

Measures to ensure the conformance of purchased material, equipment and services to procurement documents include procurement source selection and inspection.

- Procurement source evaluation and selection is on the basis of historical quality performance data, source surveys, or source qualification programs. This is routinely handled by the LLL Procurement and Supply Department.

- Source inspection is performed when necessary to assure the required quality of an item. When the quality of the item can be verified by review of test reports, inspection upon receipt or other means, then the source inspection may be deleted.

17.1.8 Identification and Control of Materials, Parts and Components

Materials for the fabrication of critical items shall carry markings that can readily be used to identify them. The marking can be the purchase order number, the specification number, or the lot or heat number. When the materials are fabricated into parts, the parts shall be stamped or otherwise marked with their drawing number. The documentation provides material records explicitly traceable to the part drawing number.

Critical components will be identifiable by purchase order number, specification number or manufacturer's name and catalog number.

17.1.9 Control of Special Processes

Process specifications shall designate applicable codes, standards, and requirements to be used for steps in the manufacture, installation or operational tests of critical items. Examples are:

- a) Manufacturing and quality control (QC) steps and inspection verifications during fuel element and control rod fabrication.
- b) Welding and inspection of installed reactor component welds.
- c) Material and installation inspections for reactor structures.

These process specifications will be designated by the cognizant Engineering Support group and approved by the Reactor Supervisor.

17.1.10 Inspection

The QC steps in the QA process are outlined in the critical item procurement documentation. These steps are generally inspection points, carried out at specific points in the manufacture, testing or installation using specified equipment.

The results of QC inspection or testing are documented in inspection reports in compliance with requirements and are submitted for review and acceptance by the respective Engineering Support group and Reactor personnel.

Verification of compliance with the inspection steps shall be done, wherever possible, by groups other than those who performed the activity being inspected. Independent testing laboratories as well as the LLL Nondestructive Testing Group may be employed to make the necessary tests or inspections.

For installed critical items subject to maintenance a tag or log entry will be used to document the QC check immediately upon completion of inspection or testing.

Inspection tags or log entries will include, as a minimum, the unit's identification, inspection results, date, and the inspector's signature.

17.1.11 Test Control

Although test activity normally appears in the critical item specifications, which generally are a part of design or procurement documentation, it may be documented in separate Critical Test Specifications. These test specifications may represent the primary controlling document for the conditions, frequency, results, etc., for the critical item test. New procurement documents may be supplemented by test specifications for critical items requiring special source, receiving or installation testing.

Test reports will be submitted to the engineering support group covering significant test results. The basis on which a test report will be accepted will be determined by compliance with test specifications. Systems or components that fail to meet specifications will be reworked and retested or rejected unless the engineering group recommends accepting it on a nonconformance basis. Test reports recommending such action will be reviewed by the QA Review Group and the Reactor Supervisor as indicated in paragraphs 17.1.1.2 and 17.1.1.5.

17.1.12 Control of Measuring and Test Equipment

Measuring and testing devices used in activities affecting quality shall be properly controlled, calibrated, and adjusted at specified periods to maintain accuracy within necessary limits.

17.1.12.1 Mechanical Measurements—Mechanical measurement apparatus used in the manufacture of fuel elements and control rods requires certification by the manufacturer's QA division.

Mechanical measurement apparatus at the LPTR will be calibrated to high precision by periodic checking or equivalent, using LLL Metrology Section equipment or personnel.

17.1.12.2 Electronic Measurements—Electronics Shop equipment used at the LPTR will be calibrated and logged by the LLL Electronics Instrument Shop. Calibration will be indicated by appropriate labels placed on the equipment. Electronic components associated with reactor instrumentation will be bench tested in accordance with technical manuals using calibrated electronics equipment and will be logged into LPTR electronics shop logs.

17.1.13 Handling, Storage, and Shipping

Fuel elements constitute the major critical component requiring special handling, storage, and shipping. Fuel elements are stored in the central vault under custody

of Materials Management. They are shipped in accordance with Materials Management procedures and LPTR requirements.

Other components such as electronic assemblies will be retained either in their installed configuration or located in storage racks within the appropriate shop or reactor area.

Procurement documentation for material and equipment requiring special handling, cleaning, preservation, shipping and storage shall include notes detailing the special requirements. Receival inspection must verify that the instructions were followed and that no damage or deterioration has occurred.

17.1.14 Inspection, Test, and Operating Status

Work related to a critical system that could be in a hazardous configuration, or that interfaces with other critical systems that have hazardous configurations will be governed by operational procedures either as an Operation and Maintenance Procedure (OMP) or as an Operational Safety Procedure (OSP).

The continual routine maintenance of the LPTR is governed by operating and maintenance procedures or checklists when applicable, Sec. 13.5.

Appropriate log entries into the reactor operations daily log, or applicable stepwise procedure, will be made for each critical operating system and associated components. These logs will include failure, repair, and replacement history, etc. A component out of service shall be tagged and have a mating tag placed at its control station showing appropriate conditions. In addition, it may be locked out as necessary to prevent it from jeopardizing the reliable and safe operation of its associated system.

Before placing a new or modified critical system in operation, the cognizant engineering groups must furnish the Reactor Supervisor evidence indicating the test and inspection status of the system. Operation of the new or modified system requires the approval of the Reactor Supervisor.

17.1.15 Nonconforming Materials, Parts, or Components

Items not conforming to specifications may be rejected outright if warranted. However, if the cognizant Support Engineering Group recommends acceptance, the Reactor Supervisor (17.1.1.2) and the QA Review Group (17.1.1.5) will review the proposal for acceptance.

Nonconforming items that are rejected will be appropriately marked and returned to the vendor or to salvage. Nonconforming items that are accepted must be so marked on their control documentation.

17.1.16 Corrective Action

Measures to correct weaknesses in the quality of critical components and equipment are a continuing function of the support and operations staff. High component failure rates will be reviewed and analyzed to identify and correct the faults.

17.1.17 Quality Assurance Records

Completed Operation and Maintenance Procedures, Operation logs, and Checklists are kept on file. Retention periods for each type of record are set forth in Sec. 13.6.

The component and system QA documentation is the responsibility of the cognizant engineering group. Documentation such as occurrence reports, nonconformance reports etc., however, are kept in the LPTR Administrative Office. Other documents such as plans, drawings and technical manuals may be kept in the engineering files separate from the LPTR Administrative Office. An index code divides the reactor complex into a number of major categories, systems, subsystems, and components. The QA records are filed in this system and retrieval is easily accomplished by starting with the system title.

17.1.18 Audits

Internal audits are initiated by the Directors Office, Sec. 13.4.1.

Appendix A.
[Signature]
Environmental and Earth
Sciences

LAWRENCE LIVERMORE LABORATORY
University of California/Livermore, California/94550

UCRL-51592

**A GEOLOGICAL AND SEISMOLOGICAL INVESTIGATION OF
THE LAWRENCE LIVERMORE LABORATORY SITE**

L. H. Wight

MS. date: June 3, 1974

film as Appendix A to UCRL-51423
TWC

Satellite Photo of San Francisco Bay Area

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A GEOLOGICAL AND SEISMOLOGICAL INVESTIGATION OF THE LAWRENCE LIVERMORE LABORATORY SITE

Abstract

In response to an AEC request for Safety Analysis Reports for all critical structures on the Lawrence Livermore Laboratory site, we examine the seismic hazard at LLL. We review the geology and seismology of the Livermore Valley and surrounding regions and identify all the active faults in the region based on

seismic history, geomorphology and geodetic data. After carefully estimating the accelerations that the site would experience from maximum magnitude earthquakes on each active fault, we conclude that the Tesla fault has the most potential for doing damage at the LLL site. The response spectrum is derived.

Introduction

The AEC has requested that LLL prepare, in a format consistent with that specified in Ref. 1, safety analyses for certain critical structures.* One part of these analyses deals with the definition of the seismic hazard in terms of the geology and seismology. This report will define the seismic hazard at LLL in a manner consistent with the AEC specifications in Section 2.5 of Ref. 1. This hazard will be defined in terms of a maximum possible earthquake that the site might experience. This maximum possible earthquake will be derived from a current interpretation of the geology and seismology; we will not discuss the probability of such an earthquake occurring.

Two important reports have preceded this. Bernreuter and Tokarz,² on the basis of the geology known at that time, estimated the earthquake hazard at LLL. Subsequently, John A. Blume and Associates,³ performed a comprehensive geological survey of the LLL site area and on this basis discussed the seismic hazard. We will consolidate the information in these two reports and will reinterpret the hazard in terms of the most current theories and data.

Our analysis shows that the seismic hazard at the site is controlled by earthquakes on the nearby Tesla fault. The peak acceleration at the site from a maximum magnitude earthquake on this fault will be taken as 0.5 g. The acceleration response spectrum associated with the waveshape is given in Fig. 1.

*Those structures whose integrity is essential to the health and safety of the public.

We shall begin by reviewing the provincial geology and seismology and their relation to the Livermore Valley. We shall then discuss the geology and structure of the valley, and specifically, the site. We shall discuss the geologic status of nearby faults and shall derive the maximum magnitude earthquake that each of these faults might produce. We shall then examine the response of the site to earthquakes on each of these faults.

Fig. 1. Response spectrum of maximum magnitude earthquake (SSE) on the Tesla fault.

Provincial Geology

The Livermore Valley is physiographically in the California Coast Range Province. Because the origin and nature of the valley and the province are so closely related, we will first review the provincial geology and its history.

The Coast Range Province is a longitudinal geologic area (Fig. 2) easily distinguished from neighboring provinces by its unique structure and tectonics. Topographically, the region is marked by a system of parallel mountain ranges and valleys all trending NNW by SSE. This relief reflects the general state of stress that the province is experiencing, for most active and inactive faults align with this topography (Fig. 3). While many of the faults shown in Fig. 3 are relatively small and are more related to the local geology, there are three faults which have been fundamental in the history of the province: the San

Andreas, the Sur-Nacimiento, and the Coast Range Thrust (which is now represented by the Tesla-Ortogonalita and Stoney Creek faults).⁴ These faults delineate two completely different lithologic blocks. The Salinian Block⁵ lies between the Sur-Nacimiento and the San Andreas and is comprised of granitic-metamorphic rocks. The other block lies between the San Andreas and the Coast Range Thrust faults, and is composed of eugeosynclinal Franciscan rocks. The juxtaposition of these two distinct basement rocks has presented structural and chronological problems which have still not been completely resolved.

Many of these problems arise from the complexity of the Franciscan assemblage and its unusual contact with surrounding formations. The assemblage is a melange⁶ of graywacke, siltstone,

shale, chert, limestone, and mafic volcanic rocks that have been so deformed as to resemble a megabreccia.⁷ Yet, abutting the

Franciscans on the east is the contemporaneous miogeosynclinal Great Valley sequence which is ordered and conformable.

Fig. 2. Relief map of California showing the natural provinces (from Ref. 5).

In spite of this intense deformation in the Franciscan, it is possible to recognize some large-scale structure

to the assemblage. Most interesting for our purposes is the Diablo Antiform. This anticlinal structure trends

Fig. 3. Structure of the Coast Range Province (from Ref. 5).

N35°W from Parkfield to at least Clear Lake, a distance of 400 km.⁸ The anti-form is not continuous, but has been broken by erosion and local structural variations. The Livermore Valley is one such structural depression.

GEOLOGIC HISTORY

There are many theories regarding the evolution of the California landscape and, specifically, the Diablo Antiform. Most appealing, and currently most

popular, are those theories which describe the evolution in terms of plate tectonics.^{7, 9-12} These theories differ in detail, but all assume that the Franciscan assemblage formed during Jurassic time in an ocean trench at the American plate margin [Fig. 4(a)]. At this time, most of California was covered by a shallow inland sea. The sediments which collected on this sea floor formed the beginning of the Great Valley sequence. Relatively rapid sea floor spreading,^{7,10,11} or the collision of the American

Fig. 4. Northern California Plate Tectonics.

plate with an ancestral Salinian block,¹² thrust the cooler trench material deeper into the mantle, causing it to be pervasively sheared and locally metamorphosed during the late Jurassic (Nevadan Orogeny). By mid-Cretaceous, the spreading rates had slowed and the lighter, highly deformed Franciscan rocks were buoyed to the surface [Fig. 4(b)]. This and subsequent east-west compression from sea floor spreading thrust the Great Valley sequence over the melange of Franciscan rocks [Fig. 4(c)]. Thus emerged the primeval Diablo Antiform and the Coast Range Thrust fault. Localized uplifting of the Franciscan and associated displacement on the Coast Range Thrust fault probably continued at least through Eocene time⁵ while, perhaps episodically, the direction of spreading changed from east-west to northeast-southwest. This latter mode of plate interaction expressed itself in the formation of an ancestral San Andreas fault in the early Tertiary.^{9,13} By Oligocene time, the trench had completely consumed the Farallon plate, and had intersected the eastward-moving Pacific Rise [Fig. 4(d)]. It seems fairly certain that it was this intersection which activated the present San Andreas fault system. The large strike slip displacements of 500 km^{4,7,12} that have occurred on the San Andreas and its ancestor have served to relieve the compressive stresses within the Franciscan and across the Coast Range Thrust, and instead have put the region in a state of shear. This in turn activated strike slip motion on the Coast Thrust fault and its branches.

Uplifting of the Diablo Antiform and surrounding structures continued, some-

times dramatically, during the Miocene and Pliocene times. The inland sea still covered most of the area, although it began to drain by mid-Pliocene due to regional uplifting. By then, thrusting of the Great Valley sequence over the Franciscan formation had been replaced by localized strike slip on new faults that truncated the older Coast Range Thrust fault.

Drainage of the inland sea was completed by late Pliocene. This was followed by a resurgence of regional uplift and folding. This most recent orogeny resulted in the landscape as we know it today.

PROVINCIAL STRUCTURE

As Fig. 3 illustrates, the structure of the Coast Range Province is exceedingly complex. However, for our purposes, the only significant structural features are the Diablo Antiform and the three major provincial fault systems:

- The San Andreas fault
- The Sur-Nacimiento fault
- The Coast Range Thrust and its second generations (the Tesla-Ortiguera and the Stony Creek faults)

The Sur-Nacimiento, although important in the geologic history, is presently inactive⁵ and as a geologic hazard need not be considered further. The other fault systems, however, are presently active and represent a potential seismic hazard to the Livermore Valley area. In the next sections we will discuss some of their more important features.

The San Andreas System

The San Andreas fault and its associated branches represent the best known and most spectacular geologic feature in California. The fault extends over 1340 km from near the Mexican border to the Klamath Mountains. Twice in the last 116 years the fault has been the site of major earthquakes; the Fort Tejon earthquake of 1857 in the south, and the San Francisco earthquake of 1906 in the north. In both of these cases, surface faulting occurred for over 320 km along the strike of the fault with offsets measured in many meters. Yet, other parts of the fault respond in quite a different way. For example, the central portion apparently responds to tectonic forces by gradual deformation; creep rates of 1-3 cm/yr have been measured at Hollister.^{14,15} In any case, whether by creep or sudden slip, the total horizontal displacement along the fault has been an impressive 500 km since Jurassic time.⁵

The fault zone topography is classic. Extensive Quaternary activity has developed a rift valley along the entire length of the fault. Within the rift valley, which varies in width between hundreds and thousands of feet, is a zone of brecciated rocks. The fault cannot be traced as a single fault through this zone; surface rupture usually occurs discontinuously across en echelon faults. Frequently, as a result of local stress concentrations, an en echelon fault will branch off from the fault zone to form a subsidiary fault. The tectonic stresses around Hollister are apparently particularly conducive to branching, for in this region several significant branch faults leave the San Andreas. Two of the most

significant branches are the Hayward fault and the Calaveras fault.

Both of these faults are among the longest in the state; they branch from each other and the San Andreas around Hollister, and both disappear beneath alluvial cover in the Sacramento delta region. Both faults have been the site of large earthquakes (Tables 1 and 2), and both are presently readjusting to stress by creeping. Both faults have historical significance, for block faulting between and beside these two faults has given rise to the impressive horst structure of Sunol Ridge and to the depressions of San Francisco Bay and the Livermore Valley.

The Coast Range Thrust

As we emphasized in discussing the history of the province, the Coast Range Thrust was active only between Jurassic and early Tertiary time. At that time the fault was the most significant structural feature in California; the fault length was at least 1000 km, and thrust displacements across the fault were measured in miles.⁵ During and after early Tertiary time the thrust fault surface was warped, folded, and sheared, and consequently developed into a complex en echelon fault system. North of the Sacramento River this system is known as the Stony Creek fault system and south of the Livermore Valley it is known as the Tesla-Ortigalita fault system.¹⁶ There is no surface expression of the system across the Sacramento Delta region; it is presumably buried under alluvial cover.

In general, both the Tesla-Ortigalita and the Stony Creek faults are relatively quiescent compared to the San Andreas.

Table 1. Magnitude 4.0 or greater earthquakes on the Hayward Fault.

Date	Magnitude/Intensity	Location
10/21/1868	X	Hayward
12/4/1887	VI	Hayward
7/31/1889	VII	San Leandro
6/11/1903	VIII	Hayward
8/3/1903	VIII	Fremont
7/1/1911	6.6	W. of Mt. Hamilton
10/8/1915	VIII	Piedmont
10/26/1917	VI	W. of Mt. Hamilton
3/8/1937	4.5	Berkeley
12/29/1942	4.3	San Leandro
6/10/1949	4.6	W. of Mt. Hamilton
6/22/1949	4.1	W. of Mt. Hamilton
10/13/1952	4.2	Hayward
12/17/1954	4.5	San Leandro
10/31/1958	4.2	N. W. of Mt. Hamilton

Table 2. Magnitude 4.0 or greater earthquakes on the Calaveras fault.

Date	Magnitude/Intensity	Location
3/31/1898	VIII	Mare Island
7/6/1899	VII	Pleasanton
10/26/1943	4.9	N. W. of Mt. Hamilton
3/9/1949	5.2	S. W. of Mt. Hamilton
3/14/1949	4.4	S. W. of Mt. Hamilton
9/5/1955	5.8	W. of Mt. Hamilton
10/21/1955	5.4	Walnut Creek - Concord
5/31/1958	4.1	Concord

It is possible that only certain segments of the system are still active, but hypocentral data are inadequate to establish this. Seismicity data indicate that the Tesla-Ortigalita system is much more seismic than the Stoney Creek fault. This may be a result of the former's proximity to the San Andreas. Table 3 gives the locations of magnitude 4 and greater earthquakes which have occurred

since 1934 along these faults. These data were the basis for the detailed structure in the California Seismic Risk Map (Fig. 5) southeast of Livermore.¹⁷

The Diablo Antiform

We emphasize again that generally the basement rocks between the San Andreas and the ancient Coast Range thrust are Franciscan, and that the Diablo Antiform

is a broadly folded and eroded topographical high of highly deformed Franciscan rocks which is breached at its middle by the Livermore Valley. The antiform is dissected by many minor faults, some of

which run out into the Livermore Valley. These will be discussed in detail later. The Antiform (particularly that part south of the Livermore Valley) is exceedingly seismic; literally hundreds of magnitude 4

Fig. 5. California seismic risk map (from Ref. 17).

Table 3. Magnitude 4.0 or greater earthquakes occurring along the western edge of the Great Valley since 1934. (Ordered from south to north.)

Date	Magnitude/Intensity	Location
2/25/1947	4.2	N. of Coalinga
10/22/1955	4.2	N. of Coalinga
8/13/1940	4.0	N. of Coalinga
11/30/1943	4.0	N. of Coalinga
11/27/1958	4.3	W. of Ortigalita Peak
4/22/1932	4.0	E. of Ortigalita Peak
9/18/1941	4.0	N. of Ortigalita Peak
12/1/1938	4.5	N. of Ortigalita Peak
5/2/1946	4.6	E. of Livermore
4/15/1943	4.0	Near Livermore
4/15/1943	4.0	Near Livermore
4/15/1943	4.1	Near Livermore
4/21/1943	4.2	Near Livermore
9/10/1965	4.9	Antioch
11/28/1949	4.0	N. W. of Winters

and 5 earthquakes have been located by the University of California Seismic Station at Berkeley within the antiform in the recent years.

In the preceding sections we have tried to summarize the provincial geology and

structure as it relates to the Livermore Valley. We have tried to emphasize that structure local to the Livermore Valley often has regional significance. In the next sections we shall discuss the detailed local structure and geology of the valley.

Livermore Valley Geology

PHYSIOGRAPHY

The Livermore Valley roughly bisects the Diablo Antiform as an east-west synclinal depression (Fig. 6). To the south sits the Diablo Range, the most prominent unit of the antiform. The range extends broken but unbreached over 90 miles south to Panoche Valley. Elevations of the range average 1000 meters, but rise to 1289 meters near Mt. Hamilton. A single mountain, Mount Diablo, and its

foothills dominate the northern horizon. The mountain (elevation 1173 meters) is a piercement of Franciscan rocks that was very recently thrust up through an anticline of Cretaceous rocks.^{18,19} This anticline, an impressive structure by itself, trends SSE from the mountain and intersects the Altamont anticline to the northeast of the valley. These anticlines are the northeastern and eastern boundaries of the valley. The valley opens slightly to the west before intersecting a

north-south trending valley; to the north this valley is known as the San Ramon Valley, and to the south as the Sunol Valley. At the western edge of these valleys is the Calaveras fault, described earlier.

STRATIGRAPHY

A generalized stratigraphic column for the valley is given in Fig. 7. The section was derived by Huey²⁰ for the southeastern corner of the valley, but

Fig. 6. Satellite photo of Livermore Valley.

Fig. 7. Stratigraphic column for Livermore Valley (from Ref. 20).

well logs indicate that it generally represents the stratigraphy elsewhere.²¹

Basement rocks are everywhere the Franciscan assemblage. These rocks are exposed throughout the entire Diablo Range in the south and at Mt. Diablo in the north, but are covered by thousands of feet of alluvium on the valley floor. These are the oldest rocks in the area and, as we discussed earlier, are thought to have their origin as sedimentary deposits in an ancient offshore trench. The bulk of the formation consists of broadly deformed and folded sandstones, shales and cherts; although diabase, pillow basalts, serpentine and peridotite have been intruded locally.

Overlying the Franciscans are various sedimentary layers. Directly overlying the Franciscan is the Panoche formation, a major unit of upper Cretaceous age. It is widely exposed in the Altamont anticline to the northeast of the valley. These rocks, which are well cemented and indurated concretionary sandstones and shales, may have had their source in the ancestral Sierra Nevada.²⁰

The next major unit is the Tesla formation. It is exposed as a narrow band along Tesla Road, and its extension under the valley has been inferred from deep well drilling. The formation was mined at the old town of Tesla for coal, sand and clay until 1940 in a unique, although barely profitable, venture; all three products were used to their fullest extent by using the coal to fire the kilns for production of clay pottery and fine-grained bricks. Many fossils have been found in the formation, which have accurately dated the formation as middle Eocene.

Overlying the Tesla formation is the Cierbo formation, a relatively thin layer of poorly consolidated white buff sands that has proved to be a minor producer of oil. It is poorly exposed and has been delineated mostly by deep well drilling.²⁰ The surface exposures that do exist have been marked and dated by the occurrence of large fossil oysters of upper Miocene age.

Overlying the Cierbo is another upper Miocene unit, the Neroly formation. The Neroly outcrops widely southeast of the Livermore Valley, and the unit has been followed under the valley by deep well drilling. It consists mostly of bluish sandstones, andesitic conglomerates and clays.

Capping the entire sequence is a thick layer of Plio-Pleistocene clays, gravels, and sands known as the Livermore formation. The source for this formation was the Diablo Range.

HYDROLOGY

The hydrology of the Livermore Valley has been the subject of intense investigation since the turn of the century when the city of San Francisco started exporting water from the Pleasanton area. The high extraction rate steadily lowered the water table and precipitated construction of the Hetch Hetchy Aqueduct in 1934 and the South Bay Aqueduct in 1962. Out of concern for water quality, in 1962 local agencies launched a coordinated program of water table management and quality surveillance. Water quality remains a problem, but as a result of injection of large volumes of water, the water table is gradually returning to its turn of the century position.

The water-bearing rocks are in the upper strata of the Livermore formation; deeper strata are so impervious as to be insignificant producers of water. The depth to the present water table is highly variable, and in many cases these variations can be used to delineate fault zones.²²⁻²⁴ Some of this evidence will be discussed later.

GEOLOGIC HISTORY

The Livermore Basin began to form during late Pliocene time by folding and block thrusting between the Calaveras and other faults, on the western boundary, and a complex system of faults in the east.* Drainage off the surrounding hills deposited Cretaceous and Jurassic debris on the basin floor, which was then swampy and marsh-covered. This deposition

*We shall discuss these faults in detail in a following section.

marked the beginning of the Livermore series; the shales and sandstones underlying the marsh bed resulted in the Neroly series. At that time drainage from the valley was north through the San Ramon Valley. Judging from detailed stratigraphy of the Livermore formation this drainage was apparently blocked episodically, either as the result of landsliding or block faulting. When the drainage was open thick layers of gravel were deposited, and when blocked, the basin filled and covered the gravels with silts and sands which later consolidated into clay. We can reconstruct at least four such episodes from deep borehole data. By mid-Pleistocene the Sunol Valley had been block-faulted downward, and by late-Pleistocene the basin drainage had been captured in south-flowing Alameda Creek. The drainage became much more efficient as a result of regional uplifting and subsequent erosion and deposition of alluvial fill gave rise to the present landscape.

Site Geology And Structure

In the last section we considered the general aspects of the valley geology. Here we shall discuss the detailed structure and geology of the eastern edge of the valley as it applies to the earthquake hazard at the site.

SITE GEOLOGY

The site is at the eastern edge of the Livermore Valley near the western limb of the Altamont anticline. As Fig. 8 illustrates, the geology in this area is particularly diverse. The oldest rocks

in the area, the Franciscan, outcrop in the Diablo Range south of the site. These are overlain by successively younger strata toward the site. The Altamont anticline to the east is largely Cretaceous and Tertiary, while the site is on Quaternary alluvium.

SITE STRUCTURE

The geology indicates the structural complexity of the area. This is best illustrated with Fig. 9 which delineates most of the known faults in the Bay Area.

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The faults can be conveniently grouped in terms of their proximity to the site; those in the western part of the valley, in the central part and in the eastern part.

In the west is the Calaveras fault; the properties and seismicity of this fault were discussed earlier.

Crossing the center of the valley are the Livermore and the Mocho faults.

Fig. 9. (a) Faults in Bay Area.

wells.^{20,23,24} It is the northernmost segment of the Tesla-Ortogonal fault system. This system, composed of many faults of varying ages, runs along the eastern flank of the Diablo Range from the Livermore Valley to Panoche Valley.¹⁶ The northernmost extent of the fault is not known, but it certainly crosses the Livermore Valley, and may connect with the Stony Creek Fault north of the Delta region.^{4,5}

Ramp Thrust Fault

Evidence concerning this fault is scanty, but conclusive. Deep well drilling has revealed displacements on this fault, and there is evidence for surface faulting along its trace.³ The fault is certainly active although, as far as can be determined, a minor structure. There is no evidence that this fault extends under the site.³

Corral Hollow Fault

This fault branches from the Tesla fault 16 km southeast of the site. It has been located by infrared imagery, by gravity surveys, magnetic anomalies, deep drilling, and surface morphology.^{3,20} From the nature of the gravity survey, it has been concluded that the fault does not extend to the west beyond Greenville Road. It does not appear to be an active fault.

Doutherty Fault

This is another fault that was located by the comprehensive effort of J. A. Blume and Associates.³ It was located by infrared imagery, gravimetric and magnetic surveys, geodetic and water table data, and seismic refraction experiments.

The gravity data indicate that the block north of the fault is displaced upward relative to the south block. All evidence indicates that the fault is a minor structure which could be significant only in terms of surface faulting.

Carnegie Fault

This fault has been carefully delineated because of its proximity to the Livermore syncline and the oil production. Huey²⁰ mapped it as a high-angle thrust fault on the basis of surface morphology. It was subsequently better mapped by strong magnetic anomalies, deep well drilling and gravimetric anomalies.³ Displacement locally on the fault is in the same sense as on the Doutherty. This may represent sympathetic faulting on the relatively minor Doutherty fault. To the southeast, the fault truncates the Tesla fault where the motion is strike slip. This fault may be responding to the same tectonic stresses as the San Andreas system.¹⁶

Patterson Pass Fault

This fault, which was first recognized by Huey, has many characteristics in common with the Greenville fault. The two faults apparently merge to the northwest, although the Patterson Pass fault appears more like a branch of the Tesla fault.²⁰

Greenville-Riggs Canyon Fault

This fault was considered by Clark¹⁹ and Vickery²⁵ to be a major structural feature of the area. The connection between the Greenville fault on the south and the Riggs Canyon fault on the west flank of Mt. Diablo is not conclusive, but

One of the most impressive structural features of the valley is a buried syncline which plunges under the site from the southeast. It has been well defined by deep well drilling and seismic refraction, and evidence indicates that downfolding on the syncline is sympathetic to two flanking faults, the Carnegie and Tesla faults. Figure 10 is from Huey (Ref. 3, Appendix A) and illustrates the

relation of the syncline to surrounding features.

SEISMICITY

There is a great deal of evidence which indicates that the tectonic forces that induced this geologic and structural complexity are still operating on the valley today. We have already discussed the activity on the Calaveras fault; the other faults in the valley are equally seismic, at least in terms of earthquake frequency. Since 1900, over 120 seismic events have been located by the U. C. Seismic Station in the central and eastern reaches of the valley. Most of the events were less than magnitude 3.5, although there have been several magnitude 4.0 to 4.6 events (Table 3). In addition, there was a very large earthquake in the valley on July 4, 1861. Its exact epicenter is very uncertain, but it appeared to be most severe in Livermore (Intensity IX).^{*} This is a very important piece of evidence for it shows that the faults immediately around Livermore could support magnitude 6-7 earthquakes. Even the microearthquake activity provides interesting evidence, for it shows that the entire valley is readjusting to tectonic stresses. Geodetic level data^{3,26} has shown that the readjustment is lowering the valley floor and uplifting the surrounding hills. Geodetic data for horizontal creep is sparse,²⁶ but indicate that, at least at the western edge of the valley, there is a NW-SE shear stress acting on the valley. This stress is presumably related to the

regional NW-SE shear that the province is experiencing.

STATUS OF FAULTING AROUND THE SITE

We would like to define here the active/inactive status of each fault. There are several criteria available for defining this status. Because we are particularly interested in defining the earthquake hazard for AEC critical structures, we will follow the AEC guidelines from 10CFR, Part 100, Appendix A.²⁷ These guidelines tell us that a fault should be considered active if:

1. There has been movement at or near the ground surface at least once in the past 35,000 years or more than one in the past 500,000 years. In the absence of data permitting absolute dating, faults with recent enough movement to leave perceptible evidence of surface rupture, surface warping, or offset of geomorphic features are considered active faults.
2. There is instrumentally well-determined macroseismicity for a fault located in the continental United States west of the Rocky Mountain Front, or in Alaska, Hawaii, or Puerto Rico.
3. Or there is a relationship to an active fault according to characteristics (1) or (2) such that movement on one could be reasonably expected to be accompanied by movement on the other.

Clearly, several of the faults near the site must be classified as active by any standard; we include here the San Andreas, the Hayward, and the Calaveras faults.

^{*}The intensity was derived by the U. C. Seismic Station.

Table 4. Faults considered active by AEC Guidelines.

Fault	Fault length (km)	Rupture length (km)
San Andreas	1340	670
Hayward	250	125
Calaveras	250	125
Livermore	34	17
Mocho	34	17
Tesla	34	17
Ramp Thrust	17	8.5
Corral Hollow	25	12.5
Doutherty	8.5	4.25
Carnegie	25	12.5
Patterson Pass	13.5	6.75
Greenville - Riggs Canyon	42	21

Further, the geomorphic evidence discussed earlier classifies the Livermore, Greenville-Riggs Canyon, and second strand of the Tesla faults as active by criterion 1. There remain, however, several faults whose status cannot be defined by criterion 1:

- Mocho fault
- Ramp Thrust fault
- Corral Hollow fault
- Doutherty fault
- Carnegie fault
- Patterson Pass fault

The most recent movement on each of these faults cannot be dated with enough confidence to be declared active. Furthermore, there is no apparent geomorphic evidence to indicate that these faults are active. On the other hand, because the errors in locating epicenters is greater than the spacing between these faults, we

cannot attribute all the activity to just a few faults and assume that the rest of the faults are quiescent.

And in addition, geodetic evidence indicates that the depression and shearing of the entire valley is continuing today and that, therefore, there must be differential stresses across each of these faults. We can, therefore, conclude that all of the above listed faults can be considered active by criterion 3, as movement on the

- Livermore fault
- Greenville-Riggs Canyon fault
- Second strand of the Tesla fault

could reasonably be expected to produce movement on the formerly listed faults.

In summary, all of the faults listed in Table 4 are considered active per the AEC guidelines.²⁷

Seismologic Evaluation

The previous sections described the geologic history and the active or inactive status of faulting which could affect the Livermore site. In the following sections we will apply this information to an estimate of the earthquake hazard for the site. Rather than approaching this probabilistically, we will instead search for the worst possible earthquake that the site might experience. This earthquake will be used for design of critical structures, and is designated the Safe Shutdown Earthquake (SSE) for the site. We discuss the application of the SSE to the design and analysis of critical structures in a later section.

Calculationally and physically, the site response to earthquakes conveniently separates into two types. First, we consider the site response which results from shaking of the bedrock surface beneath the site. Here the surface response depends on the intensity of bedrock shaking and the properties of the soil layers which mantle bedrock. This response is clearly associated with earthquakes on more distant faults. Second, we consider the site response to surface rupture on very nearby faults.

However, before examining these two classes of site response, we need to characterize the potential earthquakes that might occur on each of these faults. We do this by first characterizing the earthquake source for each fault in a manner consistent with the known geology, and then associating each source with a maximum possible magnitude earthquake.

CHARACTERIZATION OF THE EARTHQUAKE SOURCE

Ideally, we would like to be able to predict a set of source parameters from the fault geology. For example, a convenient set might include:

- Length of rupture
- Offset
- Area of faulting surface
- Stress drop
- Rock strength

Knowing these properties of the source, we could then resort to empirical correlations to calculate a maximum magnitude earthquake associated with each fault.

In practice, however, all of these properties except the rupture length are difficult to associate with the geology, and hence their estimates may contain large errors. Because of this, we will estimate only the rupture length. This is the length of a fault that ruptures in an earthquake and is usually determined by the length of surface rupture and the distribution of aftershocks. We, however, require a priori estimate of the rupture length and therefore we use the results of Albee and Smith.²⁸ Their analysis of California earthquakes showed that although there was a wide scatter in the ratio of rupture length to fault length, * the ratio tended between 0.2 and 0.5. Guided by this, we

*We want to emphasize here that the fault length is the total length of the fault and is defined by geomorphic features and alignment of earthquake epicenters. The rupture length is the length of the fault that slips in a given earthquake, and is determined by the length of surface rupture and the distribution of aftershocks. The surface rupture length is the surface expression of the rupture length.

take the rupture length of each fault to be one-half its total length.

In Table 4 we apply this concept to all the faults near the site. The fault lengths are our best estimates, based on seismic history and geologic and geomorphic evidence. We have uniformly taken the rupture length to be one-half the fault length. We will see later that our results are not sensitive to this ratio.

RUPTURE LENGTH AND EARTHQUAKE MAGNITUDE

Figure 11 summarizes the presently available data relating rupture length to magnitude. The points displayed, which are taken from Ref. 29, are empirical and theoretical. The empirical data are derived from the length of surface rupture and the distribution of aftershocks. The theoretical points are derived from surface wave spectral densities³⁰ and seismic movement calculations.³¹ Also shown is Housner's³² fit to these and other data.

Fig. 11. Rupture length and magnitude data.

We use this relation to relate a given rupture length to an earthquake magnitude.

We summarize in Table 5 our application of this relation to each of the faults that surround the site. The important observation here is that the magnitudes for the faults in the very near vicinity of the site are not very different, in spite of a relatively large variation in rupture length. This shows the insensitivity of the earthquake magnitude to our estimate of the rupture length.

We will now apply this information toward an estimate of the dynamic environment at the site as a result of earthquakes on each of these faults. We consider first the effects of bedrock shaking.

BEDROCK SHAKING

Response of the Site to Earthquakes on the San Andreas System

Our aim here is to calculate the dynamic environment at the site surface due to earthquakes on the San Andreas, Hayward, and Calaveras faults. We start by estimating the bedrock acceleration at the site resulting from maximum magnitude earthquakes on each fault. We then calculate the response of the surficial soil layers to these accelerations.*

Peak Bedrock Accelerations

What we require is data relating the attenuation of peak acceleration with distance as a function of magnitude. The

*The alternative to this procedure is to estimate the accelerations directly at the surface. We believe that these estimates would contain large errors as the empirical data available are inadequate to represent the LLL site.

Table 5. Relation of rupture length and magnitude for faults surrounding the L.L.L. site.

Fault	Rupture length (km)	Magnitude (Richter)
San Andreas	670	8.3
Hayward	125	7.5
Calaveras	125	7.5
Livermore	17	6.5
Mocho	17	6.5
Tesla	17	6.5
Ramp Thrust	8.5	6.2
Corral Hollow	12.5	6.4
Doutherty	4.25	5.8
Carnegie	12.5	6.4
Patterson Pass	6.75	6.1
Greenville - Riggs Canyon	21	6.7

most current and comprehensive compilation of data has recently been presented by Schnabel and Seed³³; their interpretation of the data is summarized in Fig. 12. The accelerations were recorded on rock, and the distance is the shortest distance to the fault rupture.

In Table 6, we summarize the application of these curves to the San Andreas, Hayward, and Calaveras faults. Because the bedrock accelerations at the site are so similar, we will not continue to examine each fault individually, but instead we represent earthquakes on the San

Fig. 12. Earthquake acceleration attenuation (from Ref. 33).

Fig. 13. Taft N21°E accelerogram, scaled to 0.5 g.

Table 6. Relation of attenuation of peak bedrock acceleration to distance as a function of magnitude on three faults.

Fault	Magnitude (Richter)	Distance (km)	Peak bedrock acceleration at site (g)
San Andreas	8.3	58	0.4
Hayward	7.5	42	0.35
Calaveras	7.5	17	0.5

Andreas system by a single representative earthquake record. In our opinion the Taft record from 1952 Kern County earthquake (M7.7) is the most representative. The data (Fig. 13) were recorded at a range of 42 km and the magnitude is within the range of magnitudes estimated for the San Andreas system. Although the Taft record was not recorded on bedrock, we feel that the record nevertheless adequately captures the bedrock frequencies. We will, however scale the accelerations to bedrock values. We shall search for the most severe response of the surface by calculating the response of the soil layers to several bedrock accelerations.

We will apply the waveshape in Fig. 13 to the base of the soil layers beneath the site, and we will calculate the soil response by a method described by Seed and Idriss.^{34,35} The method solves for the dynamic response of horizontally bedded layers to vertically incident shear waves by solving the equations of motion for an equivalent lumped mass system. By means of equivalent linear analysis, the method takes account of the mechanical nonlinearity of the soils.³⁴ The method assumes that the soil response is not modified by slumping hillsides or soil liquefaction. These are reasonable assumptions to apply to the site as the topography is nearly flat and liquefaction presents no hazard.³⁶

Input to the method is the mechanical properties of the soil layers. These layers were first investigated in 1971 by Shannon and Wilson Inc.³⁷ Based on a log from a 90-ft borehole, they concluded that bedrock was at 400 ft. Additional information from logs of deep oil wells²¹ has become available since their study.

Fig. 14. Soil layer geology to 1000 ft.

These latter logs indicate that the sands, gravels and clays, which are the principal constituents of the surface layers, continue down to a depth of about 1000 ft. Based on these data, we assume that the soil layers at the site are 1000 ft thick and have the properties given in Fig. 14.

Our calculations show that a bedrock acceleration of 0.5 g produces the most

Fig. 15. Peak surface and peak bedrock acceleration.

severe surface accelerations (Fig. 15) and thus we present these results in detail in Fig. 16. Clearly the thick alluvial cover significantly attenuates the peak accelerations as well as modifying the frequency content of the waveshape.

EFFECT OF EARTHQUAKES ON NEARBY FAULTS

The distinguishing characteristic of earthquakes on the nearby faults is the effect of surface rupture. Whereas we could assume that for more distant faults the effects at the site would simply be shaking from below, here we must take into account the accelerations from nearby ruptures. We would expect that these accelerations could be very large simply because of the proximity of the source. There are indications, however, that the accelerations near surface ruptures are significantly higher than attenuation curves would indicate. This has been explained to be a result of the rupture interaction with the free surface.³⁸

There are two earthquake records that are useful in quantizing the site response. They are the accelerographs recorded during the Parkfield (M 5.5)

earthquake and the San Fernando (M 6.6) earthquake.

The Parkfield data are significant because:

1. The causative fault was right lateral strike slip (similar to what we expect near the site, based on regional tectonics and nearby geodetic measurements).
2. The data were recorded 200 ft from the surface rupture (many critical structures on the site are 200 ft or less from the Tesla fault).
3. The data were recorded on a thick layer of alluvium (the site is underlain by 1000 ft of alluvium).
4. The earthquake magnitude was 5.5 (in the range of magnitudes we estimate for very nearby earthquakes).

A peak acceleration of 0.5 g was recorded at Parkfield on a component transverse to the fault (Fig. 17). Although the component parallel to the fault malfunctioned and no data were recorded, 3.8 sec of the record has been reconstructed by Trifunac and Hudson³⁹ from seismoscope data by a calibrated technique. The reconstruction shows that the acceleration parallel to the fault was at least 0.6 g and that the vector sum of the horizontal components was at least 0.8 g.

The Pacoima Dam record, which indicated a peak acceleration of 1.25 g, is important because the station was only 4 km from the surface rupture and the earthquake magnitude was 6.6. The topography in the vicinity of the station was rugged, and this certainly amplified the peak acceleration. There have been several attempts to correct for this

Fig. 16. Soil response to the Taft record, scaled to 0.5 g.

amplification.⁴⁰⁻⁴³ These analyses indicate that the 1.25 g peak was amplified by as much as a factor of two, but the several cycles of 0.8 g accelerations represent unamplified bedrock accelerations (Fig. 18).⁴⁰

Fig. 17. Parkfield N65°E accelerogram scaled to 0.8 g.

Fig. 18. Corrected Pacoima Dam S74°W accelerogram. (The 1.25 g acceleration has been reduced to 0.8 g).

We can apply both of these data to the site by noting that other acceleration data indicate that close to the surface rupture:

1. The dependence of peak acceleration on magnitude is weak.⁴⁴
2. The correlation of peak acceleration with site geology is low.⁴⁵

In addition to the above data, there is some indirect evidence that is helpful. First, there are the order-of-magnitude calculations performed by Brune⁴⁶ and Ida⁴⁷ which indicate peak accelerations in the vicinity of the surface rupture can easily exceed 1 g. Second, there is some field evidence from near-epicentral damage studies that indicates accelerations greater than 1 g have already been experienced.^{48,49}

All these data, calculations, and observations clearly indicate that accelerations approaching 1 g are possible in near-epicentral regions for earthquakes of all magnitudes. In estimating the

Fig. 19. Response spectra from scaled Parkfield and corrected Pacoima Dam accelerograms, at 2% damping.

peak acceleration possible at the site we consider all these facts, but we place heavy emphasis on the Parkfield and Pacoima Dam records. Both these records indicated peak accelerations of around 0.8 g, and we take this as the peak acceleration that could be experienced at the site. We take the envelope of the Parkfield (N65E) and corrected

Pacoima Dam (S74W) response spectra as the response spectrum associated with this peak (Fig. 19) and we recommend that a synthetic seismogram be generated to represent the waveshape. The synthetic seismogram should have the property that its response spectrum closely matches the response spectrum shown in Fig. 19.

The Safe Shutdown Earthquake

We have considered the response of the site to two classes of earthquakes; those on the San Andreas system (the San Andreas, Hayward, and Calaveras faults) and those on the very nearby faults. The response of the site to each class of earthquakes was very different. For the San Andreas system, we calculate that the site will experience 0.32 g with the response spectrum shown in Fig. 16. For earthquakes on the nearby Tesla fault, we calculate that the peak acceleration at the site will be 0.8 g with the spectrum given in Fig. 19. These, in substance, are the results obtained in Ref. 2.

Our definition of the SSE would be similar to that given in Ref. 2. Because the accelerations from the very nearby faults are so much greater than the accelerations from the San Andreas system, we would recommend that the SSE response spectrum be taken as the envelope of the response spectra in Fig. 19. This spectrum is different both in shape and in peak acceleration, compared to the SSE (Fig. 20) given in Ref. 3. However, the Laboratory management,⁵⁰ after a careful evaluation of the SSE's in this report

Fig. 20. Response spectrum from Reference 3.

and in Refs. 2 and 3, has concluded that the degree of conservatism contained in this report and Ref. 2 is excessive. Their selection of the SSE and the SSE they recommend for inclusion in all SAR's for the site is the SSE in Ref. 3. For completeness, we have included a description of the SSE in Ref. 3 as Appendix A. This is taken directly from the original report.

We will extend this definition of the SSE by recommending a procedure to obtain the SSE waveshape. This waveshape should be a synthetic seismogram that has the property that its acceleration response spectrum closely follows the SSE spectrum in Fig. 20. We have not generated this synthetic seismogram, although there are standard techniques for doing this.⁵¹

The application of the SSE to structures depends on their use and criticality. For this purpose the AEC has grouped structures, piping and equipment into three categories:

1. "Category I Structures, Piping and Equipment" are those structures, piping, and equipment whose failure or malfunction might cause or increase the severity of an accident which would affect public health and safety. Also included are those structures, systems and components necessary to the safe shutdown and isolation of the facility.

2. "Category II Structures, Piping, and Equipment" are those structures, piping, and equipment other than Category I Structures, Piping, and Equipment that are necessary for continued operation without undue risk to the health and safety of the public.

3. "Category III Structures, Piping, and Equipment" are those structures, piping, and equipment not necessary for continued safe operation and that are not in Categories I or II.

Category I structures, piping and equipment are to be designed in terms of the SSE. Category II structures, piping, and equipment are less critical and, consistent with AEC guidelines, we recommend that these items be designed in terms of an Operating Basis Earthquake (OBE) whose spectral accelerations are one-half those given for the SSE. We have not considered here the design parameters for Category III items, as they are presently covered by the Uniform Building Code.

Summary

In response to the AEC request that we submit Safety Analysis Reports for all critical structures on the site, we have examined the seismic hazard at LLL in a form consistent with Section 2.5 of Ref. 1. We have thoroughly reviewed the geology and seismology of the Livermore Valley and surrounding regions. We have identified all the active faults in the region based on seismic history, geomorphology, and geodetic data. After carefully estimating the accelerations that the site would experience from

maximum magnitude earthquakes on each active fault, we conclude that the Tesla fault has the most potential for doing damage at the LLL site. The peak acceleration at the site will be taken as 0.5 g and the waveshape is assumed to have the acceleration response spectrum shown in Fig. 20. The waveshape corresponding to this response spectrum should be constructed synthetically.

We have covered all the applicable items in Section 2.5, with the exception of the risk associated with surface faulting.

Here the hazard is associated with large displacements that occur directly on the faults rather than with the accelerations that result from these displacements. The information presently available is

inadequate to assess this risk. Until this risk has been appraised, critical structures should be sited at least 330 m from the surface expression of the Tesla and Doutherty faults (Fig. 9).

Acknowledgments

I am deeply indebted to Frank Tokarz for his encouragement and technical assistance during the preparation of this

report. The report was reviewed by Larry McKague, to whom I am indebted for his valuable suggestions.

References

1. Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, U. S. Atomic Energy Commission (1972).
2. D. L. Bernreuter and F. J. Tokarz, Design Basis Earthquakes for the Lawrence Livermore Laboratory, Lawrence Livermore Laboratory, Rep. UCRL-51193 (1972).
3. Investigation of Faulting at the Lawrence Livermore Laboratory, (John A. Blume and Associates, Engineers, San Francisco, Calif., 1972).
4. L. H. Bailey, M. C. Blake, Jr., and D. L. Jones "Character and Significance of the Ophiolitic Crust that Forms the Basis of the Great Valley Sequence in Western California, " Geological Society of America Abstracts with Programs (Cordilleran Section), Vol. 2, No. 2, p. 68-69, 1970.
5. B. M. Page, "Geology of the Coast Ranges of California, " in Geology of Northern California, E. H. Bailey, Ed., Bulletin 190, California Division of Mines and Geology (1966).
6. K. J. Hsü, Geological Soc. Am. Bull. 79, 1063 (1968).
7. W. G. Ernst, J. Geophys. Res. 75 (5), 886 (1970).
8. E. H. Bailey, W. R. Irwin and D. L. Jones, Franciscan and Related Rocks and their Significance in the Geology of Western California, Bulletin 183, California Division of Mines and Geology (1964).
9. T. Atwater, Geological Soc. Am. Bull. 81, 3513 (1970).
10. B. M. Page, Geological Soc. Am. Bull. 81, 667 (1970).
11. W. F. Barbat, Geological Soc. Am. Bull. 82, 1541 (1971).
12. K. J. Hsü, J. Geophys. Res. 76 (5), 1162 (1971).
13. Z. Garfunkel, Geological Soc. Am. Bull. 84, 2035 (1973).
14. T. H. Rodgers and R. D. Nason "Active Faulting in the Hollister Area, " in Proc. Conf. Geological Problems of San Andreas Fault System (Stanford Univ., Stanford, Calif., 1968) p. 42-45.
15. R. D. Nason "Fault Creep and Earthquakes on The San Andreas Fault, " in Proc. Conf. on Tectonic Problems of the San Andreas Fault System, (Stanford Univ., Stanford, Calif. 1973) p. 275-285.
16. L. A. Raymond, Geological Soc. Am. 84, 3547 (1973).
17. Energy Dilemma, (State of California, The Resources Agency, 1973).
18. J. A. Taff, Geological Soc. Am. Bull. 46, 1079 (1935).
19. B. L. Clark, Geological Soc. Am. Bull. 46, 1025 (1935).
20. A. S. Huey, Geology of the Tesla Quadrangle (Bulletin 140, California Div. of Mines, 1948).
21. D. Shepard, Hersey Oil Company, private communication (March 29, 1974).

22. Alameda County Investigation (Bulletin 13, California Dept. of Water Resources, 1963).
23. Livermore and Sunol Valleys - Evaluation of Ground Water Resources (Bulletin 118-2, Appendix A, California Dept. of Water Resources, 1966).
24. Livermore and Sunol Valleys - Evaluation of Ground Water Resources, through 1968 (unpublished memorandum, California Dept. of Water Resources, 1970).
25. F. P. Vickery, J. Geology **33**, 608 (1925).
26. W. M. Gibson and H. A. Wollenberg, Geological Soc. Am. Bull. **79**, 627 (1968).
27. Nuclear Power Plants, Seismic and Geologic Siting Criteria (10 CFR Part 100), U. S. Atomic Energy Commission (Federal Register, Vol. 36, No. 228, Washington, D. C., Thursday, November 25, 1971), p. 22601.
28. A. L. Albee and J. L. Smith, "Earthquake Characteristics and Fault Activity in Southern California," in Engineering Geology in Southern California (L. A. Sec., Assoc. Engrg. Geologists, 1966) pp. 9-34.
29. R. C. Liebermann and P. W. Pomeroy, Bull. Seismol. Soc. Am. **60** (3), 879 (1970).
30. A. Ben-Menahem and D. G. Harkrider, J. Geophys. Res. **69**, 2605 (1964).
31. M. Wyss and J. N. Brune, J. Geophys. Res. **73** (14), 4681 (1968).
32. G. W. Housner, "Engineering Estimates of Ground Shaking and Maximum Earthquake Magnitude," in Proc. World Conf. Earthquake Engineering, 4th (Editorial Universitaria, Santiago, Chile, 1969).
33. P. B. Schnabel and H. B. Seed, Bull. Seismol. Soc. Am. **63** (2), 501 (1973).
34. H. B. Seed and I. M. Idriss, J. Soil Mech. Found. Div., ASCE **95** (SM5) p. 1199, (1969).
35. H. B. Seed and I. M. Idriss, Soil Moduli and Damping Factors for Dynamic Response, Univ. of California, Berkeley, Rept. EERC-70-10 (1970).
36. R. C. Murray and F. J. Tokarz, Seismic Analysis Livermore Pool Type Reactor Facility, Lawrence Livermore Laboratory, Rept. UCRL-51507 (1973).
37. R. J. Dietrich, Shannon and Wilson, Inc., (private communication) to S. Szybalski, Lawrence Livermore Laboratory, (Nov. 24, 1971).
38. A. Niazy, "Further Study on the Pacoima Dam Record of the San Fernando Earthquake of Feb. 1971," in Trans. Am. Geophys. Union **55** (4), 352 (1974).
39. M. D. Trifunac and D. E. Hudson, Bull. Seismol. Soc. Am. **60** (3), 785 (1970).
40. D. M. Boore, Bull. Seismol. Soc. Am. **63** (5), 1603 (1973).
41. C. M. Duke and P. J. Hradilek, "Spectral Analysis of Site Effects in the San Fernando Earthquake," in Proc. Fifth World Conf. Earthquake Engrg. (Ministry of Public Works, Rome, Italy, 1973), No. 77.
42. W. V. Mickey, V. Perez and W. K. Cloud, "Amplification Studies of the Pacoima Dam from Aftershocks of the San Fernando Earthquake," in Proc. Fifth World Conf. Earthquake Engrg. (Ministry of Public Works, Rome, Italy, 1973), No. 86.

43. R. B. Reimer, R. W. Clough and J. M. Raphael, "Evaluation of the Pacoima Dam Accelerogram," in Proc. Fifth World Conf. Earthquake Engrg., (Ministry of Public Works, Rome, Italy, 1973), No. 293.
44. N. C. Donovan, "A Statistical Evaluation of Strong Motion Data Including the February 9, 1971 San Fernando Earthquake", in Proc. Fifth World Conf. Earthquake Engrg., (Ministry of Public Works, Rome, Italy, 1973), No. 155.
45. D. M. Boore, "Empirical and Theoretical Study of Near Fault Wave Propagation" in Proc. Fifth World Conf. Earthquake Engrg., (Ministry of Public Works, Rome, Italy, 1973), No. 301a.
46. J. N. Brune, J. Geophys. Res. 75 (26), 4997 (1970).
47. Y. Ida, Bull. Seismol. Soc. Am. 63 (3), 959 (1973).
48. N. N. Ambraseys, "Maximum Intensity of Ground Movements Caused by Faulting," in Proc. Fourth World Conf. Earthquake Engrg., (Editorial Universitaria, Santiago, Chile, 1969), A-2.
49. C. F. Richter, Elementary Seismology (W. H. Freeman and Co., San Francisco and London, 1958).
50. J. L. Olsen, Lawrence Livermore Laboratory, private communication, Oct. 26, 1973.
51. G. W. Housner and P. C. Jennings, ASCE J. Engrg., Mech. Div. 90 (EMI) 113 (1964).

NOTICE

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Appendix A — Seismologic Evaluation*

GENERAL STATEMENT

The prior Blume report of November 1971, which was prepared for Maher and Martens, Architects, of San Francisco, provided a seismologic evaluation for a proposed plutonium facility at the Lawrence Livermore Laboratory. This evaluation was not intended to be in complete compliance with the seismic and geologic siting criteria for nuclear power plants of the AEC. A discussion of seismic history, tabulation of earthquakes, and map of earthquake locations from the 1971 report are included in the Appendix.

One of the objectives of this investigation was to obtain the necessary geologic information to provide seismic design criteria which would be in compliance with the AEC Reactor Site Criteria (see Appendix 10 CFR Part 100). The seismic design criteria previously submitted were reviewed and revised in the light of the new information obtained during the course of this investigation.

The seismic design criteria previously presented were based upon large earthquakes located on more distant faults. The faults on which earthquakes were located for design purposes were the San Andreas, the Hayward, and the Calaveras. The scope of the prior seismic evaluation did not permit the investigation of the location or possible active status of faults in the alluvium near the LLL site. Results of this investigation indicate that

faults do pass through the alluvium near LLL and that these faults are apparently active.

FAULTS NEAR THE LAWRENCE LIVERMORE LABORATORY

The faults which have been identified and located near LLL have been described elsewhere in this report. Beginning south of the Laboratory they are the Tesla fault, consisting of three strands, the Corral Hollow fault, the Ramp Thrust fault, the Doutherty fault, and the Carnegie fault. The Carnegie fault was known prior to this investigation, but the locations and nature of the other faults have not been described in detail prior to this report. The Doutherty fault is a new fault. The Ramp Thrust fault appears to die out at the southeast corner of the LLL property and extends from there to the southeast. The Corral Hollow fault also appears to die out in this area or merges with the Tesla fault. The Doutherty fault has been traced over a distance of about 3 miles, and its eastward continuation would be intersected by the Carnegie fault which puts a definite limit on its possible length. The Corral Hollow fault is about 9 miles long.

The two larger faults which will govern for considerations of vibratory motion are the Tesla, passing southwest of the site, and the Carnegie, passing northeast. A third fault, the Greenville, ends in the hills east of the Laboratory beyond the Carnegie fault and extends from there northwestward. Huey (1948) states that the Greenville fault does not

*Excerpted from Ref. 3.

join the Carnegie, and that its possible connection with other faults to the northwest is very doubtful. He estimates displacement on the Greenville to be about 100 to 300 ft in the vertical direction. The horizontal component is unknown, but is probably not large. Therefore, it is concluded that the fault is not a major structure. The Carnegie fault is about 16 miles long from a point at its junction with the Tesla fault to an estimated point extending 4 miles across the alluvium of the Livermore Valley. Its surface trace is about 1 mile from LLL.

The Tesla fault is the largest fault with the nearest location to LLL, and evidence of its recent activity is more conclusive than in the case of the other faults. The Tesla fault strikes S80°E and continues to the southeast a distance of 11 miles from LLL, where it intersects a major fault trending N45°W. This is considered to be the southeasterly end of the Tesla fault, since it is highly unlikely that two faults such as these intersecting at an angle of 35 degrees would act as one fault in case of surface rupture or fault displacement. The northwesterly limit of the Tesla fault is not known, but appears to be well beyond the area which was investigated as part of this study. A conservative estimate for the northwesterly extension of the Tesla fault is 5 miles across Livermore Valley beyond LLL. The total length of the Tesla fault is therefore estimated to be approximately 16 miles.

MAGNITUDE AND ACCELERATION

To estimate the probable magnitude of an earthquake that might occur on the

Tesla fault, it is necessary to estimate the length of a surface rupture that could occur on such a fault. The distance over which the Tesla fault has controlled the course of Arroyo Seco Creek is about 4 miles. This might be considered indicative of the length of surface rupture for a single displacement. It would be unreasonable to presume that surface rupture might take place along the entire length of the fault simultaneously. A surface rupture length of 6 miles is considered to be a conservative estimate because it equals more than one-third of the total fault length. Data published by Bonilla (1967), relating length of surface rupture on the main fault to earthquake magnitude, show that an earthquake magnitude of 5.7 would be a conservative estimate for a 6-mile length of surface rupture on a reverse fault.

This earthquake of magnitude 5.7 is the Safe Shutdown Earthquake as defined in the reactor siting criteria. It is located on the Tesla fault at its nearest point to LLL. Allowing for an 80-degree south dip on the Tesla fault and a focal depth of 5 miles for the earthquake, the epicentral distance would be about 2-1/2 miles from the center of Section 12, which is occupied by LLL. Because this is a very close earthquake, it is necessary to estimate the peak ground acceleration by comparison with available records of peak acceleration at very small distances from the epicenter. A recent publication by Boore and Page (1972) reviews data relating to accelerations near faults that have moved during moderate-sized earthquakes. This paper, in addition to reviewing prior data, presents a compilation of data which

have become available in recent years and have not been compiled previously.

For an earthquake magnitude of 5.7 and epicentral distance of 2-1/2 miles, a conservative estimate of peak ground acceleration based upon the data of Boore and Page is 0.5 g. Therefore, as a result of this review, our prior estimate of maximum acceleration at the site of 0.4 g must be revised to 0.5 g.

DESIGN BASIS EARTHQUAKE SPECTRA

The attached spectra are recommended for the LLL site. These spectra are based on a standardized configuration tentatively approved by the AEC for nuclear power plants. This spectral shape is based on a report prepared by John A. Blume & Associates for Division of Reactor Licensing entitled "Evaluation and Recommendations for Shape of Earthquake Ground Motion Response Spectra Based on Statistical Analysis of Thirty-Three Earthquake Records" (Report draft submitted October 31, 1972, final report in preparation).

DETERMINATION OF NEED TO DESIGN FOR SURFACE FAULTING AND DETERMINATION OF ZONE REQUIRING DETAILED FAULTING INVESTIGATION

Strand No. 2 of the Tesla fault and the Doutherty fault are nearest to LLL facilities. Both pass through Sec. 12 which is occupied by LLL. Each of these faults has been located only as a single

trace of surface rupture, the width of which is unknown but could be estimated in each case to be on the order of 50 to 100 ft wide. This width would be the control width as defined in 10 CFG 100. For an earthquake magnitude of 5.7, the width of the zone requiring detailed faulting investigation is twice the control width or from 100 to 200 ft wide.

However, the requirement that all Quaternary fault traces along each fault for a distance of 10 miles in each direction from the site be mapped to establish the control width has not been met. Because the faults are not exposed the traces cannot be mapped in the usual sense, but can only be located by indirect geophysical means.

It is stated in 10 CFR 100 that the zone requiring detailed faulting investigation is a zone "in which the possibility of surface faulting is to be considered." This is interpreted to mean that the structure must be designed to resist surface rupture if located within this zone. It is further stated in 10 CFR 100 that if the zone requiring detailed faulting investigation is less than 1/2 mile in width when determined as above, the zone shall be at least 1/2 mile in width.

Reference: John A. Blume & Associates, Engineers, Letter dated February 9, 1973, to Carl Forsberg, Lawrence Livermore Laboratory.

REFERENCES CITED

Bonilla, M. G., 1967, Historic Surface Faulting in Continental United States and Adjacent Parts of Mexico: Inter-agency Report, U.S. Dept. Interior, Geological Survey.

Boore, D. M. and Page, R. A., 1972, Accelerations near faults that have moved during moderate sized earthquakes: Geol. Survey open-file report.
Huey, A. S., 1948, Geology of the Tesla Quadrangle, California: Calif. Div. of Mines, Bull. 140, p. 75.

RE-RUN 12/2/74

R. Crawford 75

B. O'Dell 75

GENERAL OPERATIONS FOR THE LPTR
BUILDING 281

PREFACE

The Laboratory's basic safety policy is that operations must be planned and conducted in a manner that will prevent injury to personnel or damage to property. The minimum safety rules and practices necessary for achieving an accident-free work environment for the General Operations in Building 281 are stated in this Operational Safety Procedure.

To protect himself and other personnel from injury, everyone in the building -visitor, maintenance and service employee, and regularly assigned employee -is required to abide by this procedure and the LLL Health and Safety Manual. Regular building personnel are responsible for assisting others in the building in understanding the safety regulations relating to their particular activity and abiding by the safety requirements.

This procedure was prepared by building management assisted by Hazards Control. Confer with the Hazards Control safety team assigned to Building 281 for interpretation of this procedure and for planning your operations to comply with safety requirements. Each individual in the building must use this guidance effectively to avoid injury.

This Operational Safety Procedure was reviewed by:

R. R. Walston	Reactor Supervisor
L. G. Mann	Physics Section Leader
W. E. Nervik	Radiochemistry Division Leader
G. Mackanic	Hazards Control Department

This Operational Safety Procedure is approved by:

J. L. Olsen
Plant Manager

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GENERAL OPERATION FOR THE LPTR
BUILDING 281

A. GENERAL

1. Purpose

This procedure is a formal statement of responsibilities and general controls for safe performance of all operations in Building 281.

2. Facility

The Livermore Pool Type Reactor (LPTR) is a light water moderated and cooled reactor consisting of plate-type fuel elements and boron-containing control rods located in a 26' deep by 6.5' diameter tank. Various beam tubes and irradiation facilities allow access to neutron fluxes ranging as high as 7×10^{13} n/cm²-sec and gamma dose rates above 10¹⁰ Rads/hr. The reactor is used for basic and applied research in support of laboratory programs. The reactor is operated on a scheduled basis that permits a routine maintenance period, standard safety system checks, and a maximum operating schedule for experiments and irradiations.

Associated with the reactor are supporting safety systems designed to prevent industrial and nuclear accidents, and to minimize the hazard to the public in case of accident. These systems are described in various reports (UCRL-4919, UCRL-50147, and Seminar Notes). Maximum credible accidents and their relationship to safety equipment are described in the Safety Analysis Report for the LPTR (UCRL-50147).

Operating limits are set forth in LPTR Operating Limits dated October 26, 1966.

Associated with Building 281 are radiochemistry spaces, preparation spaces, shops and offices. Radioactive sealed sources, fissile materials, beryllium, toxic chemicals, and corrosive liquids are present in the building.

3. Changes

Any change in operations that significantly alters the scope or significantly decreases safety control will not be made until additional reviews have been completed, and appropriate supplements

written and signed by the person authorized to approve this Operational Safety Procedure (OSP). Any change in operations that does not decrease the safety control may be made with the joint concurrence of the Physics Section Leader, L. G. Mann, and the Hazards Control Area Representative.

4. Review

This OSP shall be reviewed annually by the Physics Section Leader and the Hazards Control Area Representative to establish that its contents are appropriate and adequate for current operations.

5. Criticality Safety

Any question relating to the criticality safety of a proposed operation that is not clearly answered in this procedure shall be referred to the LLL Criticality Safety Committee for resolution.

B. SCOPE

This procedure establishes guides for the conduct of all operations in the Building 281 complex.

Paragraphs A thru G of this procedure establishes a guide to the personnel responsible for the operation of the LPTR. Specifically included in this section is designation of functions, responsibilities, and limitations for operation of the reactor and any reactor-related system as defined in the current Safety Analysis Report for the LPTR (UCRL-50147).

Paragraphs H thru L of this procedure provides instructions of a more general nature for the entire facility.

This procedure must be general in scope to include the more obvious situations that are expected. In situations where supervisory or operations personnel feel that the safety of the facility or personnel is in doubt, they will take immediate steps to attain the safest possible condition under the circumstances.

C. RESPONSIBILITIES

There are two divisions of responsibility for the reactor. Each has been delegated separate safety responsibilities. The Chemistry Department, and specifically the Radiochemistry Division, has the responsibility for operating the reactor facility safely and within approved safety limits. The second chain of responsibility flows from

from the Plant Manager through the Hazards Control Department Head, to the Hazards Control Area Representative. This group is responsible for detecting and evaluating safety hazards, auditing reactor operations and establishing safety policy and procedural review. The Hazards Control Department is also responsible for environmental surveillance and for reviewing all daily operations that might present any hazard to building personnel. All reactor support personnel report to the Reactor Supervisor on technical and safety matters. Refer to Fig. 1 for organization chart.

1. The Radiochemistry Division Leader is responsible to the LLL Director through the Head of the Chemistry Department and the Associate Director for scientific support for the overall operational and experimental program related to the LPTR. The Physics Section Leader is responsible to the Radiochemistry Division Leader for general reactor operations and experiments at the LPTR. In the absence of the Radiochemistry Division Leader, the Physics Section Leader may act in his behalf. Throughout this manual where reference is made to the Radiochemistry Division Leader, it is specifically intended that the Physics Section Leader may act in his behalf.
2. The Reactor Supervisor will be directly responsible to the Physics Section Leader for the safe and efficient operation of the entire reactor facility as defined in the current LPTR Safety Analysis Report (UCRL-50147), LPTR Operating Limits, and this LPTR Operational Safety Procedure. In the absence of the Reactor Supervisor, the Assistant Reactor Supervisor may act in his behalf. Throughout this manual where reference is made to the Reactor Supervisor, it is specifically intended that the Assistant Reactor Supervisor may act in his behalf.
3. The Shift Supervisors shall be directly responsible for the operation and safety of the reactor facility on the particular shift they are working. They will also assist the Reactor Supervisor in matters such as scheduling of irradiations, burn-up calculations, and calculation of core change effects. In addition, they will advise and assist experimentalists in the use of the reactor facility.
4. The Supervisory Operator shall be responsible for certain specialized functions concerning operation of the reactor. In particular, the supervisory operator shall handle the administrative supervision of reactor operators regarding work schedules and nontechnical scheduling of reactor operations. He will also be responsible for the proper maintenance of all control room records, check lists, and logs. The supervisory operator will also function as a reactor operator when required.

----- Reactor Audits, Safety Policy Procedural Review

..... Environmental Surveillance, Safety Advice

..... Operational Scheduling

-1-1-1-1 Multidisciplined Advisory

Figure 1
Organization Chart

5. The Reactor Operators shall perform the specific function of operating the reactor console controls and perform those daily and weekly checks required for safe reactor operations. They shall also perform other duties as assigned by the Shift Supervisor or the Supervisory Operator, such work as sample insertion and withdrawal, minor shielding changes, routine electronic maintenance, and routine mechanical maintenance. It shall be their responsibility to verify proper approvals prior to target irradiations. They will maintain daily logs of events and processes.
6. The Mechanical Engineer shall advise the reactor staff on matters pertaining to the mechanical systems used in the reactor. He shall also be responsible for the design, procurement, and fabrication of mechanical equipment and will perform the required maintenance functions to keep the facility in proper operating condition. He shall be responsible for the quality assurance program for all mechanical systems. He shall maintain current "as built" drawings of reactor systems and component acceptance testing or quality control records.
7. The Electronic Engineer shall advise the Reactor staff in matters pertaining to the electronics system of the reactor facility. He shall also be responsible for the design, procurement, and fabrication of electronic equipment and will perform the required functions to maintain the facility in proper operating condition. He shall be responsible for the quality assurance program for all electronic systems. He shall maintain up-to-date drawings on electrical and electronic systems, maintenance logs and quality control records.
8. Hazards Control. The Hazards Control Department provides safety guidance and services for all LPTR health and safety matters. The Hazards Control Department is responsible for detecting and evaluating hazards associated with reactor operations and communicating these observations and results to the Reactor Supervisor. This Hazards Control liaison with the Reactor Supervisor is normally provided by the Hazards Control Area Representative. Hazards Control personnel will also perform the function of monitoring the facility for radiation and contamination level and will advise the operations staff on safety matters pertaining to the work environment. It shall be the responsibility of the Hazards Control Health and Safety Technician to make a radiation survey of the reactor building at least once a week and after any shielding change. This current radiation survey shall be posted in the corridor just outside the air-lock door for the information of all personnel. It shall be the responsibility of the Hazards Control Health and Safety Technician to call to the attention of the Shift Supervisor of any

background radiation level that exceeds acceptable levels and to post the appropriate barriers and warning signs, indicating to personnel those areas that are above acceptable levels.

9. Individual Experimenter

- a. All experimenters are responsible for the mechanical and electrical safety of their experiments. They should consult with engineering personnel if there are any questions in this regard. Each experimenter must review all proposed operations with the Operations Staff before starting work to determine whether a special OSP is required. This review must be conducted even though he may have participated in a similar operation previously.
- b. Each experimenter is responsible for the condition of the room and equipment he uses. This means disposing of trash promptly, minimizing combustible material, strapping gas bottles, maintaining the area in an orderly fashion, keeping equipment in good working condition, etc.
- c. Each experimenter is responsible that he and his support staff are completely familiar with emergency procedures (Appendix C).

10. The Reactor Safety Committee shall be an advisory committee to the Reactor Supervisor as set forth in Appendix B.

D. REGULATION OF REACTOR AND REACTOR SYSTEMS OPERATION

1. General Policies

Operation of the reactor console controls, or any equipment directly associated with the reactor, will be accomplished only by authorized reactor operators or supervisors. The operations staff will also perform the routine and nonroutine checks of the reactor operating system and maintain a continuous log on all changes made to the reactor system. Only reactor operators or qualified crane operators shall operate the bridge crane or the monorail crane.

2. General Requirements for Operating the Reactor

- a. Whenever the reactor is critical, or the control rods can be raised, or when the rods are not fully inserted, an authorized reactor operator shall be in attendance at the reactor console at all times. This attendance is defined as being within the area bounded by the reactor console side of the control room exclusion counter.

- b. Whenever the reactor is critical, or the control rods can be raised, a shift supervisor shall be in the Building 281 area.
- c. During any change in power, change in rod position, or in any situation that deviates from normal automatic steady power, a reactor operator or shift supervisor shall be in direct control of the reactor (except as in d below).
- d. The controls of the LPTR can be manipulated by an operator or supervisor in training, if such trainee is under the direction of and in the presence of an authorized operator or shift supervisor.
- e. Changes in the excess reactivity level of the reactor during operation normally will be made with the shim rods or regulating rods. If, however, it is absolutely necessary to make changes in experimental apparatus adjacent to the core of the reactor that would affect excess reactivity, these changes will be limited to that allowed in the LPTR Operating Limits.
- f. Refer to "LPTR Operating Limits" for specific conditions of reactor safety chains and startup criteria.

3. Bypass of Safety Interlocks or Safety Limits

The bypass of any safety interlock or safety limit that affects the LPTR safety chains of the operating reactor as listed in the "LPTR Operating Limits" is specifically prohibited. The bypass of any safety interlock or safety limit not affecting the LPTR safety chain may be done with the approval of the Shift Supervisor on duty; and it shall be his responsibility to ascertain that the basic safety of the reactor system has not been impaired by such a bypass. If a standard bypass switch is not available, electronics personnel installing jumper leads shall tag the jumper and place a mating tag in plain view of the reactor console Operator. The Reactor Supervisor or Shift Supervisor shall verify that the jumper has been placed on the correct terminal points.

The reactor operator on duty shall make an entry in the daily log indicating the actuation or placement of such a bypass. When a bypass is removed, either by actuation or a switch or by removal of a jumper, a second entry shall be made in the daily log to indicate that the bypass has been removed.

4. General Policy - Experimentalist

- a. Experiments involving any potential hazards to the reactor shall be governed by approved Operational Safety Procedures. Refer to Paragraph G for instructions on Experiment Controls.
- b. Attendance at an experimental facility in the reactor dome is limited to those persons who have a clear understanding of safety policy and emergency procedure. Each experimenter shall clearly understand the safety instructions listed in Appendix C.

5. General Policy - Support Personnel

- a. Routine maintenance of reactor auxiliary systems will be performed on a schedule basis by qualified support personnel with prior concurrence of the Shift Supervisor or Supervisory Operator. Nonroutine maintenance will be scheduled by the Reactor Supervisor.
- b. Each member of the support staff shall have a clear understanding of the safety instructions listed in Appendix C.

6. Definition of Category of Operation and Maintenance

a. Class I

- (1) Routine startup or operation of the reactor with the loading and configuration unchanged from previous runs. Startup after a scram whose cause is minor. Loading of samples in an S-tube, rabbit, reflector position, or loading of samples in a core facility with the reactor shut down. Maintenance performed by operators or technicians that does not involve major changes to the reactor control system or console.
- (2) The above Class I operations require the prior approval of the Reactor Supervisor or the Shift Supervisor on duty, and requires the presence of one of the above in the Building 281 area (within immediate communication).

Note: Reactor startup following an occurrence that may be within the context of Appendix D will be a Class II or higher operation.

b. Class II

- (1) Operation of the reactor after a minor core change. Startup after a scram when the cause of the scram is in doubt and believed due to a potentially hazardous condition. Maintenance operations such as changing magnets or replacing clutch switches. Minor shielding changes and loading or unloading of collimators or beam tube plugs. Removal of large sources from the reactor tank requiring the use of a coffin. Routine fuel cycles.
- (2) The above Class II operations require the prior approval of the Reactor Supervisor. The work is to be performed under the direct supervision of either the Reactor Supervisor or a Shift Supervisor.

c. Class III

- (1) Nonroutine operation or maintenance involving a major change in the core configuration or major shielding change. Startup of the reactor after an extended shutdown period. Spent fuel shipments.
- (2) The above Class III operations require prior approval of Reactor Supervisor. The work is to be performed under the direct supervision of the Reactor Supervisor.

d. Class IV

- (1) Special tests or major operations of an unusual or potentially hazardous nature as determined by the Reactor Supervisor or Radiochemistry Division Leader.
- (2) The above Class IV operations require the prior approval of the Radiochemistry Division Leader. The work is to be performed under the direct supervision of the Reactor Supervisor.
- (3) The operation shall be reviewed by the Reactor Safety Committee before starting (Appendix C).

7. Operational Limitations

The "LPTR Operating Limits" is established as a separate document approved by LLL and AEC managements. These limitations represent a maximum envelope of flexibility for which the Safety Analysis

Report and this Operational Safety Procedure are valid. Other limitations are included based on maintaining the reactor facility in a safe operating condition. These limitations are to be strictly adhered to: The reactor is to be immediately scrammed and the Reactor Supervisor notified if any condition exists that falls outside of these limitations.

8. Power Level

a. Normal Power Level

The LPTR is authorized to operate at a maximum power level of 3.15 MW. It is the responsibility of the Shift Supervisor on duty to ascertain the power level of the reactor and to make the necessary adjustments in power level instrumentation to assure that the authorized level is not exceeded. However, the power level shall be maintained as close to 3.0 MW as possible.

b. Power Level Determination

The primary standard for power determination shall be a calorimetric determination using primary coolant Δt and primary coolant flow rate determined under equilibrium conditions.

E. OPERATION AND MAINTENANCE

1. General Policies

A standard set of LPTR Operation and Maintenance Procedures (OMP's) has been written as a check-list guide and record in performing most maintenance operations in the reactor vessel of a hazardous or complicated nature.

These OMP's also cover work on components that are outside of the reactor vessel, but involve the safety of the reactor system. These procedures are to be followed, both as to instruction, and to the order of performance of work. Deviations from these OMP's will be allowed only if considered necessary by the Shift Supervisor. Before continuing with the procedure, the person supervising the operation will determine that such a deviation will not produce a condition that is either unsafe or violates the "LPTR Operating Limits." Any such deviation from the written procedure should be noted by the person supervising the procedure, with the specific reason for such deviation or rearrangement of order. Completed procedures shall be reviewed, signed off, and filed.

Operational Safety Procedures (OSP) are supplementary procedures controlling experiments or special situations and are governed by the Health and Safety Manual and Paragraph G of this procedure.

2. Reactor Top and In-Tank Maintenance Requirements

a. General Policy

All work performed in the reactor tank shall be done by authorized reactor operators or technicians. Routine operations such as insertion of irradiation samples or vacuuming the top of the fuel elements shall be Class I operations. All other work, particularly that involved with the movement of equipment in or near the reactor core, shall be Class II or higher, depending upon the extent of the change involved. All movement of fuel elements, dummy fuel elements, and irradiation facilities in the core or tank storage rack shall be authorized by a Transfer Order Form. This form must be approved by the Reactor Supervisor if fuel elements or reflector elements are involved or if a significant reactivity change is expected. Transfer order forms for dummy fuel elements or irradiation facilities must be approved by a Shift Supervisor. Major core changes shall be a Class III operation.

b. Reactor Top Structure

Reactor top work not involving work on the reactor control system or requiring removal of deck plates must have operator concurrence. An operator must be in attendance either in the reactor room or in the control room. All other reactor top work will be considered in-tank maintenance.

c. In-Tank Maintenance

In-tank maintenance shall consist of any work involved in maintenance of the reactor system inside the main reactor tank, or work on actuators and other mechanisms in the control structure on the reactor top. Such work will be performed only when the reactor is completely shut down with shim rods inserted in the core, except as in Paragraph (1) below. An operator must be in attendance on the reactor console with the proper nuclear instrumentation operative, including the Log N, log count rate channel, and radiation monitoring systems if the core is loaded or is being unloaded.

As far as possible, work performed in the reactor tank will be in accordance with maintenance procedures written either to cover a general operation, or maintenance procedures written to cover specific maintenance to the reactor system.

(1) Rod Maintenance

To remove, replace, or perform maintenance on a shim rod or its associated special fuel element, sufficient fuel elements shall be removed so that the resultant core is left subcritical by at least the worth of the most reactive remaining shim rod. This removal of fuel elements must be accomplished before the removal of a shim rod or its associated special element. Maintenance work on, or replacement of, any portion of the regulating rod system, including its associated special element, does not require removal of surrounding elements.

(2) Core Change

Core configuration changes may be required to modify reactivity or to shift the core for experimental purposes. Before performing any reassembly procedure, the Reactor Staff shall calculate the excess reactivity available in the core upon reassembly and the worths of the shim rods if they are expected to change. The Reactor Supervisor shall then verify that it is safe to load the core in the proposed configuration and whether this can be done within the limitations stated in the "LPTR Operating Limits." Upon startup these calculated values shall be verified by direct measurement.

(3) Critical Assembly Procedure

A Class III major core change will require a critical assembly procedure for reassembly of the core in accordance with the established Operational Maintenance Procedure No. 2. The critical assembly procedure shall, in general, consist of loading the core to a starting condition that is safely below critical, and from this point, inserting fuel elements one at a time taking both rods-in and rods-out count rates after insertion of each element. During this critical assembly procedure, the controls will be operated only by an authorized reactor operator or supervisor. The minimum critical mass shall be determined by the critical assembly procedure from the inverse multiplication plot method. When criticality is reached, the

procedure shall continue with the insertion of one fuel element at a time, making excess reactivity measurements as each element is introduced. The critical assembly procedure will continue until the proper excess reactivity is loaded in the core.

3. Reactor Control System Changes and Maintenance Requirements

a. General Policies

All work performed on the reactor control system shall be done by qualified technicians or authorized reactor operators. This work shall include any electronic or mechanical maintenance or changes performed on the reactor top structure that directly affects the reactor. This work must be concurred with and scheduled by the Supervisory Operator or Shift Supervisor.

b. Routine Electronic Maintenance

Electronic maintenance shall consist of only that work involved with the maintenance of the electronic components in their normal design condition. Reactor operators shall only replace complete chassis with spare chassis or replace minor components that can be changed without adversely affecting the remainder of the system. Electronic technicians shall troubleshoot and replace major components where spare parts are available. Routine electronic maintenance will generally fall in a Class I maintenance category.

c. Routine Mechanical Maintenance

Routine mechanical maintenance shall consist of only that work involved in maintaining rod drive actuators, manometers, valves and other such mechanical components in the reactor control system in their normal design condition.

d. Major Electronic or Mechanical Changes

If it is considered advisable to make a change in the control system of the reactor, such a proposed change will be discussed with the operations staff and the Reactor Supervisor to determine the feasibility and safety of such a change. Included in this appraisal shall be the determination of the class of operation involved, and whether the proposed change would retain the control system characteristics within those listed in the "LPTR Operating Limits." If the proposed change is categorized by the Reactor Supervisor as a Class III operation, the Electronic

Engineer or Mechanical Engineer will be requested to make a presentation to the Reactor Safety Committee. This presentation will include the justification for the change, the expected effect on the safety system, and any other data pertaining to the proposed change. Upon completion of such a major electronic or mechanical modification to the reactor control system, the subsequent startup procedure shall be at least a Class II operation.

4. Schedule of Safety Checks and Surveillance Requirements

a. Checklist Before Startup

Prior to a startup of the reactor under any of the above classes of operation, either the Startup Checklist or the Interim Checklist shall be completed by the reactor operators. The Startup Checklist shall be used in cases where the reactor has been previously shut down and secured and a Shutdown Checklist completed, or a sufficient time has elapsed since the original startup was completed.

b. Weekly Checklist

The Weekly Checklist shall be completed by the reactor operators and approved by the Shift Supervisor on duty prior to reactor startup following the weekly shutdown.

c. Shutdown Checklist

Prior to leaving the reactor unattended for an extended period, a Shutdown Checklist shall be completed and approved by the Shift Supervisor (See Paragraph E 6).

d. System Checks

The operational limitations indicated in the "LPTR Operating Limits" and other parameters and systems shall be checked as shown in the following schedule:

Worth of shim bank	Change in core configuration or every 5 years
Core excess reactivity	Weekly
Ratio of shim bank $\Delta k/k$ to core $\Delta k/k$	Change in core configuration
Shutdown margin	Change in core configuration
Stuck rod criteria	Change in core configuration

Core fuel loading	Start and end of each fuel cycle
Core thermal characteristics analysis	Change in core configuration
Core flux profile	Change in core configuration or every 5 years
Scrams and rundowns	Weekly
Rod drop times	Fast-slow, alternate fuel cycles respectively
Rod thickness gaging	Semiannually
Magnet holding and drop currents	Weekly
Composite safety amplifiers	Weekly
Count rate channel	Weekly
Log N channel	Weekly
Linear level channels	Weekly
Air exhaust system	Weekly
Building seal system	Weekly
Scrubber system	Weekly
Emergency decontamination system	Weekly
Radiation monitoring system	Weekly
Fission product monitors	Weekly
Primary Temp. sensing system	Quarterly
Primary flowrate indication	Quarterly
Lead nose piece/fuel element interference check	Annually

A weekly area radiation survey will be conducted.

5. Operation or Maintenance During Other than Normal Operating Hours

a. General Policies

Operation of the reactor during hours other than the regular operating hours will be performed with the required complement of personnel as set forth in Paragraph D.1. and D.2. Supervision and approvals shall be in accordance with Paragraph D.6. Such work specifically includes any and all maintenance performed on the reactor as covered in Paragraph F.1., F.2., and F.3.

b. Access of Personnel

Work in reactor dome when the reactor is unattended by operators shall be strictly administered, so that personnel will in no way tamper with the reactor controls or associated equipment. Only

persons on the off-hours access list maintained by the Security Department will be authorized entry in the reactor building when the security alarm is in effect. This off-hours access list includes the reactor operating staff, scientific personnel attached to the LPTR facility, and Support Personnel who perform routine work such as the filling of traps during off-shift hours. The list, restricted to the absolute minimum number of persons, is reviewed bi-annually by the Reactor Supervisor. All persons on this off-hours access list, except members of the operating staff, are required to have met the requirements of Appendix C. Personnel not assigned to the Operations staff are forbidden access to the top of the reactor and are instructed not to change the status of the reactor console without first contacting a member of the operations staff.

6. Shutdown Procedures

a. Routine Shutdowns and Scrams

The LPTR may be routinely shut down from any power level by initiating a manual scram on the reactor console, or it may be scrambled by instruments. After the scram, an operator or supervisor shall remain in attendance at the reactor console during the insertion of the magnet assemblies. The reactor is considered shut down when the control rods are fully inserted and the control rod magnet power is turned off.

b. Extended Shutdowns and Subsequent Startup Procedure

If it is necessary to shut the reactor down for extended periods, the Shutdown Checklist shall be completed; and if it is considered advisable by the Radiochemistry Division Leader or the Reactor Supervisor, the reactor core will be disassembled and the fuel elements will be transferred to the storage rack in the reactor tank. Startup after such an extended shutdown necessitates a Class III procedure.

F. NUCLEAR FUEL MANAGEMENT AND ACCOUNTABILITY

1. Fuel Program

All core changes shall be made in accordance with Paragraphs D and E of this procedure. The operation of removing a spent fuel element, rearranging a series of the remaining fuel elements, and inserting a new fuel element, shall be a Class II procedure.

Fuel element, reflector, and dummy element locations within the pool shall be displayed in the control room by serial number tags placed on the LPTR lattice board in the same relative location as the component in the pool. Spent fuel elements will normally be retained in the storage rack until shipment for reprocessing. The Shift Supervisor in charge of this fuel change will determine the new loading of the core and log the core excess reactivity normalized to xenon free and 70°F and the change in core excess reactivity due to the fuel movement. It shall be the responsibility of the Shift Supervisor to ascertain that the normalized excess reactivity is within the limits set forth in the "LPTR Operating Limits." The reactor will be immediately scrammed and the Reactor Supervisor notified if the reactivity change does not agree with the expected value. A current file of these weekly excess reactivity surveys will be maintained.

2. Fuel Accountability Procedures

A Material Balance Area will be maintained in accordance with the general Laboratory Accountability Procedures Manual. Accounting of fuel elements will be on the basis of accepted shipping weight of U²³⁵ less the burn-up in each position the element has occupied. Burn-up calculations will be made, and the current U²³⁵ content of each element recorded after core change. Location of fuel elements will be recorded by a number on the LPTR Lattice and Storage Inventory Form. The fuel element serial number will be checked each time it is cycled in the core and when it is transferred to the rack.

Fuel elements in the fuel storage rack will be carried on the accountability records as having a constant U²³⁵ content and will be physically inventoried by simply observing their presence in their storage positions. Upon movement of a spent fuel element elsewhere within Lawrence Livermore Laboratory, the amount of U²³⁵ shall be held constant in the LPTR accountability records. The elements, when shipped to the reprocessing facility, will be transferred as having the original U²³⁵ content less the sum of calculated burns in the various core positions. At that time, it may be dropped from the LPTR accountability records.

G. EXPERIMENTAL AND IRRADIATION PROGRAMS

1. Safety Analysis Review

All major experimental programs shall be reviewed by the Reactor Safety Committee. The experimental personnel and support

engineering personnel involved in any such experimental program shall prepare an in-depth analysis of hazards, if any, associated with the experimental program and present this to the Reactor Safety Committee. The Committee shall make recommendations for changes necessary in the experimental program or equipment and advise the Reactor Supervisor on the safety of the proposed experiment. An Operational Safety Procedure for out-of-core and in-core experiments dealing with certain transuranic elements, tritium, or similarly hazardous materials, shall be written by the experimenter and reviewed by Hazards Control and the Reactor Safety Committee for each experiment.

2. Non-LLL Experiments

The Plant Manager and the Business Office must approve any irradiations or experiments for any agency outside of the Lawrence Livermore Laboratory. The Plant Manager approves those matters pertaining to Reactor Safety, the Business Office approves those matters pertaining to contractual relationships.

3. Procedure Category

Installation of long-range experimental programs using any of the beam tube facilities, the through tube facility, or thermal columns shall be considered a Class III operation.

4. Sample Irradiation

a. General Policies

The LPTR provides convenient facilities for the production of radioisotopes, such that loading and unloading can be accomplished while the reactor is operating. These sample irradiations shall be strictly administered by the operations staff. Samples to be irradiated shall be described according to material, quantity, container, expected activity in curies, flux desired, and desired time of irradiation on a Standard LPTR Irradiation Request. The Irradiation Request will be reviewed by Hazards Control and will be reviewed and approved by the Shift Supervisor on duty before the sample irradiation. Identical requests need not be resubmitted for Hazards Control review.

b. Rules and Regulations for Sample Irradiations

- (1) Irradiations in the core central void and core reflector positions require the determination of the reactivity effect upon inserting or removing the sample.

- (2) All fissile samples shall be at least doubly contained.
- (3) No explosive or unstable materials shall be irradiated in the core.
- (4) Use of liquid gas or stored energy devices in the tank require special review.
- (5) Sample containments shall be able to withstand 100% over any expected pressures and under maximum temperature transients.
- (6) A Health and Safety Technician shall be present
 - (a) For all rabbit shots.
 - (b) Removing sources from reactor tank and other irradiation facilities.
 - (c) On all movements of large sources in and around the dome.

It will be the responsibility of the Shift Supervisor or the Reactor Supervisor to interpret the specific request for irradiation to determine what category of operation applies. If it is determined that the proposed irradiation would fall into a Class III or higher operation, it will then be necessary for the requestor to present the proposed irradiation to the Reactor Safety Committee for review.

c. Source Handling Requiring Transfer Coffin

The insertion or removal of any source from the reactor tank that is judged to be sufficiently active to require the use of a transfer coffin will be a Class II operation and will require a written OMP to cover the entire operation. Approval is required by the Reactor Supervisor for each specific operation. The transfer coffin shall be used over the reactor top structure only when the reactor is scrammed.

5. Shielding

a. General Policy

The design and installation of external shielding located adjacent to the experimental apparatus on the reactor floor or balcony, shall be the responsibility of the experimental and

engineering support personnel. The responsibility for the determination of adequacy of such shielding, however, shall rest with the operations staff and Hazards Control personnel.

b. Minor Shielding Changes

A shielding addition to improve a particular shielding arrangement by experimental or operations personnel, shall be a Class I procedure and shall be checked during the approach to power, and at power by the Hazards Control Health and Safety Technician.

c. Major Shielding Changes

Any new installation of shielding or significant change to a shielding configuration shall be a Class II or higher procedure. This category of shielding change shall include operations such as beam plug insertion or removal, collimator insertion or removal, etc. This procedure shall also include the first approach to power after the installation of any major shielding arrangement located adjacent to experimental apparatus. Such changes will be checked by the Hazards Control Health and Safety Technician.

H. MATERIAL CONTROL

1. General

Some materials used in this building are potentially hazardous because they are radioactive, toxic, or flammable.

Some materials require special accountability, while others may be hazardous in a reactor environment.

These materials shall be used only in the rooms designated as appropriate by the Reactor Supervisor and Hazards Control team personnel. Radioactive material must not be used or stored in offices except that sealed radioactive source associated with radiation detection instruments may accompany the instruments.

2. Radioactive Material

The quantity of unencapsulated radioactive material in a laboratory equipped only with a hood shall not exceed the limits established for workplaces for radionuclides (Health and Safety Manual Supplement 33.42). The maximum quantity of radioactive material for a gloved box operation shall be evaluated for each operation by the

Hazards Control Area Team. The radioactive material may be delivered to the laboratory in which it will be used but only the Operations Personnel, Hazards Control Technician, or Area Representative are authorized to affix the acceptance signature on the Controlled Materials I.D. card. Operations will notify the Hazards Control Technician of any receipt of controlled materials.

Large radioactive samples are used in Building 281. The use of sealed sources defined as Class II, III, or IV is outlined in the LLL Health and Safety Manual Subsection 33.45.

Control of targets and activated components is the responsibility of reactor operations personnel in coordination with the Hazards Control Technician.

Fissile materials must be carefully accounted for and may require Operational Safety Procedures for control. Fissile materials in the form of targets or samples will not exceed 350 grams ^{235}U , and 220 g of ^{239}Pu . All targets and samples for irradiation in the reactor are closely administered by Irradiation Request forms (see Paragraph G.4.).

Radioactive materials are not to be taken into or removed from the dome without the operations staff approval.

3. Transportation

Movement of all radioactive materials in and out of Building 281 except Class I sealed sources, must be done by or under the control of the Materials Management Group. (See LLL Health and Safety Manual Subsection 33.41). "Controlled Materials I.D. Tag" must accompany each container of radioactive or toxic material. Arrangements for transportation may be made through the Hazards Control Health and Safety Technician. Materials Management will deliver all radioactive or toxic material for Building 281 to LPTR Operations or to the Hazards Control Health and Safety Technician. Materials Management will not deliver hazardous materials exceeding the amount, or containment requirements stipulated in this OSP unless authorized by a supplement.

If an experimenter wishes to transport radioactive or toxic materials himself because of their short half life, fragility, etc., he will:

- a. Contact the Hazards Control Technician advising him of his desire to transport the material.
- b. Identify the material, level of activity, and destination. The Health and Safety Technician will contact Materials Management advising them of the request and obtain permission for the experimenter to transport the item. Material Control Tags will be completed and accompany the material.

4. Accountability

SS materials in Building 281 are assigned to Materials Balance Area 510. Accountability will be supervised by Materials Management. Class II, III, and IV sources are accounted for by LLL source inventory system. Class I sources are of insufficient hazard to warrant accountability controls.

5. Storage

All radioactive materials must be stored in containers approved by Hazards Control and have adequate shielding when not in work enclosures. These containers must have labels stating the nature of the material, the radiation levels and the owner. The loss of a sample, or sudden reduction in its radiation level should be reported immediately to Hazards Control.

All radioactive material that is not in experimental or evaluational use shall be stored in an approved enclosure. The enclosure must be kept locked when a Class IV source is present.

6. Waste Disposal

Radioactive waste retention tanks are supplied for storage of liquids from the chemistry lab sink in Rooms 1311, 1184, from the sample handling space, Room 1323, and from the sink in the reactor dome. These sinks shall be used for potentially contaminated liquids. Known high level contaminated fluids are to be collected in carboys. Solid radioactive waste showing less than 2.5 m rem/hr or less than 1 μ Ci of alpha may be disposed of via regular radioactive waste cans located in the area. All other radioactive waste must be disposed of by the Health and Safety Technician.

7. Toxic or Flammable Material, or Material Hazardous in a Reactor Environment

The usage of industrial purge gases, particularly argon, around the reactor shall be cleared with the Shift Supervisor.

Materials with high thermal neutron absorption cross sections can cause reactor effects or unusual heating effects (in the material) and their usage around the reactor must be cleared with the Shift Supervisor.

Only the minimum needed supplies of volatile flammables is permitted in the reactor area. Plastic bottles may be used to contain small (1/2 liter or less) quantities of volatile flammables. Larger volumes must be kept in safety cans that are equipped with flash arresters. In general, these materials will not be stored in the reactor building (Dome).

I. GENERAL BUILDING CONTROLS

1. Work Areas

Building 281 consists of the reactor containment building, an adjoining laboratory building with shops, offices, and lab spaces, a cooling tower and other related structures. All radioactive work is strictly limited to the laboratory spaces or reactor containment building. Hoods, shielded caves, and gloved boxes are available in the building to assist in preparation of new samples or radioactive targets. Any work involving radioactive or toxic materials will require the responsible participant to be aware of the hazardous properties of such materials, the proper use of applicable detection instruments and the possible requirements for respirator protection.

a. Shops

The shops are not normally radioactive work areas. Work on radioactive objects that will produce filings, dust, or chips is not allowed except with proper controls and under close surveillance by a Health and Safety Technician.

b. Hot Chemistry Laboratories (Rooms 1311, P-11, 1307, and 1323)
Radioactive samples (targets) may be received from the reactor via the pneumatic rabbit systems that exist in the shielded caves in each of these laboratories. The pneumatic rabbit systems will be operated only by qualified operators. A hood

is available to the experimenter for ventilation control in the Hot Chemistry Laboratory. Use of the hood, use of shielding in the hood, and control of materials in the hood are the responsibility of the experimenter. Hazards Control will verify that the experimental arrangement conforms to the criteria for radioactive nuclides (Health and Safety Manual, Subsection 33.42, Supplement 33.42, and OSP 281.11). Hazards Control will maintain routine surveillance for these spaces to prevent the spread of contamination or unnecessary exposures.

c. Other Areas

All work with radioactive materials in other areas of the reactor facility, in addition to those areas delineated above, must conform to the established criteria for suitable work place and protective measures as specified in the LLL Health and Safety Manual.

2. Operation Controls

- a. Treat these operations as particularly hazardous:
 - (1) Use of equipment that controls a radiation beam.
 - (2) Use of powdered or liquid radioactive materials.
 - (3) Use of stored energy or open flame related to 1 or 2 above.
- b. Request the Hazards Control Area Representative to review every proposed operation that fits any of the above categories to establish whether:
 - (1) The operation is within the control capabilities of the building's facilities.
 - (2) The operation can be performed under this OSP or will require a supplement.
 - (3) The operation will require unusual environmental or operational surveillance.
- c. Proceed with an operation or experiment only when all the controls are available and operative to prevent a hazardous occurrence.
- d. Use available personnel protective equipment to prevent injury if a mishap should occur, and know where local monitoring and emergency equipment is located.

- e. Maintain work areas in a clean, orderly, and uncluttered condition.
- f. Store supplies or equipment in locations so designated. Storage in corridors or mechanical equipment rooms is not acceptable.

J. CRAFT ACCESS TO BUILDING 281

Access by the crafts to various sections of Building 281 including the cooling towers must be controlled because of the nature of the work normally performed in this building. Requirements in any given area vary so that no hard and fast rules can be made. The following criteria are to be used as a guide for all persons working in the building:

1. Building in General

The Building Supervisor and/or Building Health and Safety Technician must be notified before the start of any work in any area and also upon its completion. Notification before the start of each day is required if a job takes longer than one day.

Do not move anything in any laboratory work area without contacting either the responsible chemist or the Hazards Control Health and Safety Technician. Exceptions to this rule are wastebaskets, chairs, other office-type equipment, and laboratory equipment mounted on wheels not connected to an experiment. It is better not to move something than to move the wrong thing.

2. Reactor Room

Access to the reactor room, control room, or the reactor dome roof is controlled by the Reactor Operator on duty. Work in these areas must be cleared with the Reactor Operator. All rules of conduct within these areas will be specified by the Reactor Operator on duty.

3. Roof Access

Access to the roof of Building 281, except the dome, is controlled by Hazards Control because of the occasional hazardous nature of fumes being exhausted through hood and glove box blowers.

The following rules have been established to ensure that persons working on the roof are not exposed to hazardous exhaust fumes:

a. To Obtain Permission

- (1) Check with the Health and Safety Technician, Room 1169, Ext. 7326, to determine if a hazardous condition exists. He will need to know the nature and location of work to be done.
- (2) If the Health and Safety Technician is not readily available and the work is routine, contact: (1) Virgil McIntosh, Ext. 7909, (2) Jim Ruys, Ext. 7321, or (3) personnel working in the rooms over which you plan to work. Determine by inquiry if toxic materials are being used.
- (3) Use only the designated access ladders outside the building. If you must use some other route, contact the building Health and Safety Technician first.

b. While on the Roof

- (1) Stay in the area of planned work. If additional work must be done in other areas, check with the Health and Safety Technician.
- (2) Avoid breathing exhaust from the blowers. If possible, turn off the blower being worked upon. Permission to do so must be obtained from the chemist or Health and Safety Technician.

If there are any questions regarding these instructions, contact the Building Supervisor, V. McIntosh, Ext. 7909, or the reactor operations staff, Ext. 7321, for clarification.

K. CRITICALITY SAFETY

Prevention of a nuclear criticality accident is based on limitation of the amount of fissile material in any work station at any one time. The maximum amount of fissile material, with the exception of fuel elements, allowed in any work station in Building 281 is 350g ²³⁵U or 220g ²³⁹Pu.

L. EMERGENCIES (See Appendices A and E)

If an accident occurs that injures personnel, damages property, starts a fire, or causes a dangerous situation, the Laboratory requires that it be reported to Hazards Control immediately so that help can be provided promptly. The Laboratory's policy is that it is better to have a false alarm than to delay reporting a small occurrence which

then might develop into a serious situation. Therefore one of the first actions after an accident must be to report it to Ext. 7333 (Emergency Dispatcher). If the accident is potentially dangerous to all personnel, a Klaxon horn alarm will be sounded and instructions will be given on the public address system by the Emergency Dispatcher. Appendix E provides a list of alarms and preferred responses. Building personnel shall not reenter their work areas until Hazards Control reports that it is safe to do so. An individual injured on Laboratory property is required to report to the Medical Department regardless of how slight the injury appears to be. An injured person shall be taken to Medical only in an official ambulance, except when the injury is very minor.

LPTR DISASTER CONTROL PLAN

A. INTRODUCTION

1. Scope

This plan contains procedures related specifically to the Building 281 Reactor Facility. Familiarity with the LLL Disaster Control Plan is a prerequisite to understanding this LPTR Disaster Control Plan, especially since some of the general procedures contained in the Laboratory plan are not repeated here. Revisions of this Disaster Control Plan are the responsibility of the Reactor Supervisor and will be reviewed annually and revised whenever continued operation reveals the need for changes or improvements in emergency procedures.

2. Type of Hazards Involved

The hazards involved in the operation of the LPTR can be grouped into two general categories: (a) hazards resulting from equipment failure, and (b) large scale hazards from catastrophic events.

- a. Hazards resulting from equipment failure cover many possibilities. Most serious of these are fuel element failure, loss of coolant, or failure of experimental facilities. Some of these events, individually or compounded, could result in extensive contamination, possibly extending throughout the entire reactor building. The extent of the accident would depend on a number of factors, such as the amount of excess reactivity introduced into the core, and the degree to which an aluminum-water reaction might occur. Thus, the type of failures mentioned above could cause (roughly in ascending order of seriousness):

- (1) Flooding of the reactor equipment pit and/or floor.
- (2) Loss of water shielding from reactor tank.
- (3) Spills of radioactive materials and dangerous beams of radiation from experimental failures.
- (4) Highly contaminated demineralized and/or pool water from fuel element rupture.

- (5) Melting of the fuel elements with subsequent fuel element rupture, contamination of the pool, and an aluminum-water reaction with expulsion of some pool water, fission products, and core materials, into the building.

Although the probability of occurrences of the above is extremely small, the reactor was necessarily designed with the possibility of their occurrence in mind. Accidents in this category would almost certainly lead to reactor shutdown and sealing of the building, pending repair of the component or system that failed. In some instances, evacuation of the building would be necessary with subsequent decontamination before operations could be resumed.

- b. Hazards resulting from catastrophic events could involve the rupture or destruction of the reactor building with simultaneous destruction of the fuel elements, thus releasing the fission products and active core materials into the atmosphere. Although building destruction would be due to war-caused or natural disasters, release of materials into the atmosphere would not necessarily follow.

However, the LPTR Safety Analysis Report assumes that a major fraction of the fission products from a total aluminum-water reaction would be released to the atmosphere. The related cloud diffusion study and fallout problem is naturally complicated by the degree or extent of the damage and releases. An on-the-spot evaluation would be required; the meteorological service at LLL would be called in to provide the necessary data.

3. Emergency and Disaster Classification

To guide the LPTR Disaster Control Organization in determining the classification of the situation and the advisability of activating the LLL Disaster Control Organization, emergency situations are evaluated according to the following definitions found in the LLL Disaster Control Plan.

- a. Emergencies--Emergencies are accidents that endanger life and property. Emergencies do not interrupt wide segments of Laboratory operations, and the general public or the local community does not become involved. Emergencies are reported to the Fire Department and are referred, in turn, to the person or persons responsible for controlling the particular type of accident. Examples of emergencies are hazardous spills of water, flammable materials, toxic materials, or radio-nuclides;

accidental explosions, fires or suspected fires; and injuries requiring first aid.

- b. **Disasters**--Disasters are serious accidents that cause injuries or extensive damage and threaten to cause additional injuries and damage. Generally, accidents that can cause serious off-site effects and create public relations problems with the community are classified as disasters. The entire Disaster Control Organization is available during disasters to control the extent of injuries and property damage. The plan outlined in this document is concerned with disasters or serious accidents that involve the total Disaster Control Organization.

B. ORGANIZATION AND RESPONSIBILITIES

1. LPTR Staff and Organizational Structure

The organization of the normal operating staff of the LPTR is shown by Fig. 1, in OSP 281. This staff may operate as much as three shifts a day, five days a week.

2. LPTR Disaster Control Organization

The Disaster Control Organization for the LPTR is shown by Fig. A-1. This Disaster Control Organization is the same as the Disaster Control Organization shown in the LLL Disaster Control Plan. The entire LPTR Disaster Control Organization functions under the Deputy Disaster Control Director, as defined in the LLL Disaster Control Plan.

3. Personnel Assignment

- a. **Deputy Disaster Control Director** and his alternates are listed in the LLL Disaster Control Plan Supplement 2, Emergency Call List. LPTR personnel assignments are listed in Paragraph F of this document and are on file with the Emergency Control Coordinator. The Reactor Supervisor assumes the functions of the Emergency Control Coordinator until the Senior Fire Officer arrives. In the event that the Reactor Supervisor is not available, the Shift Supervisor will act in his place.
- b. **Reactor Supervisor**

If present, the Reactor Supervisor, or his assistant, takes charge of the reactor facility. If both men are absent, the Shift Supervisor assumes the Reactor Supervisor's responsibilities.

Fig. A-1. Disaster Control Organization for LPTR.

c. Hazards Control Department

The senior member of the Hazards Control Department on duty in the building assumes responsibilities of the Hazards Control Area Representative for Area 4 until the Area Representative or his alternate arrives.

d. Reactor Operators

The Reactor Operators on duty will retain their normal functions in the Disaster Control Organization. If the Shift Supervisor becomes disabled, a Reactor Operator will assume his functions.

4. Duties of Personnel

a. Deputy Disaster Control Director

The Deputy Disaster Control Director's duties are outlined in the LLL Disaster Control Plan.

b. Emergency Control Coordinator

The Emergency Control Coordinator directs all on the scene efforts to bring the disaster condition under control until the Deputy Disaster Control Director Arrives. The Emergency Control Coordinator's specific duties are stated in the LLL Disaster Control Plan.

c. Reactor Supervisor

The Reactor Supervisor directs all on the scene efforts to bring the disaster under control until the Senior Fire Officer arrives. The Reactor Supervisor will take immediate action to:

- (1) Initiate the Plan of Operation (Paragraph C).
- (2) Brief the Senior Fire Officer on the radiological hazards, events leading to the disaster, and the action taken so far when the Laboratory emergency control forces arrive. Advise him of the condition of the emergency decontamination system, the emergency scrubber system, and the emergency fire-foam system.
- (3) Order the evacuation of personnel not required in the emergency control operation. Assemble Building 281 personnel at a safe location for the purpose of briefing, head count, and decontamination.

- (4) Direct the setup of contamination check points for personnel monitoring.
- (5) Request security officers to restrict access to emergency personnel required for the emergency control operation.

d. Hazards Control Department

The Area Representative coordinates the safety discipline and technical skills of Hazards Control Department personnel to achieve the emergency control objectives judged necessary by the Emergency Control Coordinator. The specific duties and responsibilities of the Hazards Control Area Representative are stated in the LLL Disaster Control Plan.

e. Reactor Operators

Upon concurrence of a disaster situation, Reactor Operator in the absence of the Reactor Supervisor will:

- (1) Initiate the Plan of Operation (Paragraph C).
- (2) Summon the Reactor Supervisor and inform him of the situation.
- (3) Continue to work under the direction of the Reactor Supervisor as he directs the remaining emergency control operations.

f. Other Personnel in Building 281 Area

- (1) Personnel in the reactor building evacuate by the nearest exit and report immediately to the Control Point.
- (2) Personnel in the laboratory building remain at their stations until directed elsewhere. In the event that the building criticality horn sounds, evacuate the building by the nearest exit and report to the Control Point.

5. Authorization Required for Permitting Emergency Exposures

The following designated emergency response personnel may authorize exposures for emergency situations in the Building 281 area of the Laboratory within the limitations and for the purposes indicated. Maximum dose limits are given in rem.

Emergency Response Personnel	To Rescue Personnel (rem)	To Protect Community (rem)	To Protect LLL Property (rem)
Reactor Supervisor	100	25	12
Shift Supervisor	100	25	3
Disaster Control Director	150*	150*	25
Deputy Disaster Control Director	150*	100	25
Emergency Control Coordinator <u>plus</u> Health Physicist	100	100	25
Any Hazards Control Health Physicist	100	25	12
Shift or Building Monitor <u>plus</u> Senior Fire Officer	100	25	12
Any Hazards Control Area Representative	25	25	12
Emergency Control Coordinator	25	25	12
Shift or Building Monitor	25	3	3
Senior Fire Officer	25	3	3

*In arriving at the upper limit of 150 rem, consideration was given to the fact that an underestimation of the radiation fields could differ as much as 35% from the instrument readings. If such an underestimation were made, the maximum exposure would be 200 rem. This should be the upper limit received by any rescuer under any conditions.

C. PLAN OF OPERATION

The following course of action is to be followed in the case of a major nuclear excursion that has released large quantities of radioactive fission products into the reactor building. In the event of another type of disaster appropriate modifications to the Plan of Operation should be made as conditions indicate. A summary diagram of action to be taken by LPTR Operations Personnel is shown in Fig. A-2.

Fig. A-2. Summary Diagram of Emergency Action to be Taken by LPTR Operations Personnel.

1. Action to be Taken by LPTR Operations Personnel

The emergency control actions to be taken in a sequence appropriate to the nature of the accident by the operations personnel.

- a. Scram reactor. If shutdown is not complete, insert rod drives.
- b. Seal building.
- c. Sound evacuation horn in reactor building. Order evacuation of reactor building over P.A. System. Direct evacuees to report to the hallway outside the control room. This will be the initial control point.
- d. Dial 7333 and alert the LLL Disaster Control Organization. This notification should include what has happened, where it has happened (room number), the condition of the reactor, the condition of the reactor containment building, nature of personnel injuries (if any), fire conditions (if they exist), and your name. Direct a member of the operating staff to stand by outside the Building 281 area and direct emergency forces to the emergency. See Fig. A-3 for floor plan of Building 281 and immediate surroundings.
- e. Determine the radiation level in control room. Activate Emergency Decon System if necessary. If radiation levels are tenable, retain the disaster control center in the control room. Determine the radiation levels at all reactor building monitor stations.
- f. Make head count in an attempt to identify any unaccounted for personnel. Scan building with TV and use the intercom or P.A. System to repeat evacuation instructions to reactor building personnel. If reactor shutdown is assured and radiation levels are tenable, enter reactor building (equipped as outlined below) and survey to see that all personnel have evacuated.
- g. Start the emergency recovery immediately if injured personnel are still in the reactor building. Using several instruments, determine radiation levels in the building. Do not rely 100% on the accuracy of any one instrument. Estimate the length of exposure time required for the recovery team to accomplish the rescue. Maximum actual exposure time during the emergency rescue is based on

Fig. A-3. Floor plan of Building 281 and associated structures.

consideration of a reasonable exposure dose, estimated length of exposure time, and the expected result. Use breathing apparatus as necessary and do not work alone. Under certain conditions, rescue is impossible. See Paragraph B.5. (exposure limits).

- h. Using the P.A. System, direct all personnel in the laboratory portion of Building 281 to remain where they are and to close windows. Direct operations personnel to turn off air-conditioning equipment. Check windows. Specifically request that all unnecessary personnel leave the control room.
 - i. If necessary, call for additional help from Hazards Control personnel to handle decontamination duties and to help at check points.
 - j. Start a supplementary log of reactor power levels and all building radiation levels, plus wind direction and speed, and all data pertinent to the disaster.
 - k. If the sustained radiation level in the control room exceeds 500 mR/hr, move the emergency control center to the main office, Room 1173, or to a disaster field command post outside the building. Telephone master and P.A. System is available in the office. A master station for hallway radiation levels is available in the lobby.
 - l. If radiation levels at the east end of the hall exceed 100 mR/hr, arrange for evacuation of Building 281 personnel. Check wind direction and speed before leaving control room. The Emergency Control Coordinator is also to direct personnel to a control assembly point.
2. Action to be Taken by Laboratory Emergency Control Forces

The sequence of emergency actions to be taken by Laboratory emergency control forces reporting to the LPTR facility is as follows:

- a. Enter only through the west main door (main office entrance). An area monitoring station is located just inside the main door on the right side of the entrance lobby. This monitoring system indicates the radiation level at three points in the long hallway leading to the reactor building.

- b. Before proceeding, check the radiation levels in hallway. If any monitor stations are tripped (red light), check level on master meter, and proceed with caution using high-level instrumentation.
- c. Check with Reactor operations personnel to determine the nature of the emergency and the actions that have already been taken. The reactor building will seal automatically from high radiation and it is imperative that this containment seal not be broken by emergency forces unless absolutely necessary.
- d. In the event of a fire in the reactor building proceed as follows:
 - (1) If a high-radiation condition does not accompany the fire, then, at the direction of the Senior Fire Officer, the reactor building may be opened to allow access by the fire control forces with equipment, hoses, etc.
 - (2) In the event of fire in the reactor building accompanied by high radiation, the building seal integrity must be maintained and access can only be made through the air lock. Equipment would then be limited to the fire extinguisher, fire hose in the building, and any portable equipment that can be carried in through the air lock door. All personnel entering must be equipped with full protective equipment including self-contained breathing apparatus. Stay-time limits will be set by the Emergency Control Coordinator and based on the radiation levels existing in the building.
 - (3) Activate the foam generator system only if fire can be contained in no other way.

D. EQUIPMENT

The following equipment and systems are available in Building 281.

1. Instrumentation on Console

- a. 48-Point Annunciator Panel
- b. Radiation Monitoring System

- (1) Area monitoring stations
- (2) Primary water monitor
- (3) Secondary water monitor
- (4) Demineralized water monitor
- (5) Fission product water monitor
- (6) Stack monitor
- (7) Duct monitor
- (8) Stack NMC
- (9) Balcony NMC
- (10) FNIF NMC

c. Nuclear Instrumentation

- (1) Log N recorder
- (2) Log count rate recorder
- (3) Period recorder
- (4) Linear level recorder
- (5) N-16 system

d. Temperature

- (1) Primary coolant (2)
- (2) Secondary coolant (2)
- (3) Pool (1)
- (4) FNIF (6)

e. Conductivity

- (1) Primary
- (2) Demineralizer

f. Water Level

- (1) Pool low level
- (2) Pool high level
- (3) Pool manometer
- (4) Sump high level
- (5) Waste tank high level (2)
- (6) Storage tank high level
- (7) Secondary low level

g. Meterological

- (1) Wind direction
- (2) Wind speed
- (3) Dry bulb temp
- (4) Wet bulb temp
- (5) Microbarograph

2. Communication Equipment

a. Telephone System (LLL)

Extensions:	7321	7325
	7322	7326
	7323	7327
	7324	7328

Master in control room, office (Room 1173).

b. Intercom and Page System

Master in control room, Room 1325, and two in office, Room 1173.

Eleven (11) stations in reactor building.

One (1) station in hot chemistry laboratory, Room 1311.
One (1) station in air lock.
One (1) station in scrubber building.
One (1) station at cooling tower.
Thirty five (35) stations in lab offices and trailers.
Page to all stations.
Outside speakers.

c. Television System

- (1) Core TV--fixed camera with zoom lens.
- (2) Reactor building--two camera with remote pan, tilt, and focus, displayed on one monitor.

3. Automatic Alarms

- a. Security (air lock door).
- b. Fire alarm, laboratory increment (three zones) only.
- c. Fire alarm, reactor building.
- d. Pressure alarm, reactor building.
- e. High radiation alarm, reactor building.
- f. Low pool level alarm.
- g. Criticality alarm.

4. Hall Radiation Monitoring Systems

- a. Master in lobby opposite office.
- b. Stations:
 - (1) Lobby
 - (2) Middle of hallway
 - (3) Opposite control room

5. Emergency Systems

a. Emergency Power Systems (50 kVA MG Set)

- (1) Emergency lights (control room, hallway, reactor room).
- (2) Primary coolant pump.
- (3) Reactor console instruments.
- (4) Emergency scrubber system.
- (5) Hot chemistry laboratory hood exhaust.
- (6) Receiving room hood exhaust.
- (7) Bio-Physics hood exhaust and instruments.
- (8) Counting room instruments.
- (9) Foam generator.
- (10) Duct blower.
- (11) Stack fan.
- (12) All radiation monitors.
- (13) Sump pump.
- (14) Hallway emergency panel.
- (15) Building seal relays.

b. Emergency lights, four charematic lights

c. Building Seal System

- (1) Air lock.
- (2) Escape doors (2).
- (3) Truck door.
- (4) Butterfly valves (3).
- (5) Diversion valve in air conditioning system.

- d. Emergency Decontamination System.
 - e. Emergency Scrubber System.
 - f. Foam Generation System.
6. Hazards Control Equipment
- a. Normal monitoring equipment
 - (1) Three (3) high-range Junos.
 - (2) Three (3) low-range Junos.
 - (3) Three (3) E-112 Geiger counters (beta, gamma).
 - (4) Three (3) alpha meters.
 - (5) Two (2) Keithley Micro-Micro Ammeter Ionization Chamber Continuous Air Monitors, LLL manufacture (flow approximately 1 cfm).
 - (6) Three (3) NMC Air Monitors.
 - (7) Tracerlab modified water monitor.
 - (8) Two (2) hand and foot counters (LLL manufacture).
 - (9) Filter Queen air samplers.
 - (10) Two (2) LLL neutron counters (Remeter).
 - (11) Three (3) Neutron accident dosimeters.
 - b. Emergency Equipment
 - (1) Two self-contained breathing apparatuses mounted on corridor wall outside LPTR Control Room.
 - (2) ABC, BC, fire extinguishers, hallway, and Reactor building at crane supports.
 - (3) Fire hoses in hallway and reactor building.
 - (4) Dry standpipe for fireman's hose through reactor building wall which do not allow escape of contaminants. (southwest quadrant and air lock.)

E. ORIENTATION ILLUSTRATIONS

Figures A-4 through A-10 show various features of the equipment installed for reactor control or for emergency use. Fig. A-11 gives a general layout of the facility on which the locations of these features may be located.

F. REACTOR SUPERVISOR AND ALTERNATES

The Reactor Supervisor and his alternates are:

	<u>LLL Ext.</u>	<u>Home</u>
Wade Richards - Reactor Supervisor	7321	443-4979
Gordon V. Pefley, Physicist	7321	447-9414

These people also act as Emergency Control Coordinator until the Senior Fire Officer arrives. This list is on file with the Emergency Dispatcher. The Emergency Call List (APPF) list additional reactor related personnel.

G. REACTOR OPERATORS

	<u>LLL Ext.</u>	<u>Home</u>
Joe Prindle (Supervisor Operator)	7321	537-8330 (Castro Valley)
Will Burns	7321	447-2489 (Livermore)

Fig. A-4. LPTR Control Console in Room 1325. Reactor controls and instrumentation are located here; this console is manned by operations personnel at all times when the reactor is in operation.

Fig. A-5. General view of reactor building looking southeast.

Fig. A-6. Escape door (one of two) located in reactor building on east side. These doors serve as a secondary escape route in case the reactor air lock route is not available. OPERATION: Door gaskets are deflated by manual push buttons located INSIDE the doors. Deflation takes about 20 sec. Doors must be closed after exit to allow automatic inflation of seals.

Fig. A-7. Automatic air scrubber system for reactor building (at north side of reactor building). Automatically operates when the building is in the sealed condition if internal pressure becomes higher than outside pressure. CAUTION: Filters and scrubbers could become radioactive in the case of an accident.

Fig. A-8. Recirculation and decontamination system for reactor building (outside south side of reactor building). Manually started by operations personnel in Room 1328. Removes particulate contamination from reactor building air while building is sealed. CAUTION: Filters could become radioactive following an accident.

Fig. A-9. Area Radiation Monitor Panel. This panel is located on the wall at the main entrance (west) to Building 281.

Fig. A-10. Evacuation Horn Panel. This panel is located on the wall at the main entrance (west) to Building 281.

Fig. A-11. Building Alarm Panel. This panel is located on the wall at the main entrance (west) to Building 281.

LPTR REACTOR SAFETY COMMITTEE

A. PURPOSE

A Reactor Safety Committee exists to provide a multidisciplined advisory group to the Reactor Supervisor for experimental programs, and operational matters that affect reactor safety.

B. STRUCTURE

The Reactor Supervisor is the Committee Chairman with members of the committee selected for their knowledge of reactor systems and associated hazards or nuclear physics experiments and associated hazards. Membership is recommended by the Reactor Supervisor, and the Physics Section Leader, and is approved by the Plant Manager.

Members of the committee are authorized to invite other experts to be specific consultants to the committee with the concurrence of the Committee Chairman or the Physics Section Leader.

The committee is structured so that at least half of the members are independent of all programmatic interests within the Radiochemistry Division*. Agenda for each committee meeting is the responsibility of the Reactor Supervisor. Minutes for each meeting are the responsibility of the executive secretary, who shall be independent of all programmatic interest. The minutes are reviewed and signed by attending committee members.

C. FUNCTIONS

The reactor is governed by operational limits, procedures, and instructions that must be at least as restrictive as the envelope of the Safety Analysis Report (UCRL 50147). The Safety Committee is responsible for reviewing reactor modifications, reactor operational procedures, and any technical changes to an operational procedure that could cause the reactor to be outside of the restrictive envelope established by the Safety Analysis Report.

Procedures describing experimental set-ups in the reactor building that have any interaction with the reactor shall be reviewed by the Safety Committee.

D. QUORUM

Each Reactor Safety Committee member has full voting rights. One-half of the membership with a majority of the non-programmatic members present is required to be present to carry out a full review.

*Independent of programmatic interest shall mean not subject to Radiochemistry Division supervision.

LPTR SAFETY REGULATIONS FOR NON-OPERATIONS PERSONNEL

A. ENTRY AND EXIT

All personnel not on the Authorized Access List must obtain permission from Reactor Operations prior to entering the Reactor Containment Building (dome). Reactor Operations will pass on pertinent safety information. Failure to follow proper check-in procedures could result in an inadvertent radiation exposure to yourself or possible damage to a reactor safety system.

When the reactor is secured and unattended by Reactor Operators, the Reactor Dome must not be entered without prior clearance from the Security Department (Ext. 7222). Only personnel on the Off-Hours Access List held by the Security Department (Ext. 7222) will be admitted. A sign hung across the air lock door indicates when the Reactor Room is secured. Call Security prior to entry and after exiting the air lock. The Off-Hours Access List will be compiled and maintained by the Reactor Supervisor.

Just prior to entering the Reactor Room Air Lock, push the appropriate identification button on the access board to disclose your presence. Clear the button on the access board upon leaving the Reactor Room. The reactor air lock is the only regular entry and exit point. The emergency doors are for emergencies only, see Paragraph G below.

B. VISITORS

Occasionally a visitor will desire access to a given experimental area. His presence is the responsibility of either the operations staff (Reactor Operator) or the non-operations person accompanying him. This responsibility must be clarified with Reactor Operations each time the person enters the reactor. This visitor will not be left unattended until such time as he understands these regulations, reviews them, and signs them.

C. TOP STRUCTURE AND EQUIPMENT PIT

The Reactor Top Structure and the Reactor Equipment Pit are off limits to all non-operations personnel unless specific authorization has been received from the Reactor Operator.

D. OPERATION OF EQUIPMENT

All reactor systems shall be operated only by qualified Reactor Operators. This includes thermal column doors and pneumatic rabbit tubes. Any shielding change must be concurred with by Hazards Control and by the reactor operations staff to prevent an accidental exposure.

E. TARGETS, RADIOACTIVE SOURCES

The reactor building contains many radioactive sources and targets. These are clearly labeled and should be treated as hazardous materials. All radioactive samples going into or out of a reactor irradiation facility will be handled by the Reactor Operator with Hazards Control assistance as necessary.

F. EMERGENCY AND SAFETY EQUIPMENT

Each individual shall be acquainted with the location and the use of routine emergency and safety equipment such as communications, fire extinguishers, emergency exits, respirators, and radiation monitoring instruments, which are located along the walls of the building. Any unexpected change in an experimental setup such as change in background radiation or loss of flux should immediately be called to the Reactor Operator's attention. Pay particular attention to alarms, ropes, signs, or flashing lights in the hallways or in the reactor dome. Water on the floor should be treated as if it were contaminated. Just outside the air lock is located a hand and foot counter. Monitor your hands and feet after each exit from the Reactor Dome.

G. EMERGENCY EXIT

The Reactor Containment Building may become "sealed" while routinely at power, or it may become sealed while you are alone on off hours. This condition of "Building Seal" can be automatically initiated by any of a number of monitoring circuits or manually by the Reactor Operators. It may be followed by intermittent horn blasts in the Reactor Containment Building 15 seconds after the seal is initiated. All such situations shall be considered true emergencies unless otherwise announced over the P.A. System.

The Reactor Containment Building (Dome) is designed so that the reactor and its components can be isolated from the outside environment if an abnormal condition develops. The building is sealed by closing butterfly valves in the air conditioning ducts and the exhaust duct on the roof. At the same time a rubber gasket is inflated on one of the

air lock doors. The North and South escape doors and the truck door gaskets are normally inflated at all times. If conditions are such that you cannot use the air lock, then exit via one of the two escape doors. Be sure and close the escape door behind you.

1. Remember the air lock and escape door gaskets are inflated and take 30 seconds to deflate. No amount of tugging will open the door until the gasket is fully deflated.
2. The outer air lock door gasket must deflate while the inner door gasket inflates, again taking 30 seconds. This double door arrangement prevents building air from escaping to the outside.
3. When emerging from the air lock or escape doors, proceed directly to the hand and foot counter located outside of the air lock and check your hands and feet for contamination.
4. If you have any detectable contamination, do not proceed further. Call to the Reactor Operator in the Control Room for assistance. Do not leave the area unless instructed by operations personnel.
NOTE: On off hours, when the Control Room is not manned, walk to the wall phones in the Control Room and call Ext. 7222. Ask the guard to contact the Health and Safety Technician on duty to assist you. When he arrives ask him to help monitor you for contamination and call the first person on the Call List posted on the north side of the reactor console.
5. If you have no detectable contamination on your hands or feet, check in with the Control Room for instructions. Do not leave the area unless instructed by the Reactor Operator or the Health and Safety Technician.

Name (Print) _____

Organization _____

Date _____

LPTR SAFETY REGULATIONS FOR NON-OPERATIONS PERSONNEL

CHECK ITEMS

The person going unescorted into the reactor room must have knowledge of the following items:

- (1) Front hallway monitoring system _____
- (2) Self monitoring technique _____
- (3) Control room items _____
- (4) Calls for assistance _____
- (5) Reactor Room hazards (top, pit, caves, ropes, lights) _____
- (6) Reactor Room emergency equipment _____
- (7) Activation of emergency escape system _____

*Only those people requiring routine entry into the Reactor Room.

I have read and understood the above instructions.

SIGNED _____

Approved By: _____
Reactor Supervisor

Date: _____

Please return this sheet to LPTR, L-540

REPORTING OF REACTOR INCIDENTS OR ACCIDENTS

Reactor incidents or accidents that fall into one or more of the categories listed below shall be reported by the Reactor Supervisor to the Plant Manager and the Programmatic Division Leader.

1. Conditions that result in exceeding the reactor operating limits or that result in safety system settings less conservative than the operating limit value, or that result in violation of a limiting condition for operations established in the operating limits.
2. Abnormal degradation of any of the boundaries which are designed to contain radioactive materials or release of radioactive material which indicate potential operating problems or significantly interferes with operations, or any uncontrolled or unplanned release of radioactivity from the reactor building.
3. Any uncontrolled or unanticipated changes in reactivity which could significantly and adversely affect safety of operations.
4. Incidents or conditions which prevent or could prevent the intended performance of safety, control, or shutdown systems.

Such an incident or accident shall be reported immediately by telephone by the Reactor Supervisor to the Plant Manager and the Programmatic Division Leader followed by a written report within 24 hours. The verbal report shall include:

1. Incident or accident description,
2. Safety implications,
3. Actions taken to determine the cause of the incident or accident, and
4. Corrective action proposed.

Based on this information, a determination will be made by the appropriate programmatic division and the Plant Manager as to whether operations can be safely resumed.

The written report shall include in addition to the above information:

1. An analysis and evaluation of the safety implications, and
2. Corrective action proposed to prevent similar occurrences involving similar components or systems.

EMERGENCY PROCEDURES

If total evacuation of the building is necessary because of fire, toxic gas release, or radioactive contamination, notification of building occupants is accomplished by using the building paging system and the emergency evacuation system controlled from the Fire House. If evacuation from the containment building only is required, an additional alarm and paging system is provided.

EMERGENCY EVACUATION SYSTEM

This system uses a Klaxon horn sound which can be transmitted over the paging system by either the Fire or Police Departments. Verbal messages and specific instructions can also be transmitted.

EMERGENCY PROCEDURES

1. EVACUATION - REACTOR CONTAINMENT BUILDING

If the Klaxon horn sounds, leave immediately through the air lock doors first, or emergency exits if air lock doors are not accessible. The Reactor Containment Building may become "sealed" while routinely at power, or it may become sealed while you are alone on off-hours.

The building is sealed by closing butterfly valves in the air conditioning ducts and exhaust duct on the roof, and a rubber gasket is inflated on one of the air lock doors. The north and south escape doors and the truck door gaskets are normally inflated at all times.

This condition of "building seal" is followed by Klaxon horn blasts. If conditions are such that you cannot use the air lock, then leave via one of the two escape doors. Be sure and close the escape door behind you. Check hands and feet on hand/foot counter in hallway and stand by for instructions from the operator.

2. EVACUATION - LABORATORY AND OFFICE SPACES

Persons working in any section of Building 281, upon hearing the Klaxon sound, shall leave the building immediately by the nearest exit, and assemble for instructions and identification at the southwest end of Building 281, or at a secondary staging area announced over the paging system by the Fire Department.

Personnel are cautioned to keep as far away from the building as reasonably possible while proceeding to the staging area.

EMERGENCY CALL LIST - BUILDING 281

A. REGULAR WORKING HOURS

Should information concerning Building 281 be needed during regular working hours (Monday through Friday) the following list of persons is provided:

NOTE: This is not an automatic call list for Response to an emergency. The Fire Dispatcher has an emergency Response list.

<u>Name</u>	<u>Function</u>	<u>Extension</u>
W. Richards	Reactor Supervisor	7321
G. V. Pefley	Assistant Reactor Supervisor	7321
J. Prindle	Supervisory Operator	7321
W. Burns	Reactor Operator	7321
V. G. McIntosh	Building Supervisor	7909
L. G. Mann	Physics Section Leader	8721
R. Hoff	Radiochemistry Division Leader	8711
W. L. Abernathy	Alternate Building Supervisor	8789
<u>Hazards Control</u>		
C. W. Childers	Building Health & Safety Technician	7326
L. Davis	Building Health & Safety Technician	7326
R. Trolan	Area Representative	7326

B. OFF-SHIFT HOURS

Should information concerning Building 281 be needed during off-shift hours, the following list of persons and their home telephone numbers are provided. Names are listed in the order of familiarity with current status of operations in the building:

<u>Name</u>	<u>Function</u>	<u>Home Phone</u>
W. Richards	Reactor Supervisor	443-4979
V. G. McIntosh	Building Supervisor	447-3125
L. G. Mann	Physics Section Leader	447-3672
Hazards Control	Hazards Control Technician - page at	7222
	Maintenance Machinist - page at	7391

C. REACTOR OPERATIONS

Should information concerning the status of the reactor, containment building or control room be needed, call one of the following persons:

<u>Name</u>	<u>Function</u>	<u>Home Phone</u>
W. J. Richards	Reactor Supervisor	443-4979
G. V. Pefley	Assistant Reactor Supervisor	447-9414
J. A. Prindle	Supervisory Operator	537-8330
W. K. Burns	Reactor Operator	447-2489

NOTE: Names and home phone numbers are also listed on a card located on the left side of reactor console.

NOTICE

"This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately-owned rights."