Tritium Production and Release to Groundwater at SSFL

Safety Review Report

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1.0 INTRODUCTION

Discovery of the presence of tritium, a radioactive material, in the environment at the Santa Susana Field Laboratory (SSFL) has raised questions as to the origin and possible extent and magnitude of this material. While none of the occurrences found have approached applicable regulatory limits, public concerns have been raised that require attention. This report is presented in response to those concerns. This report explores and discusses the origin and concentrations of tritium in the environment, due to natural and artificial causes; local and global considerations; regulatory limits and applications; and specifically, the production, release, and migration in groundwater of tritium due to operations at SSFL.

Tritium is a radioactive isotope of hydrogen and is produced naturally by cosmic–ray interactions in the upper atmosphere, and artificially by a variety of neutron reactions in nuclear reactors. The main production reaction in nature is $^{14}N(n,t)^{12}C$ in which a nitrogen atom (^{14}N) in the atmosphere is struck by a high–energy neutron (n) from the sun releasing a "triton" or tritium nucleus (t) and leaving an atom of stable, ordinary carbon (^{12}C). After production (or release) in the atmosphere, tritium oxidizes or exchanges with normal hydrogen in water vapor to form tritiated water (HTO), and becomes part of the Earth's global waters, in rain and snow, groundwater, streams, rivers, lakes, and the oceans. Because of its relatively short half–life (about 12 years) tritium activity decays away, and is practically nonexistent in "old" water, such as in glaciers, deep groundwater, and the ocean depths.

In this report, measurements of the tritium are discussed using the historical unit of activity, the curie, and its fractional forms, the microcurie (one–millionth of a curie, 10^{-6} or 0.000 001 Ci) and the picocurie (one–trillionth of a curie, 10^{-12} or 0.000 000 000 001 Ci). A picocurie of tritium is equivalent to 2.22 transformations of H–3 atoms to He–3 (helium–3) atoms per minute. When discussing tritium in water, the unit of volume used is generally the liter (abbreviated L), which is just slightly larger than a quart.

During atmospheric testing of nuclear (fusion) weapons (H–bombs), large amounts of tritium were released into the lower atmosphere, completely overshadowing the natural production rate. The tritium released in the weapons tests has largely decayed away and now the average tritium content of surface waters is approximately twice what would be expected from the natural production by cosmic rays.

The tritium concentration in environmental water is shown in Figure 1–1, which plots the tritium concentration in precipitation (rain, snow, sleet, and hail) and in Figure 1–2, for surface (river) water. The precipitation was collected at six locations in the western United States. These collection stations are at Portland, Menlo Park, Santa Maria, Flagstaff, Albuquerque, and Denver, and are part of a global network coordinated by the International Atomic Energy Agency (IAEA) and the World Meteorological Organization (WMO). The periodic peaks and valleys in the plot of tritium in precipitation are due to the mixing, in the spring, of stratospheric air, where most of the natural tritium production occurs (and also where much of the high–yield nuclear weapons–test radioactivity was
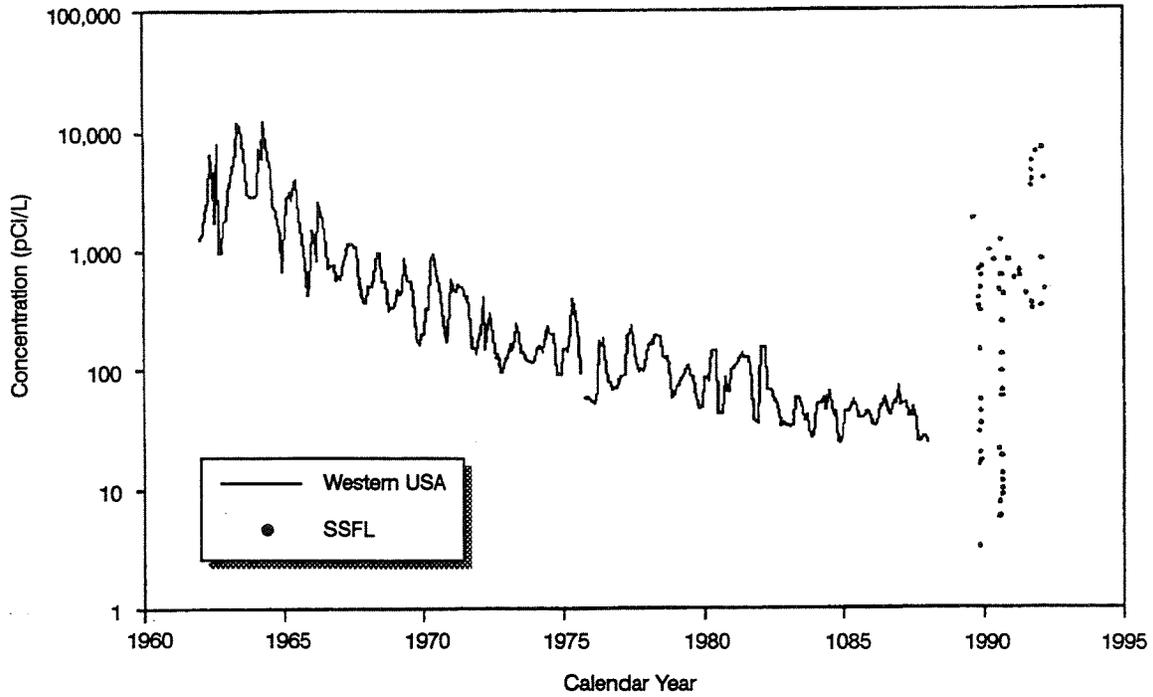
Figure 1-1. Tritium in Precipitation in the Western United States (IAEA/WMO) and SSFL Groundwater
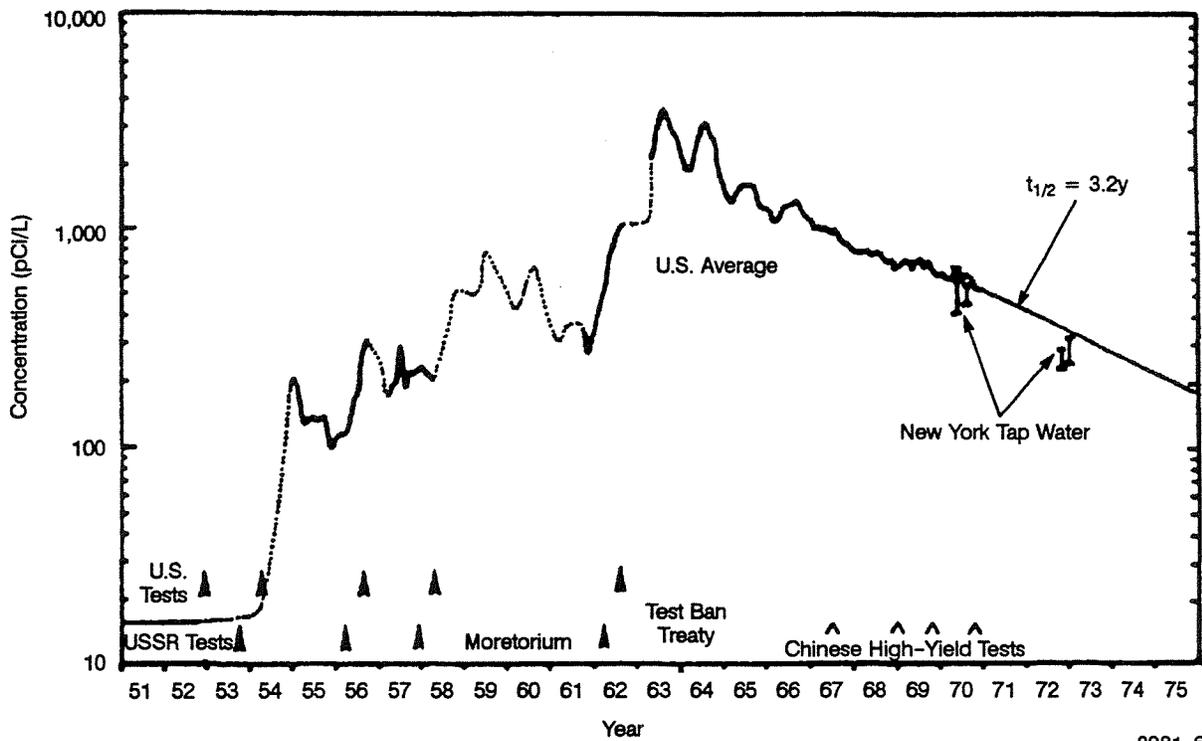
Figure 1-2. Tritium in Surface Water in the United States (USGS)

placed) with tropospheric air, which is the major source of precipitation. (In this plot, and many more to come, the common logarithm (log) or base −10 log, is used to compress the range of the data displayed. Thus, in this figure, the tritium concentration in rainfall initially ranges between about 1,000 and 10,000 pCi/L and slowly declines to about 20 pCi/L.)

Since the identification of tritium in environmental water at SSFL, many water samples from groundwater wells have been analyzed for tritium. A representative display of these results is shown as the small dots in the right−hand portion of Figure 1−1. SSFL samples have also included surface water, rain water, and process water, and show concentrations approaching the values experienced in widespread rainfall during the mid−1960s. The results of sampling and analysis for tritium in water at SSFL are discussed in detail in Sections 6 through 9 of this report.

The surface water samples (Figure 1−2) were collected and analyzed by the U.S. Geological Survey (USGS) in a wide−spread survey of tritium in the environment in the United States, and the analytical results clearly show the effect of fusion weapon tests on the tritium content of surface water. Extrapolation of the decreasing concentration from the end of 1975, with an effective half−life of 3.2 years (faster than the radioactive decay of tritium because of dilution by ocean water), to the end of 1991 shows a remaining activity of only about 6 pCi/L from the weapons tests. The natural concentration [prior to the start of fusion weapon (H−bomb) testing in 1952] is about 14 pCi/L. Thus, the total tritium concentration expected for surface water is 20 pCi/L, which is in good agreement with analytical results at SSFL (Sections 6 through 9).

The impact of nuclear weapons testing on the environmental inventory of tritium, and the significant differences in tritium concentration in environmental waters, depending on their source, is shown in Figure 1−3. This figure shows the tritium in various compartments of the environment, resulting from nuclear weapons testing. These curves were calculated at Oak Ridge National Laboratory (ORNL) for global averages, based on testing through 1974. Because of the global averaging, the maximum concentrations are not as great as shown for local concentrations in Figures 1−1 and 1−2.

Atmospheric tests of nuclear weapons that were conducted by China and France after 1974 are not included in these calculations.

At the SSFL, tritiated water ("tritium") in the environment may include, in addition to the natural tritium and tritium released by weapons tests, tritium that was produced as a side−effect of the operation of several experimental nuclear reactors.

These reactors were part of the early development of nuclear power for application to commercial electrical generation and for providing electrical power to instruments on Earth−orbiting satellites. Economic considerations and lack of an immediate application led to termination of these development programs. The test facilities have been decommissioned and most of the residual radioactivity remaining from nuclear operations has been removed, in a still ongoing process.
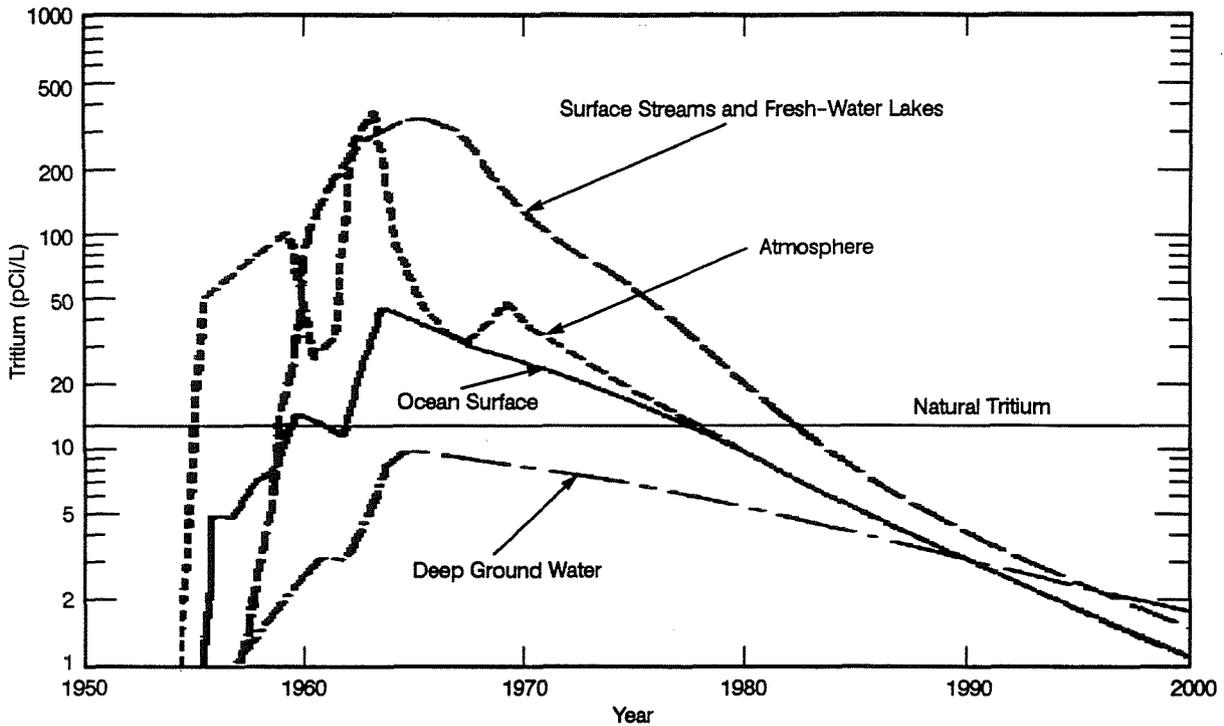
Figure 1-3. **Calculated Average Concentrations of Tritium in the Biosphere Due to Nuclear Weapons Testing**

Depending on the process and location of tritium production at SSFL, this tritium may have found its way into groundwater or surface water, or may have remained bound in the materials composing the reactor facilities and been disposed with that material as radioactive waste.

As a component of tritiated water, the tritium moves with the groundwater rather than being partitioned, or bound to soil particles, as most radioactive materials are. It is this partitioning that keeps most radioactivity in the soil localized, preventing its spread with groundwater movement, and thus tritium may be detected in groundwater without any confirmatory evidence of other readily detectable radionuclides, such as Co-60, Sr-90, and Cs-137.

Because of the low-energy radiation emitted by tritium, special techniques must be used to analyze samples for tritium content. These include liquid scintillation counting, conversion to a gas with subsequent detection in a gas proportional counter, and determination of the stable product of tritium decay (He-3) in a mass spectrometer. Liquid scintillation counting is the most commonly used method. Three different procedures for measuring tritium in water samples by liquid scintillation counting are available from commercial analytical laboratories. These are "total tritium," "EPA tritium," and "enriched tritium."

In the first procedure, "total tritium," an aliquot (small volume of water taken from the submitted sample) is placed directly in the liquid scintillation "cocktail," the mixture of solvent and liquid scintillator used for the analysis. Because this procedure includes all the radionuclides contained in the water sample, it must rely entirely on electronic pulse height/shape analysis for discrimination

against the non–tritium activity, and is subject to considerable variability and inaccuracies at low tritium concentrations. The most likely interference is probably from the beta emission of radon–222 daughters, since this radionuclide is prevalent in groundwater, and is not eliminated by filtration. Betas emitted near the wall of the vial and leaving the scintillation volume will deposit a similar amount of energy to that deposited by betas from tritium. The presence of Pb–210 (22–year half–life "great–grand daughter" of Rn–222) would cause interference that cannot be discriminated against. The beta emitted by Pb–210 has essentially the same energy distribution (E_{max} = 17 keV vs. 18.6 keV for H–3) and so the light pulses produced by these betas would be indistinguishable from those produced by betas from tritium. It is probable that this effect has caused many falsely high analytical results for tritium in old groundwater at SSFL. Radon (and Pb–210) concentrations in groundwater can range up to a few thousand picocuries per liter and some of this activity may be misidentified as tritium. Further complications are caused by the presence of mineral and organic materials, and living organisms, in the mixed sample/cocktail, due to chemiiluminescence and bioluminescence and variable quenching. The total tritium method is suitable for analysis of consistent, well–characterized samples with a high level of tritium, such as are routinely released from commercial nuclear power plants.

In "EPA tritium," the method prescribed for the routine testing of municipal water supplies (EPA 906.0), the sample is treated chemically to minimize carryover of various interfering impurities in the distillation of the sample, and a middle fraction of the distillate is taken for analysis. A very small aliquot (4–8 mL) is mixed with the cocktail for liquid–scintillation counting. (Because of the relatively high solubility of radon in water compared to air, it would be useful to assure elimination of residual radon after the distillation step, either by vacuum filtration, or by delaying counting for 2 to 3 weeks to allow the 3.82–day Rn–222 to decay). This method is intended for the purpose of demonstrating compliance with the EPA and State drinking water standard (20,000 pCi/L) rather than for measuring environmental levels of tritium (from zero to a few hundred pCi/L), and is adequate for analysis of water containing a few thousand pCi/L and more.

To measure concentrations of tritium approaching environmental background levels, it is necessary to increase the relative concentration by electrolytic enrichment. Alkaline electrolysis can increase the concentration of tritium in water by a factor of about 50, reducing the lower limit of detection to about 7 pCi/L for a 100–minute count. This method provides adequate sensitivity for actually measuring tritium content in natural waters and provides sufficient activity in water samples containing artificial tritium to greatly reduce errors due to background, chemiluminescence and bioluminescence, and quenching.

The electrolytic enrichment technique provides adequate sensitivity to permit detection of tritium in essentially any water: rainfall, surface water, groundwater, drinking water, Perrier. These levels are of academic interest in studying the distribution of tritium in nature, but can provide a misleading view when applied to investigation of artificial tritium at SSFL, by the mere "discovery" of tritium at low levels everywhere.

Since these levels are far below the levels of concern for health effects, and regulatory limits as well, emphasis in the tritium monitoring program has been placed on the standard Environmental Protection Agency (EPA) UCM method 906.0, which, when properly done, can provide useful results above a few hundred pCi/L.

Very small samples of water may be analyzed by the gas proportional counter method. High activity samples may be analyzed by mass-spectrometer measurement of the He-3 buildup.

Low levels of tritium have been measured in water from various sources at SSFL, ranging from environmental background concentrations to levels that are a few hundred times background. The present report investigates the possible sources of this tritium.

Tritium was produced in the SSFL test reactor materials, shielding, and adjacent soil as a consequence of reactor operation. Since no deliberate production was intended, and since the reactor power levels were relatively low, only small amounts of tritium were produced in this manner.

Because of the sealed nature of the reactor systems, little tritium was released from the reactor materials. Tritium would have been released to groundwater by the slow diffusion of tritium produced in the concrete shields and the adjacent soil.

Review of computer modeling of tritium concentrations in the groundwater, led to the conclusion by technical specialists, that the fractured rock structure at SSFL prevented the prediction of credible distributions. Therefore, estimates of the extent and magnitude of tritium in the groundwater must be based on groundwater sampling analysis.

A summary and interpretation of tritium analyses for a wide variety of water samples shows that most water is free of tritium, except for the extremely low concentrations measurable only by the enriched tritium method, which finds tritium in essentially all water. Artificial tritium has been measured by the standard EPA method in only a few locations, at levels that are below applicable limits.

2.0 DISCUSSION

2.1 PROPERTIES OF TRITIUM

Some physical and administrative properties of tritium are summarized in Table 2-1.

Table 2-1. Properties of Tritium

Radioactive Properties:	
Radiation: beta (normal electron)	
Maximum energy (keV)	18.6
Average energy (keV)	5.6
Maximum range (mg/cm^2)	0.59
Maximum range in air (cm)	0.49
Half-life (years):	12.346
Activity concentration in:	
Pure tritium oxide (T_2O) (pCi/L)	3.2×10^{18}
Pure tritiated water (HTO) (pCi/L)	1.6×10^{18}
Regulatory Limits:	
Occupational exposure limits:	
Concentration in water (pCi/L)	100,000,000
Release to the environment:	
Concentration in water (pCi/L)	3,000,000
Presence in drinking water supplies:	
Concentration in water (pCi/L)	20,000
Analytical Capability:	
Detection limit, liquid scintillation counting (pCi/L)	~200-500
Detection limit, with electrolytic enrichment (pCi/L)	~5-10
Detection limit, with gas proportional counting (pCi/L)	~90
Detection limit, helium-3 mass spectrometry (pCi/L)	~1,000

The low energy of the beta particles emitted by tritium both prevents this material from posing a significant biological hazard, per curie, and makes it difficult to detect. Even the X-radiation produced by the tritium betas is very low in energy and does not penetrate a significant distance in air. Because of this, tritium outside the body produces no direct radiation exposure. The beta particles do not have enough energy to penetrate the dead outer layer of the skin. While tritiated water vapor can be absorbed into the body from the atmosphere through the skin as well as through the lungs by inhalation, airborne exposure is not significant. Potential public exposure to tritium is generally limited to ingestion of tritiated water. This occurs only if the water is drunk. Only if tritium is taken into the body can a radiation exposure occur. This is the basis for the limits on occupational exposure, environmental release, and water supplies.

Tritium is used commercially in "glow-in-the-dark" watch dials and in self-powered emergency exit signs, such as are installed in airliners and public buildings. It is used in very large amounts as a component of nuclear weapons to "boost" the explosive power. It is used in research to tag organic molecules, as a tracer, and to date wines and ocean water, among other things. It will be required for the operation of the yet-to-be-developed fusion power reactors that are being investigated under current research programs.

As tritium is produced in nuclear reactions, whether in a reactor, a nuclear weapon explosion, or by cosmic rays in the atmosphere, it initially exists as single atoms, dispersed in the material in which it was formed. In an oxidizing environment, such as air, it progresses to tritiated water (HTO). In a chemically inert medium, it will remain as a single atom, migrating through voids and interstices, until it can chemically combine with some other trace elements. In materials such as stainless steel, tritium (and similarly, hydrogen) is very mobile and can diffuse through barriers relatively quickly. In more reactive materials, such as zirconium or uranium (metal), or in molten sodium, the tritium atom combines chemically with the host material and diffuses very slowly. In concrete, where many water molecules are bound to the minerals of the cement, tritium can slowly diffuse, from regions of high concentration to regions of lower concentration, by the process of exchange, where a tritium atom changes places with a hydrogen atom in a water molecule.

2.2   TRITIUM PRODUCTION IN NUCLEAR REACTORS

In nuclear reactors, there are several major neutron reactions in which tritium is produced:

1. Ternary (as opposed to binary) fission, in which three nuclear fragments are produced following fission, as opposed to two fission fragments. In some cases, the third fragment is a tritium nucleus, or "triton," shown as t.

2. Thermal (and some fast) neutron absorption by the lithium-6 isotope present in natural lithium in the coolant, moderator, and fuel: $^6Li(n,t)^4He$.

3. Fast neutron absorption by boron-10: $^{10}B(n,t)2^4He$. In this reaction, the additional energy brought in by the fast (rather than thermal) neutron is sufficient to produce a 3-body breakup of the compound nucleus initially formed, $^{11}B^*$. (Here, the * indicates an excited state of the boron-11 nucleus.) At lower neutron energies, the reaction produces ^7Li, which can produce tritium through a subsequent fast-neutron reaction.

4. Thermal neutron absorption by deuterium (^2H or heavy hydrogen) that is naturally present in ordinary water, at very low concentration, or that is artificially present in heavy water ("D_2O") at very high concentrations.

Reactors of various types produce different amounts of tritium by these reactions depending on the material and mode of operation of the reactor. A fast reactor (in which the fission neutrons are not slowed down to thermal energies) produces more tritium by the boron-10 reaction (#3) and by ternary fission (#1) than does a thermal reactor. A thermal reactor moderated by heavy water (D_2O) such as the Savannah River reactors or the Canadian CANDU power reactors, produces more

tritium by the deuterium capture reaction (#4) than does a reactor using ordinary, "light" water (Light Water Reactor [LWR]). A thermal reactor using a nonhydrogeneous moderator, such as graphite, produces essentially no tritium by this process.

Some LWRs use lithium compounds to control water chemistry. This results in tritium production by the lithium–6 absorption reaction (#2).

The LWRs currently in use for electrical power production in the United States consist of two basic types: Boiling Water Reactors (BWRs) and Pressurized Water Reactors (PWRs). Most modern reactors operate at a nominal power of 3,000 MW_t, while some older plants operate at less than 1,000 MW_t. In this report, the unit of power used is the watt (W) and its multiples, the megawatt (a million watts, MW) and the kilowatt (a thousand watts, kW). An electric iron uses about 1 kW of electrical energy during its heating phase, converting most of this energy to heat, or thermal power. Such an iron would be rated at 1 kW_t or 0.001 MW_t. The two different types of reactors produce slightly different amounts of tritium in their operation and release different amounts to the environment. These releases are in full conformance to current NRC regulations and recommendations by all recognized committees on radiation protection. The average production of tritium for 1 year of operation of one reactor of each type, and the annual releases to the environment, in curies, are shown in Table 2–2.

Table 2–2. Tritium Production and Release by Light Water Power Reactors

Reactor Type	Average Production (Ci)	Average Releases	
		Water (Ci)	Air (Ci)
BWR	19,100	10	38
PWR	17,400	302	47

From "Tritium Production, Releases and Population Doses at Nuclear Power Reactors" by H. T. Peterson Jr. and D. A. Baker in *Fusion Technology*, Vol. 8, page 2544–2550   D641-0036

The amount of tritium released from a reactor varies greatly, depending on design and construction details, and on the mode of operation. For example, liquid releases from commercial power reactors reported for 1985 ranged from 1.15 Ci/year to 128 Ci/year for BWRs and from 27 Ci/year to 5,760 Ci/year for PWRs. The greater activity released from PWRs results from the use of boron and lithium compounds to control reactivity (power level) and pH (acid/alkaline balance) in the water coolant/moderator. The tritium produced by neutron reactions with these elements is produced directly in the water and is therefore immediately available for migration into the environs, while the somewhat greater production of tritium in BWRs is due to production in boron carbide control rods, from which only small amounts are released.

Even with these releases, the radiation dose to the population living near these reactors is negligible compared to natural background radiation doses. (The natural production [and release] rate of tritium in the atmosphere is approximately 4 million curies/year. The steady–state natural inventory of roughly 70 million curies from cosmic–ray reactions causes an average individual radiation dose of 1–1.5 microrem/year as estimated by UNSCEAR and the USGS.) For comparison, naturally occurring radon is estimated to cause an average individual radiation dose of 200,000 microrem/year.

The details of construction of a nuclear reactor also affect the release of the tritium that is produced. LWRs use uranium oxide fuel that is a chemically inert ceramic material with voids, interstices, and cracks that enhance the release of tritium. The fuel cladding may be stainless steel, which permits the tritium to pass into the coolant (light water) easily, or it may be zircalloy, a zirconium alloy that forms a relatively effective barrier, trapping much of the tritium produced by ternary fission, as a consequence of the surface oxide coating. In a water–cooled reactor, tritium that reaches the coolant becomes part of the water system. Due to intense radiation and high–temperature chemical reactions with the piping, water molecules are constantly breaking down into hydrogen and oxygen atoms, with subsequent recombination. In the atomic form, the tritium can readily diffuse into and through the stainless steel piping, and be released to the environment, either directly to the air or through the steam system and cooling tower blowdown, to environmental water.

2.2.1 TRITIUM PRODUCTION IN SSFL REACTORS

In contrast to the LWRs, the nuclear reactors at SSFL present a quite different situation. A major difference was in the power level: the highest–power reactor at SSFL, the Sodium Reactor Experiment (SRE), was rated at 20 MW_t (mega–, or million, watts, thermal) and averaged 9.2 MW_t over year–long periods, while current light–water power reactors run at nominal powers ranging from about 1,000 to 3,000 MW_t. Most of the SSFL reactors used a metallic fuel: uranium, thorium, or zirconium–uranium alloy. Tritium is chemically bound in these materials, rather than diffusing to voids and cracks, and so migrates from the fuel relatively slowly. In the SRE, the fuel was surrounded by a sodium–potassium (NaK) "thermal bond" to promote heat transfer from the fuel to the cladding. Tritium migrating from the fuel would become chemically bound to the sodium or potassium, forming the tritium analog of sodium or potassium hydride (NaH, KH), that is, the "tritide" (NaT, KT). While the cladding was stainless steel in most cases, and would, therefore, let tritium pass fairly easily into the coolant, this coolant was molten sodium or sodium–potassium, and any free tritium would immediately react and become relatively immobile, except for the mass flow of the coolant. To maintain the high–purity heat–transfer performance of the coolant, cold–traps were used to remove impurities, primarily oxide, but including hydrides and tritides. Thus, the majority of tritium that did reach the coolant system was sequestered in the cold–traps and would not reach the environment.

While these characteristics of the SSFL reactors prevented significant releases of tritium that was produced in the core, compared to commercial power reactors (this will be quantified later), the shielding arrangement used in these reactors resulted in production of tritium (by neutron absorption in the lithium of the concrete) and presumed release to the adjacent groundwater. In the

SSFL reactors, the biological shield was formed by a concrete structure located directly in the ground.

The element lithium occurs naturally in rocks (the global average for the Earth's crust is 65 ppm). Natural lithium consists of 7.5% lithium-6 and 92.5% lithium-7. Neutrons that escape from the reactor are captured in the shield, predominantly by hydrogen atoms, but also by atoms of lithium-6, thereby producing tritium. This tritium becomes part of the bound water in the concrete, and slowly diffuses from the inner region of the shield, where the concentration is highest, towards the outside of the shield. There, the tritium may become part of the groundwater and move away from the reactor with any bulk flow of the groundwater.

While boron is also naturally present in rocks (the global average is 3 ppm: California sedimentary rocks and soil are likely to contain more), the fast neutron flux reaching the concrete is so low that insignificant quantities of tritium were produced.

The nuclear parameters that were used in estimating the production of tritium by reactors at SSFL are shown in Table 2-3. (A complete listing of engineering data used in the calculations discussed in this report is included as Appendix A.)

2.3 OTHER USES AND OCCURRENCES OF TRITIUM

Tritium has been used at SSFL in very few other situations. For completeness, these are described here:

1. Tritium gas at Rockwell International Hot Laboratory (RIHL)

   Several thousand curies of tritium gas was used over a 3-year period, from 1964 through 1967, at the RIHL to investigate the properties of uranium–zirconium hydride systems. This tritium was used in small increments and would have been released gradually to the atmosphere through the RIHL exhaust system. No

Table 2-3. Nuclear Parameters

Li-6	Capture cross section (thermal neutrons)	940	barns
U-235	Fission cross section (thermal neutrons)	582.6	barns
B-10	Tritium production cross section (fast-flux average)	0.02	barns
H-3	Half life	12.346	years
Li-6	Abundance in natural lithium	7.42	atom %
H-3	Yield in ternary fission of U-235 (thermal neutrons)	1.24×10^{-4}	
Normalized fission rate		3.3×10^{10}	fissions/sec per watt

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scrubbing system or absorbers were used to capture the tritium gas prior to release, and so, no tritiated water resulted from this use. Since tritium is only slowly converted to tritiated water, releases of gas dissipate in the atmosphere. No contamination of groundwater would have occurred due to this use.

2. Tritium neutron generator targets

Titanium and zirconium targets containing 1 to 120 Ci of tritium were used in a variety of neutron generators (using the D, T fusion reaction) to produce high–energy neutrons for research and development projects. Small amounts of tritium outgas during use of these targets are to be trapped by oil in vacuum pumps or discharged to the atmosphere. No transfer to groundwater would have resulted.

3. Tritium in gas–chromatograph detectors

Gas chromatographs are used for extremely sensitive analysis of organic compounds in water and other materials. An important part of the instrument is the detector, which, in some models, uses a titanium–tritium foil of roughly 0.2 Ci as a source of electrons. Essentially no tritium is released from these foils.

4. Tritium in deuterium gas for laser

A high–powered laser using a deuterium–fluorine reaction was operated at SSFL in the Laser Experimental Test Facility (LETF) in Area I. The deuterium gas, furnished by ORNL, contained a small amount of tritium as an impurity, and this was released to the atmosphere during operation of the laser. During an inspection by the Radiation Health Unit of the State Department of Industrial Relations, review of the use of deuterium gas with a trace of tritium as an impurity led to the collection of environmental samples around the LETF. Ten samples were taken in August 1983 and analyzed for tritium. Nine showed "less than detectable," while one, water from the caustic exhaust gas scrubber solution, showed 220,000 pCi/L. While this concentration is well below the allowable limit for that situation (3,000,000 pCi/L) and is even below the concentration requiring monitoring, it was decided to close the sump and replace it with a closed tank. The sump was a concrete–lined surface impoundment that was cleaned and closed in August 1984, under the review and approval of the Toxic Substances Control Branch of the Department of Health Services. The scrubber solution was then contained within a closed tank. Water samples taken from nearby wells have all shown tritium concentrations less than the detection level of the EPA standard method.

3.0 SOURCES OF TRITIUM

Several experimental and test reactors at SSFL had the capability for production of tritium as a consequence of their operation. These reactors are described in Table 3-1, showing the start-up and shutdown dates, the maximum ("rated") power level, and the average power, based on the length of generally steady operation. Since these were experimental reactors, much of the test programs involved operations well below full power.

Table 3-1. Reactors at SSFL

Reactor	Building	Operating Period		Power (MW$_t$)	
		Start	End	Rated	Average
SRE (Core 1)	T143	07/01/58	07/01/59	20	9.2
SRE (Core 2)	T143	02/01/63	02/01/64	20	9.2
SER	T010	09/01/59	12/01/60	0.050	0.028
S8ER	T010	05/01/63	04/01/65	0.600	0.306
S8DR	T059	05/01/68	12/01/69	0.619	0.315
STR	T028	12/01/61	07/01/64	0.050	0.001
STIR	T028	08/01/64	02/01/74	1.0	0.008
S2DR	T024	04/01/61	12/01/62	0.065	0.021
S10FS3	T024	01/01/65	03/01/66	0.037	0.037

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(The "Start" and "End" dates shown above differ slightly from the actual dates of initial criticality and final shutdown. The dates in this table represent the periods of generally steady power operation.) Note that two different fuel loadings ("cores") were operated in the SRE reactor, with essentially identical operating powers. Review of the table shows that three facilities, T143 (SRE), T010 (SER and S8ER), and T059 (S8DR) had the most significant reactor power operation. (The critical facilities and L-85/AE-6 and KEWB reactors have been omitted from consideration because of the much lower power levels.) Estimates of the production of tritium resulting from operation of these reactors have been made, using the best values for each parameter in the calculations. Sensitivity to variations in these parameters was not investigated; however, the goal was to produce estimates that should be consistent with reality, and, except where they are clearly speculative, the results should be accurate within a factor of about 5. The locations of various nuclear facilities at SSFL are shown in Figure 3-1.

For each facility, the production of tritium in various materials (primarily by ternary fission and the ^{6}Li(n,t)^{4}He reaction) was calculated, using measured or estimated values for neutron flux, masses, time, and impurity concentrations.
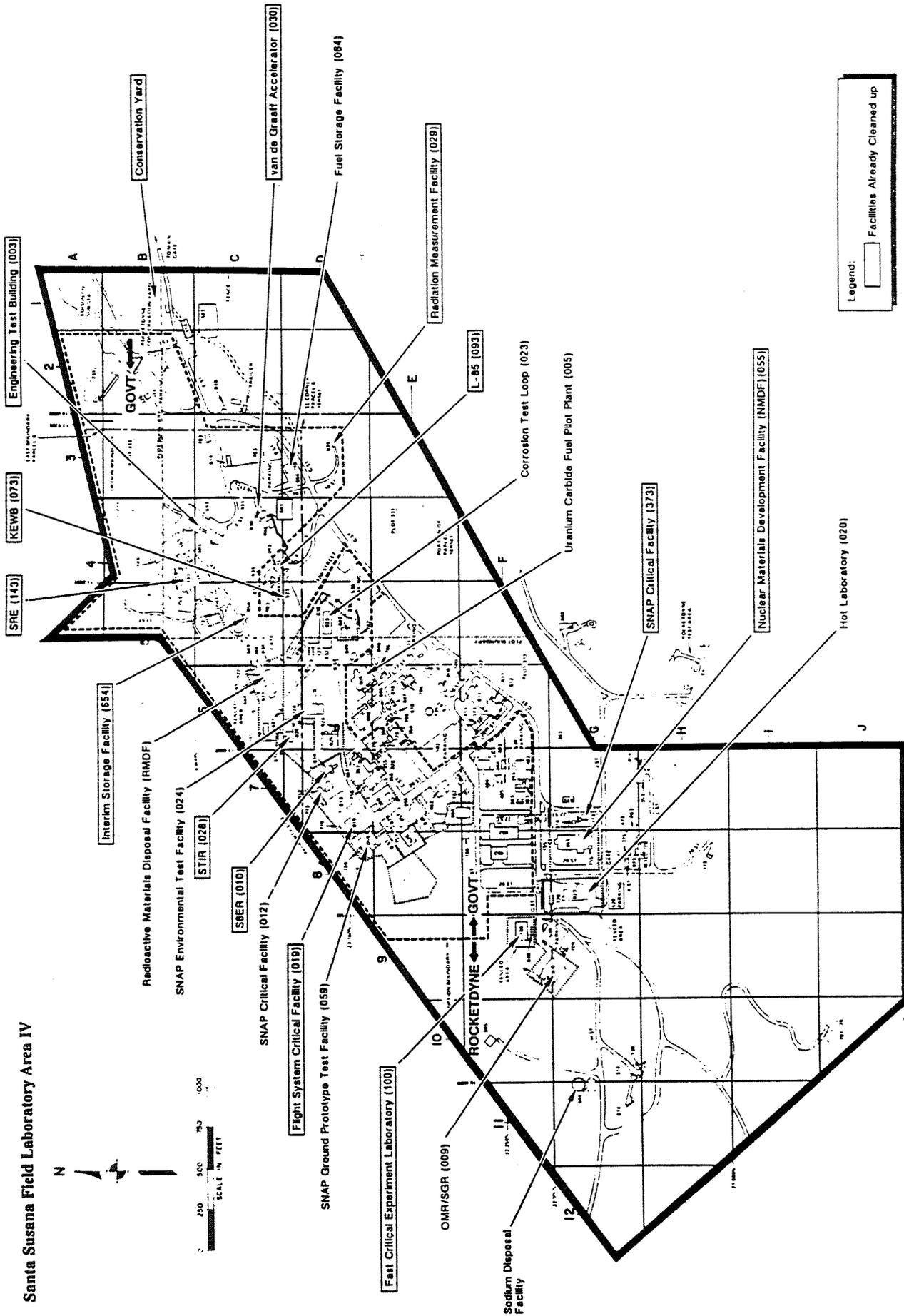
Figure 3-1. Sites Where Nuclear Activities Were Carried Out at the SSFL

3.1 PRODUCTION OF TRITIUM IN THE SRE

As an experimental reactor in the development of nuclear power, the SRE operated at relatively low power (a maximum of 20 MW$_t$) and intermittently, as test conditions were changed and equipment was modified. During two periods of one year each, the SRE averaged 9.2 MW$_t$. Approximately half of this power was sent to the Southern California Edison Company electrical generator plant. Waste heat (condenser cooling) was discharged by use of a recirculating water–cooling tower. When electricity was not being produced, the thermal energy from the SRE was discharged by an air-cooled heat exchanger. An aerial view of the SRE complex is shown in Figure 3–2. A plot plan is shown in Figure 3–3. The SRE building (T143) arrangement is shown in Figure 3–4, and a sketch of the reactor is shown in Figure 3–5. No access was provided around the reactor and the concrete structure adjacent to the reactor vessel was surrounded by back–fill soil. The cover gas, filling the space in the reactor vessel above the sodium coolant, was helium, at a pressure of about 50 psi.

The SRE used graphite to moderate the fission neutrons and this graphite was encased in zirconium metal.

The initial fuel loading at the SRE (Core I) was composed of uranium metal, enriched to 2.8% U–235. In July 1959, clogged coolant channels resulted in the overheating of several fuel elements, melting some of the fuel and the stainless steel cladding. Radioactive material released from the

Figure 3–2. Aerial View of SRE with Southern California Edison Steam–Electrical Generating Plant and Cooling Towers
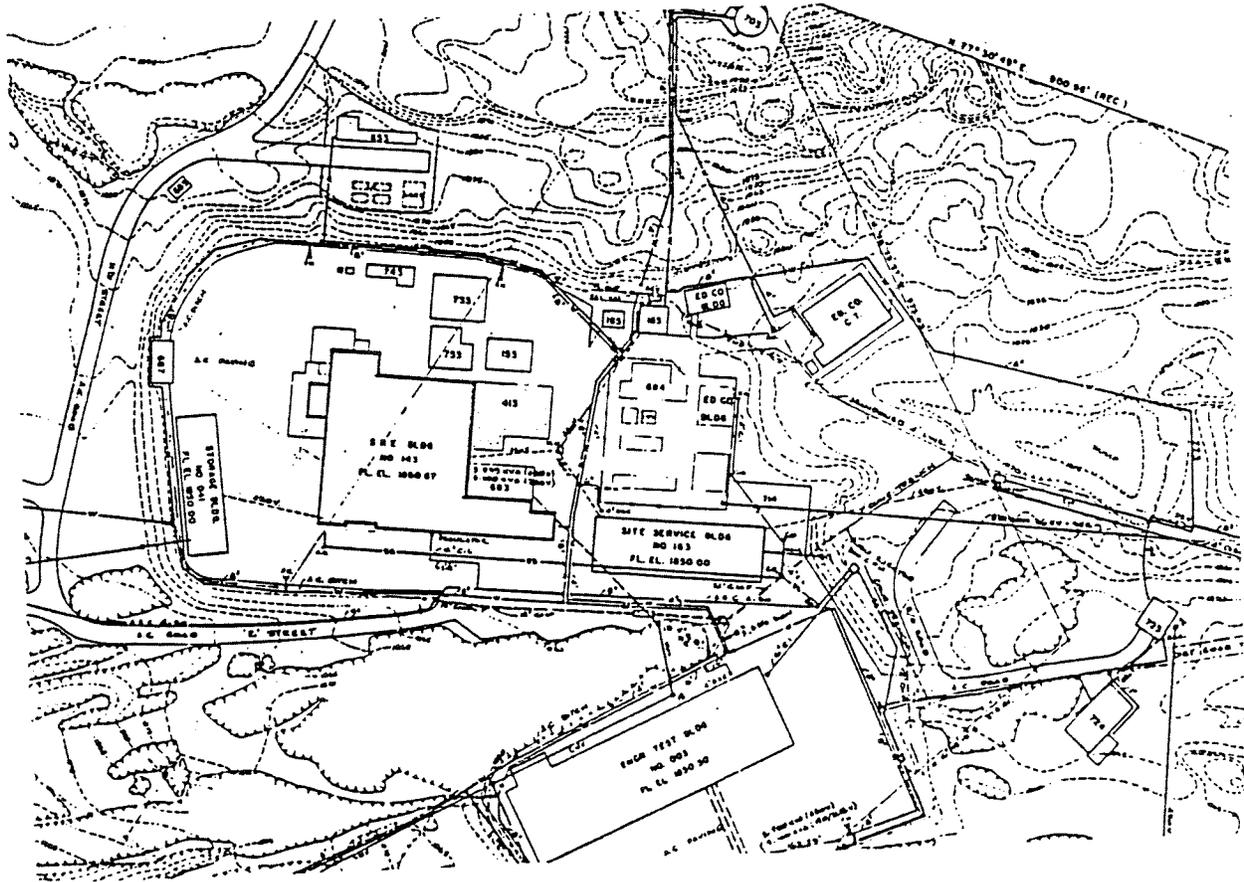
Figure 3-3. Plot Plan of SRE Facility
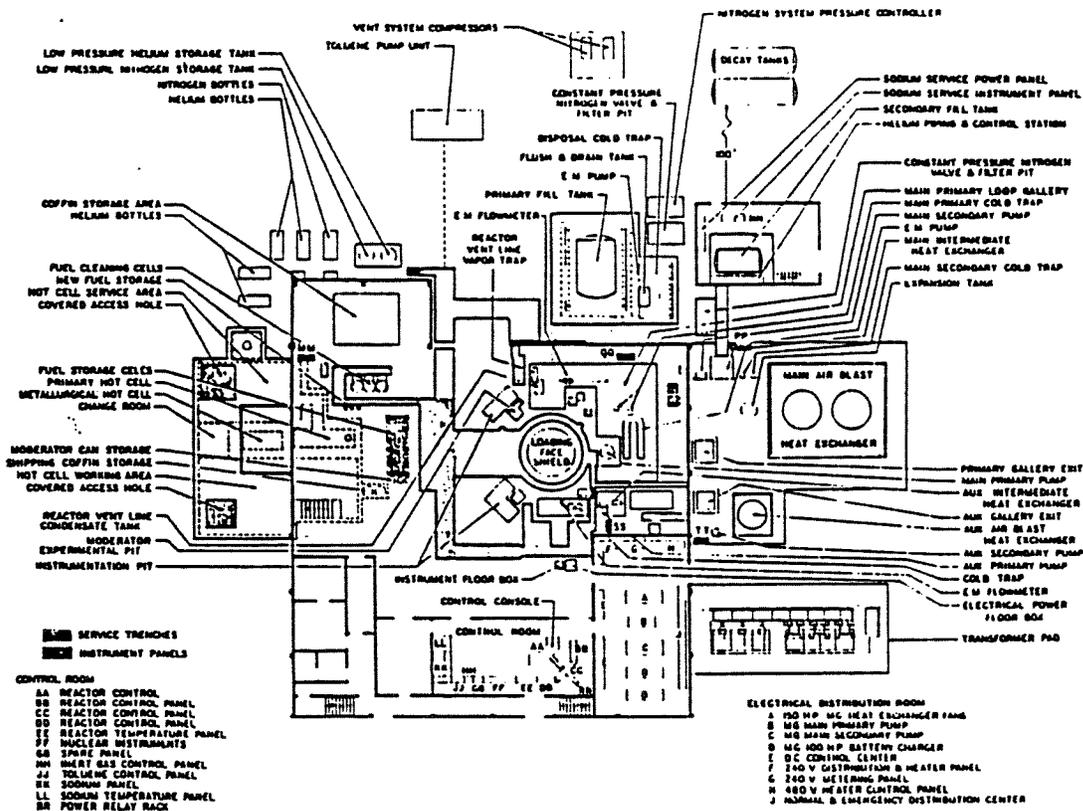
Figure 3-4. SRE Building Arrangement
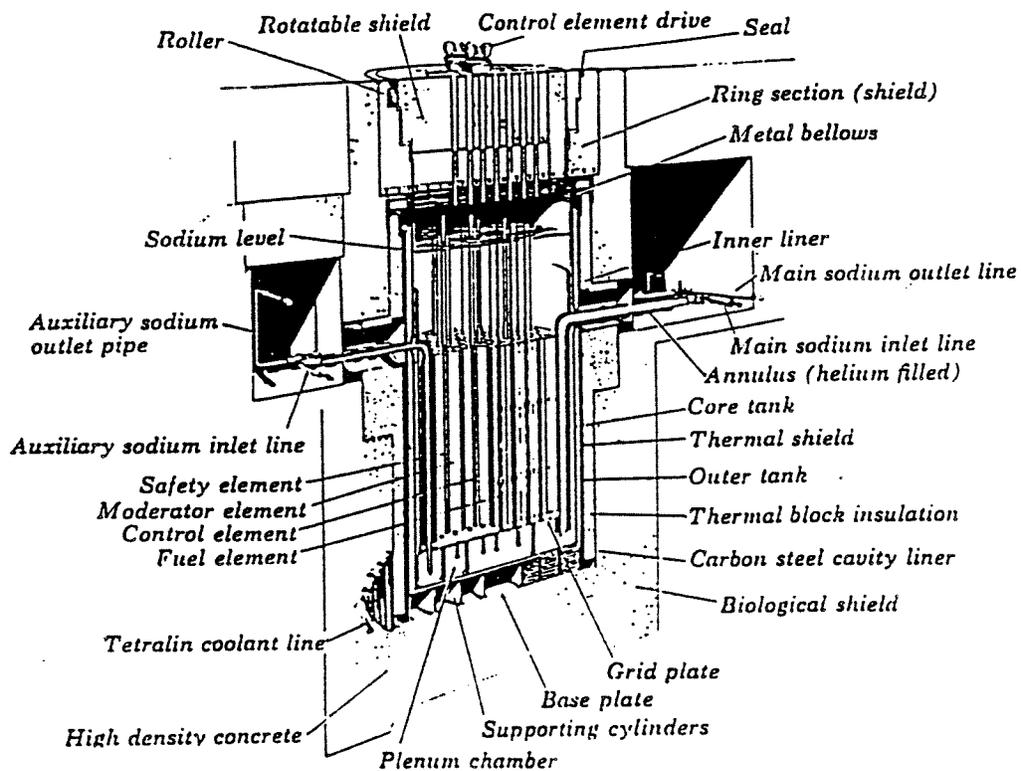
Figure 3-5.   SRE Reactor Cross Section

damaged fuel elements remained in the reactor except for a small amount of the radioactive noble gases krypton and xenon which collected in the cover gas. This cover gas was pumped to gas storage tanks to allow the short-lived xenon isotopes (Xe-133, $T_{1/2}$ = 5.2 days; Xe-135, $T_{1/2}$ = 9.1 hours) to decay before release to the atmosphere.

After removal of the fuel elements used in Core I and cleanup of the primary sodium (radioactivity that was dissolved in the sodium, or had chemically reacted, was largely removed from circulation by the cold traps), a second set of fuel elements (Core II) was installed, and testing of the reactor and power producing systems was resumed. The fuel in Core II used thorium alloyed with about 7.5% uranium that was highly enriched (93% U-235). No significant incidents occurred during operation of Core II, and operations were terminated in anticipation of further development of this reactor type at a higher power level. This project was cancelled and operation of the reactor was not resumed. Decontamination of the reactor facility started in 1975. Decommissioning was completed in 1982.

During decommissioning, the primary sodium was removed and shipped to Hanford (Washington) for possible reuse, residual sodium was passivated with alcohol, which was solidified and disposed of as radioactive waste, all radioactively contaminated components (Intermediate Heat Exchanger [IHX], valves, pumps, cold/hot traps, flowmeters, piping, etc.) were disposed of as radioactive waste, and contaminated/activated concrete and soil was removed, packaged, and

disposed of as radioactive waste. Fuel elements from both cores were eventually declad and sent to the Savannah River Plant for reprocessing.

Production of tritium in the SRE operations occurred predominantly in the following ways:

1. Neutron absorption by lithium–6 present in the concrete shield

   This was originally calculated in the planning phase of the SRE Decontamination and Decommissioning (D&D) project by use of a detailed reactor flux calculation and showed that 160 Ci of tritium would have still existed in the concrete shield at April 1975, assuming no removal. These calculations used a lithium–6 cross section of 950 barns, an abundance of 7.4%, and a tritium half–life of 12 years. The concentration of lithium in the concrete was estimated from mineralogical data to be 41 ppm. This value was in good agreement with the lithium concentration (50 ppm) measured in the shielding concrete of the power reactor* built by North American Aviation for Piqua, Ohio, as part of the civilian power reactor demonstration program. However, recent analysis of structural (ordinary) concrete taken from the SRE building showed only 5.9 ppm, as opposed to the assumed 41 ppm. (The SRE biological shield was removed in 1979 for disposal as radioactive waste and is not now accessible for direct analysis. Samples were not analyzed for tritium during demolition of the shield because tritium was a minor component of the induced radioactivity and was not significant with respect to removal or disposal.)

   Adjusting this calculation for the current values of the parameters (half–life, cross section, isotopic composition), and correcting for an overestimate of the effective irradiated volume, leads to an estimate of the maximum tritium produced in the SRE biological shield of 4.99 Ci at February 1964.

2. Neutron absorption by lithium–6 present in soil surrounding the biological shield

   While the thick concrete shield absorbed nearly all the neutrons that escaped from the reactor, it is calculated that approximately 1.5 neutrons per million absorbed in the shield reached the adjacent soil. The lithium content in local soil was measured for three samples in 1991 and found to be 27.13 ppm (standard deviation equals 4.07 ppm). It is estimated that only about 0.0001 Ci was produced in each operating period. Essentially all the soil surrounding the biological shield was removed in 1978–79 and disposed of as radioactive waste at a licensed disposal site.

3. Ternary fission

   This was calculated to be 55 Ci at the end of a 1–year operation for Core I, and the same amount is estimated for Core II. (The fuel elements from Core I were

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*This reactor, designated the Piqua Nuclear Power Facility (PNPF), was operated as part of the AEC civilian nuclear power demonstration program from July 1963 to January 1966. Its maximum power was 46 MW_t.

removed from the reactor in 1959–60. The Core II fuel elements were removed in 1964.)

4. Neutron absorption by lithium–6 present as an impurity in graphite moderator

While no data were found on the presence of lithium in the SRE graphite, the production of tritium due to this impurity was considered. Review of the literature on reactor–grade graphite indicates that lithium was not considered to be a significant impurity, perhaps because of its absence from the starting material, petroleum. However, in a sample of reactor–grade graphite contemporaneous to the SRE, an impurity concentration of 1.2 ppm was measured. (This would result in a loss in reactivity of about 0.6% Δk, which seems about as much as would be acceptable.) The production of tritium in this manner would have resulted in a maximum of 4,600 Ci at the end of Core I and an additional 4,600 Ci from Core II, resulting in a total of less than 8,200 Ci, at February 1964, allowing for radioactive decay of the Core I tritium, and accounting for the removal and disposal of some of the Core I moderator units.

5. Neutron absorption by lithium–6 present as an impurity in sodium coolant

Analyses of samples of the SRE primary sodium, performed at the Hanford Engineering Development Laboratory (HEDL) in 1975, showed a lithium content of 0.002 ppm. This concentration was used to estimate tritium production due to neutron absorption by Li–6 in the coolant. This effect was estimated to produce about 0.02 Ci during each operating period, for a total production of 0.04 Ci. (Radioactive decay during the elapsed time reduced the maximum amount present at shutdown of Core II to 0.035 Ci.) The sodium coolant was shipped to Hanford (Washington) in 1975.

6. Neutron absorption by boron–10 control rod absorber producing tritium

The control rod absorber material in the SRE was a boron dispersion in nickel. Assuming that the control rods absorb neutrons equivalent to the long–term excess reactivity, this process would have produced 35 Ci in Core I and an additional 35 Ci in Core II for a maximum of 62 Ci at February 1964, accounting for decay of the Core I tritium.

Summarizing the revised estimates of tritium production in the SRE:

1.	Concrete shield: maximum	5.0 Ci, February 1964
2.	Soil: maximum	0.0002 Ci, February 1964
3a.	Ternary fission, Core I at shutdown	55 Ci, July 1959
3b.	Ternary fission, Core II at shutdown	55 Ci, February 1964

4.   Graphite moderator: maximum   8,200 Ci, February 1964

5.   Sodium coolant: maximum   0.035 Ci, February 1964

6.   Boron-10 reactions: maximum   61 Ci, February 1964

3.2 PRODUCTION OF TRITIUM AT T010

Two reactors were tested in Building T010: the SER, rated at 0.050 MW_t, and the S8ER, rated at 0.600 MW_t. The facility arrangement and floor plan are shown in Figure 3–6, with more descriptive views in Figure 3–7. A diagram of the S8ER is shown in Figure 3–8. The full diameter of the reactor was about 2 feet. It was installed in a steel containment vessel (Figure 3–9) that was 38 inches in diameter at the reactor level, which was embedded in a concrete shield approximately 2 feet thick, that was in contact with the soil. The concrete shield and containment vessel were removed intact from the facility in February 1978 (Figure 3–10), and this unit was stored for six months in the RMDF yard (Figure 3–11) until shipment to the licensed disposal site at Beatty, Nevada (Figures 3–12 and 3–13). In July, after transfer to the RMDF, the shield was initially exposed to the environment, but it was completely wrapped in plastic sheeting sometime before May 22, precluding any further contact with rain water.

The fuel in both reactors was hydrided zirconium–uranium alloy clad in Hastelloy–N. No thermal bond was used, but the inside of the tubing was coated with a ceramic to reduce the loss of hydrogen from the fuel. This coating would similarly reduce the release of tritium from the fuel.

Heat generated by operation of both the SER and the S8ER was carried by a secondary (non-radioactive) NaK system to the air–blast heat exchanger. Cold traps were provided for the fill and drain tanks but do not seem to have been part of a flow–through coolant–purification system.

Production of tritium in the SER and S8ER operations of T010 occurred predominantly in the following ways:

1. Neutron absorption by lithium–6 present in the concrete biological shield

    This was calculated in the planning for the D&D work, using the maximum neutron flux during the S8ER operation, at the surface of the concrete, averaged over a relaxation length of 11 cm. Reevaluating this for a flux of 3.2×10^{10} $n/cm^2/sec$ (estimated from the point source model used for the similar S8DR reactor in Building T059), an exposed area of 1.1×10^5 cm^2, a relaxation length of 11 cm, and a lithium content of 6.37 ppm (the average for concrete from T143 [SRE] and T059), gives a maximum tritium content of 7.6 Ci at shutdown of S8ER on 15 April 1965. This is higher than in the other reactor facilities because of the absence of any intervening material, such as coolant, thermal shields, or high–density concrete, between the reactor and the concrete, and the lack of added boron in the concrete as was used in T024 and T059. The concrete shield was removed, intact, in February 1978 and
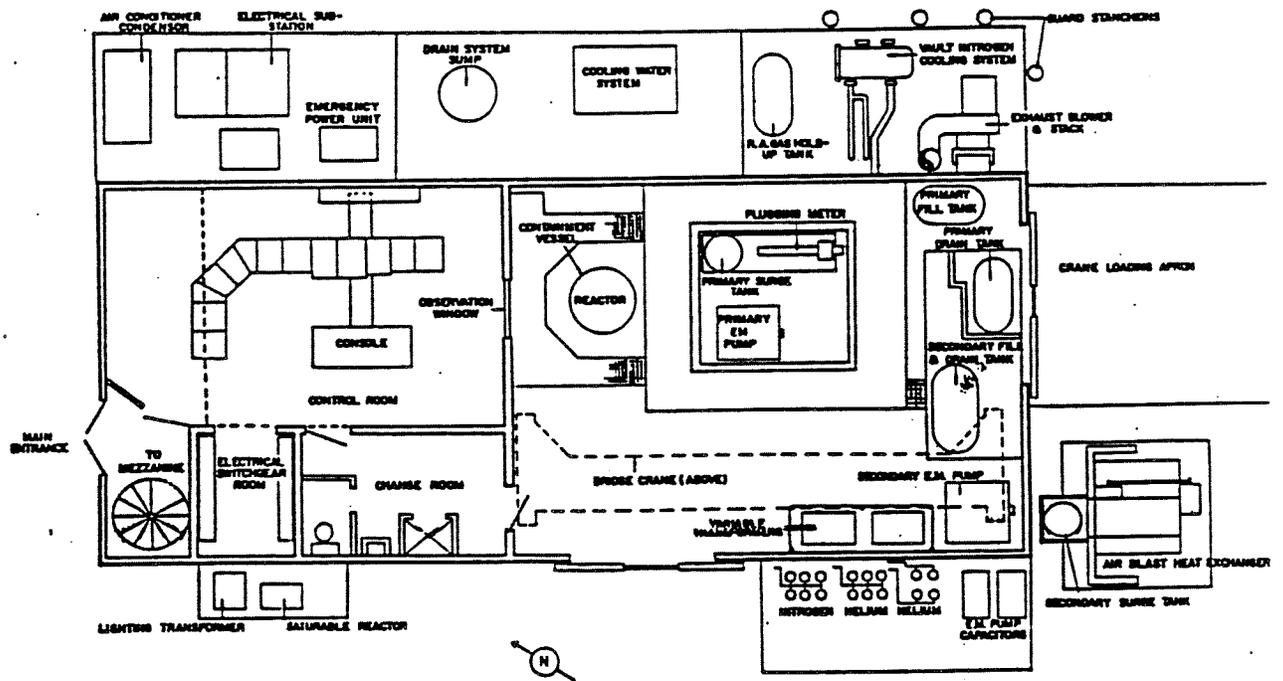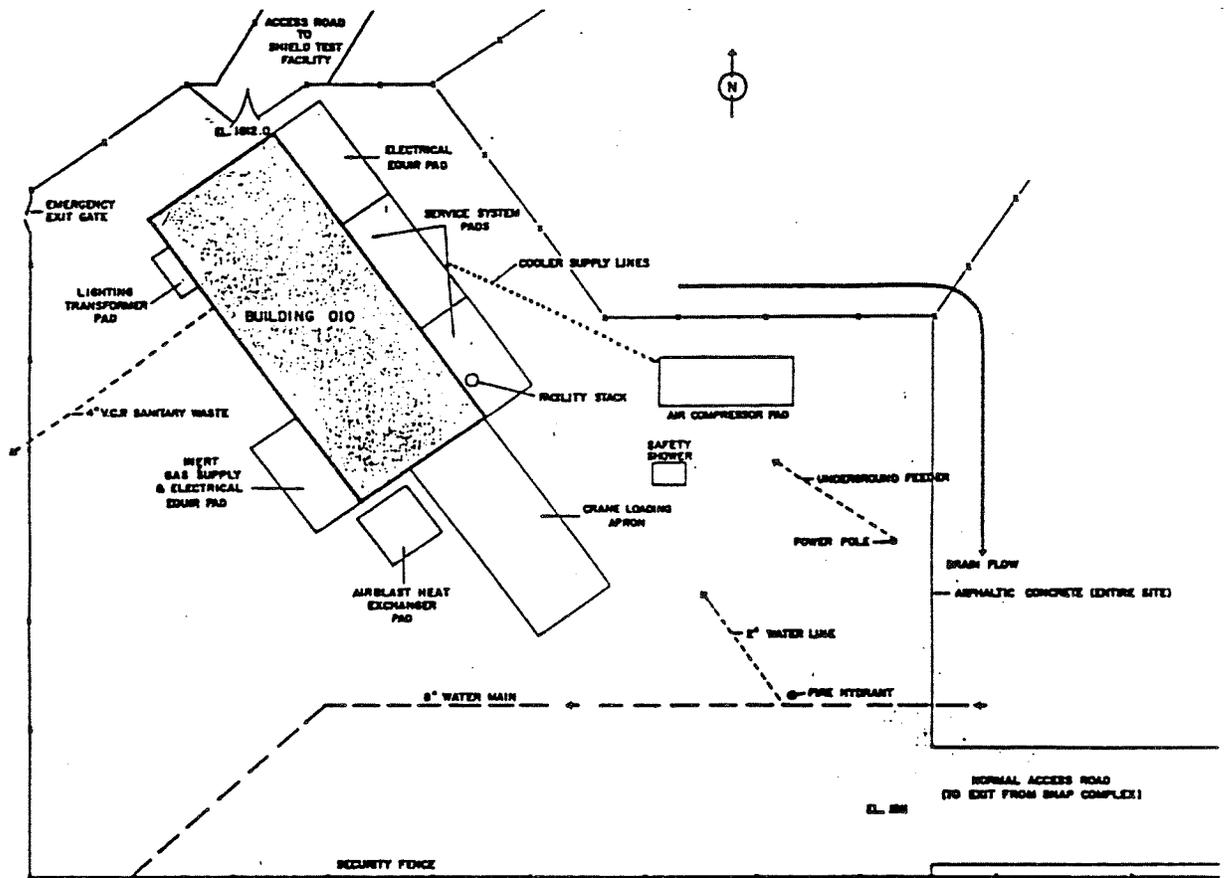
Figure 3–6. Building T010, Facility Arrangement and Floor Plan
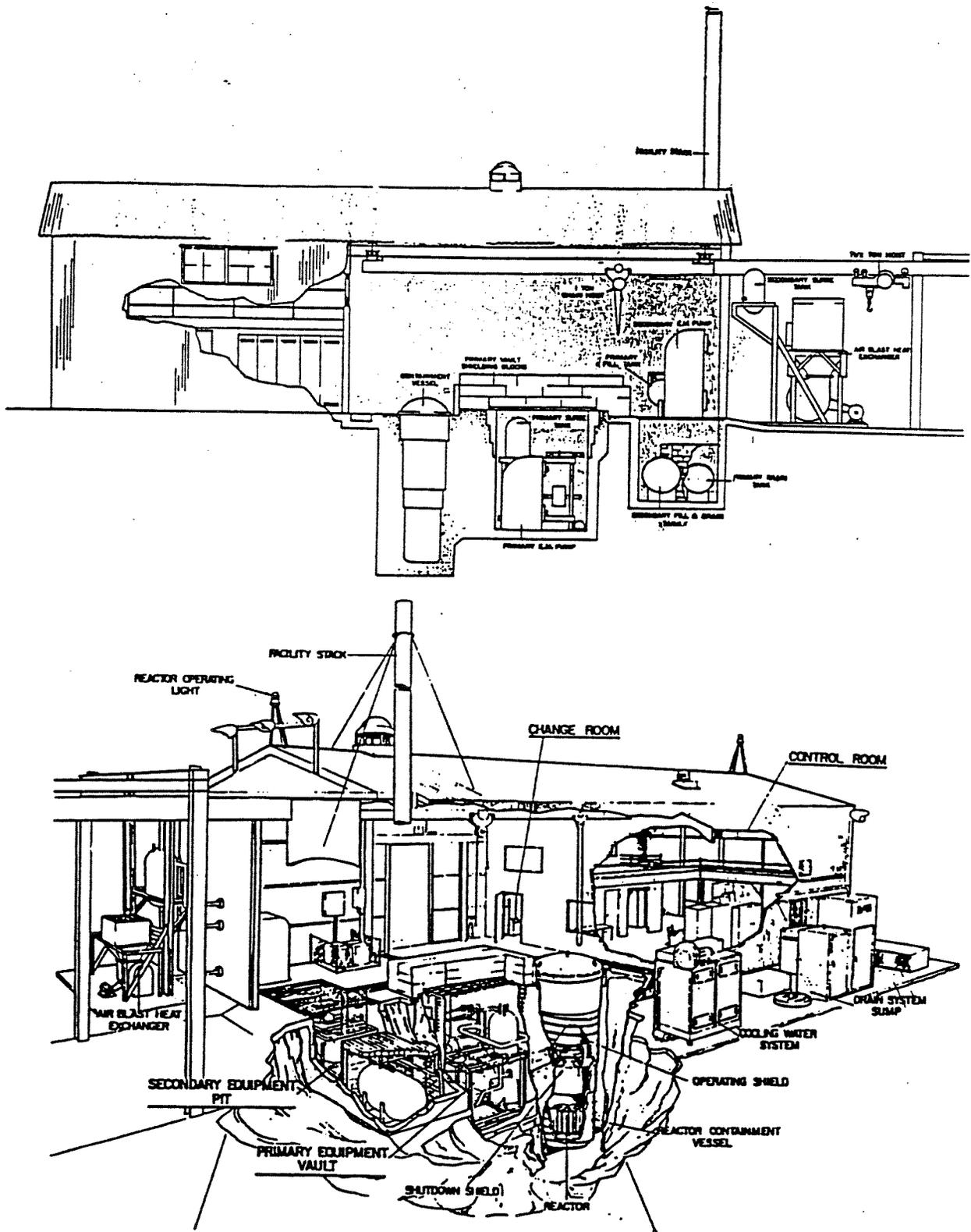
Figure 3-7. Building T010, Elevation and Cutaway Views
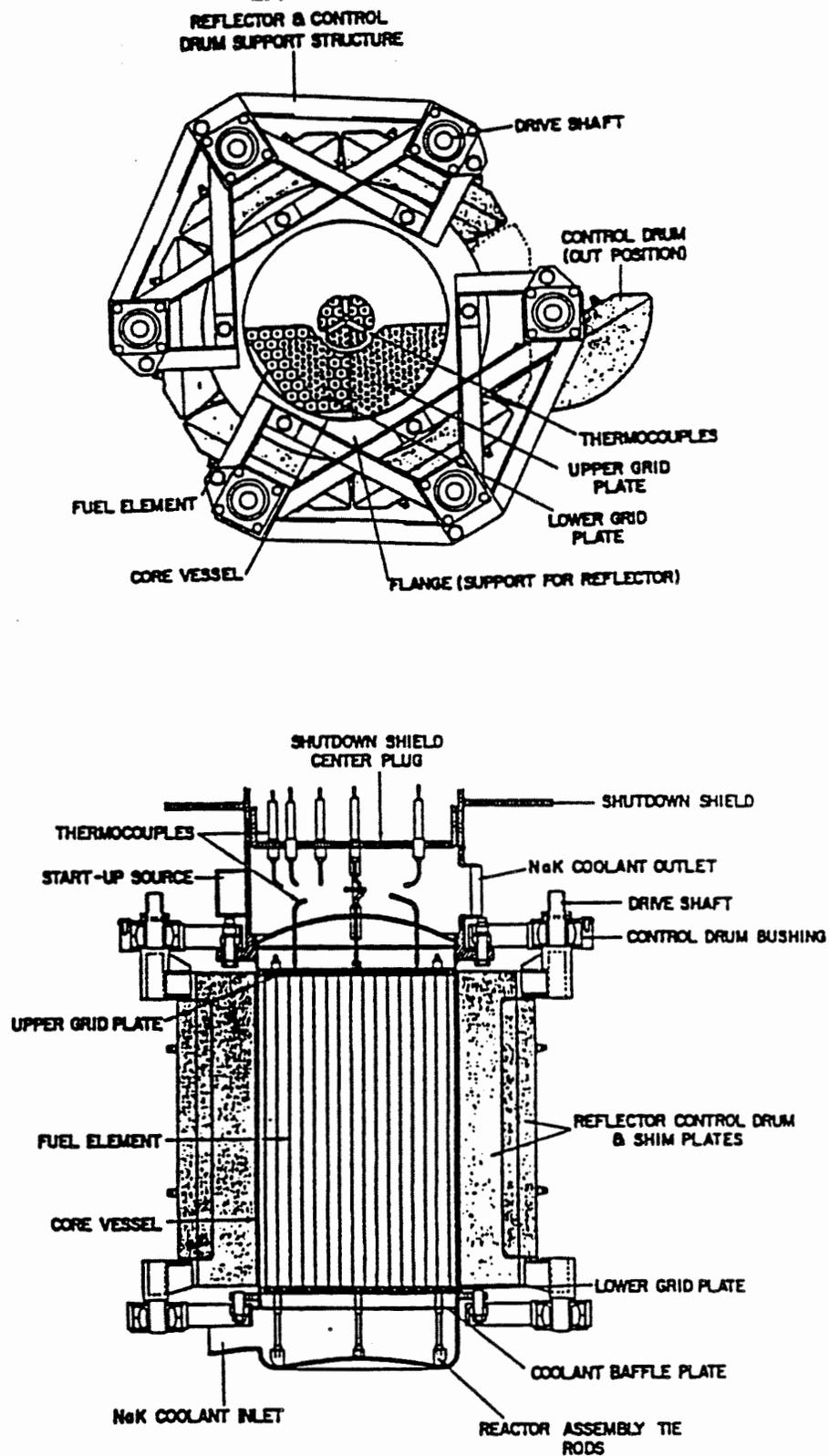
Figure 3–8.   S8ER Reactor Plan View and Section
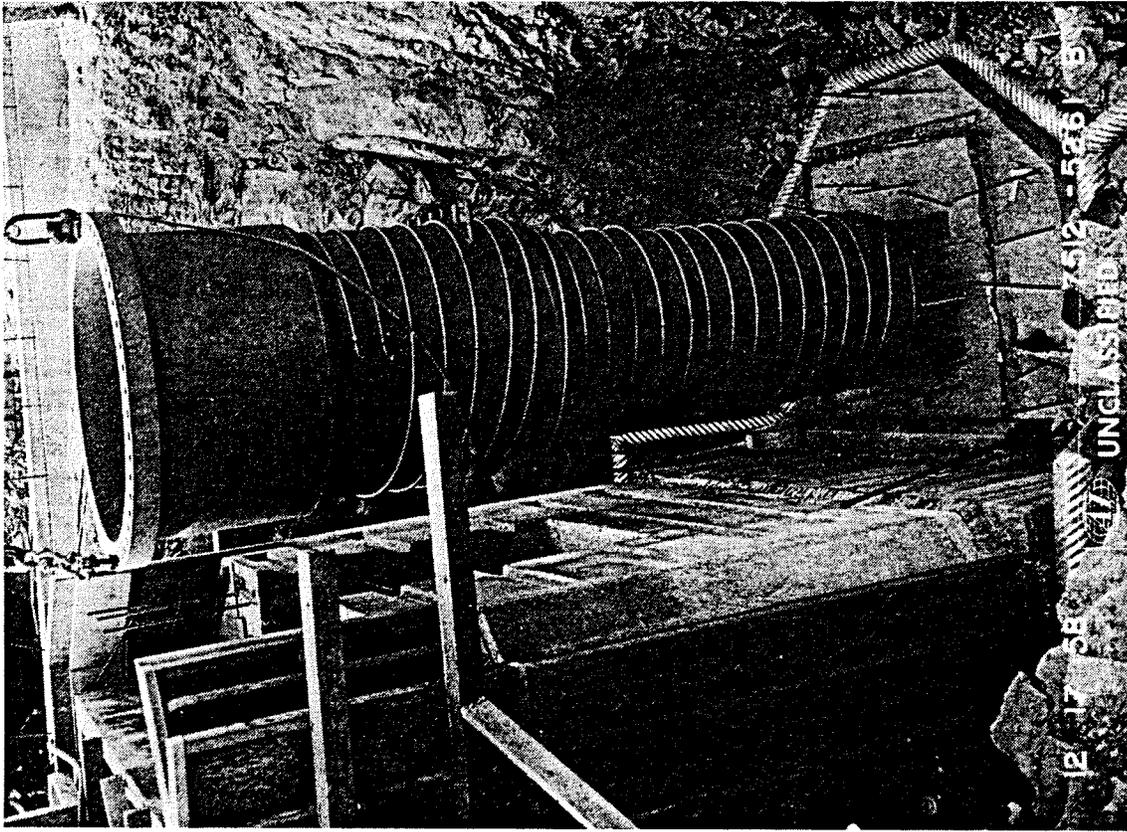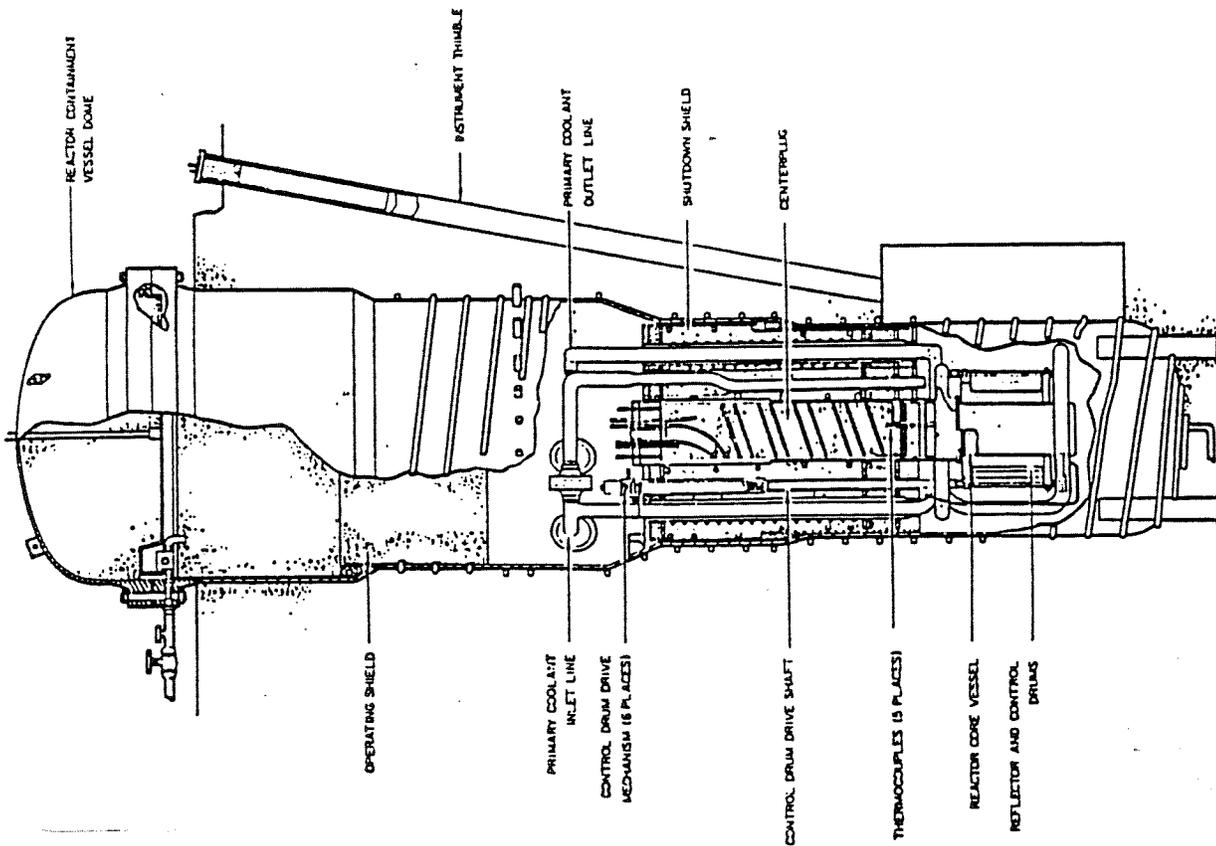
Figure 3-9. Reactor Containment Vessel at T010, Cutaway View and During Installation

REACTOR CONTAINMENT VESSEL DOME

INSTRUMENT THIMBLE

PRIMARY COOLANT OUTLET LINE

SHUTDOWN SHIELD

CENTERPLUG

OPERATING SHIELD

PRIMARY COOLANT INLET LINE

CONTROL DRUM DRIVE MECHANISM (6 PLACES)

CONTROL DRUM DRIVE SHAFT

THERMOCOUPLES (5 PLACES)

REACTOR CORE VESSEL

REFLECTOR AND CONTROL DRUMS
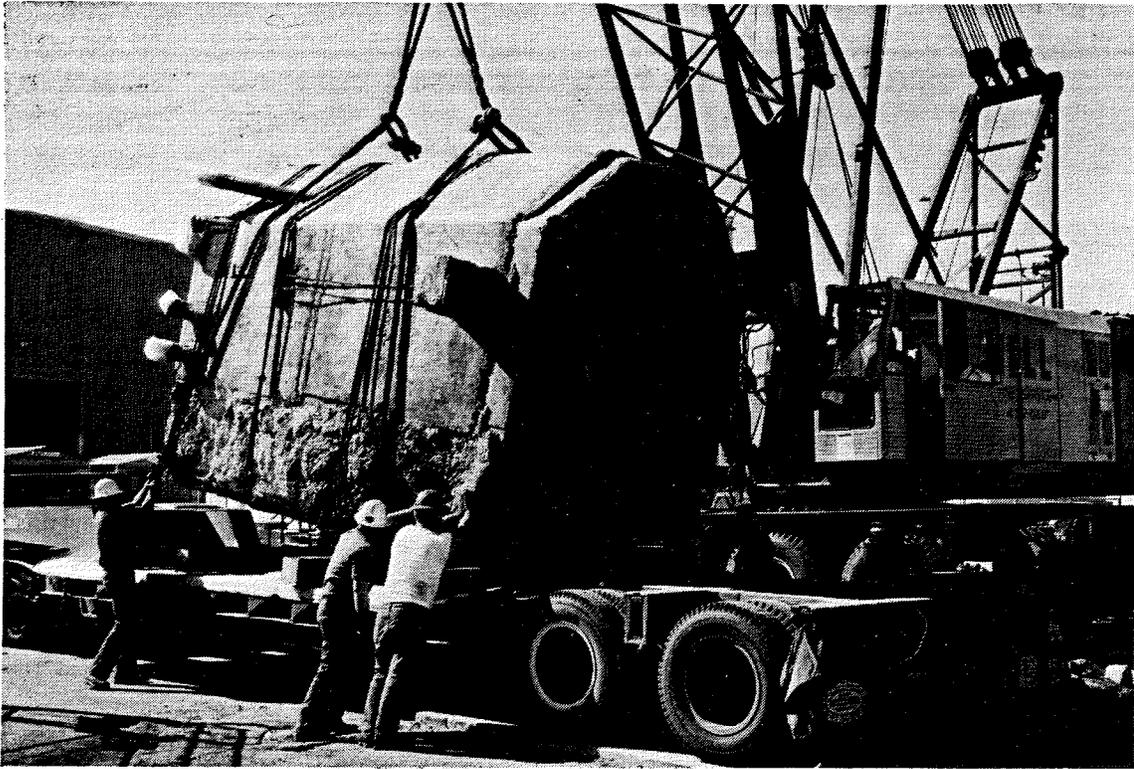
Figure 3-10. Removal of Reactor Containment Vessel and Concrete Shield from T010
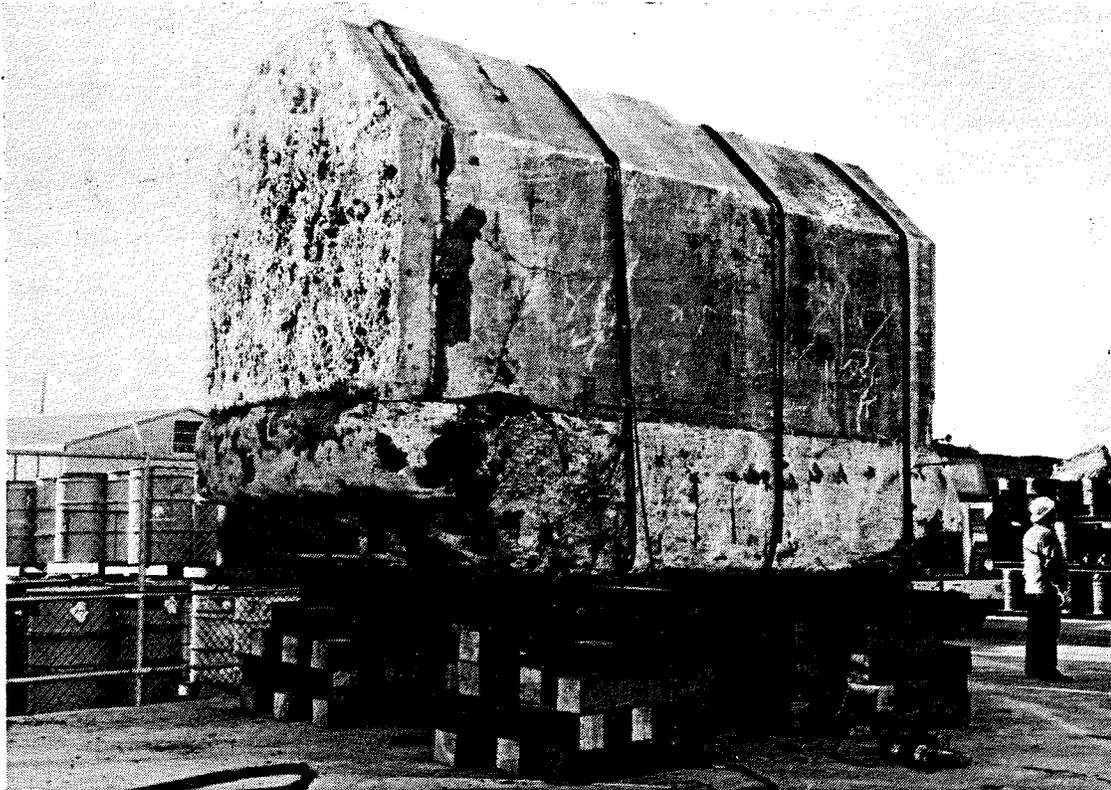
Figure 3-11. Reactor Containment Vessel and Concrete Shield in Storage at RMDF
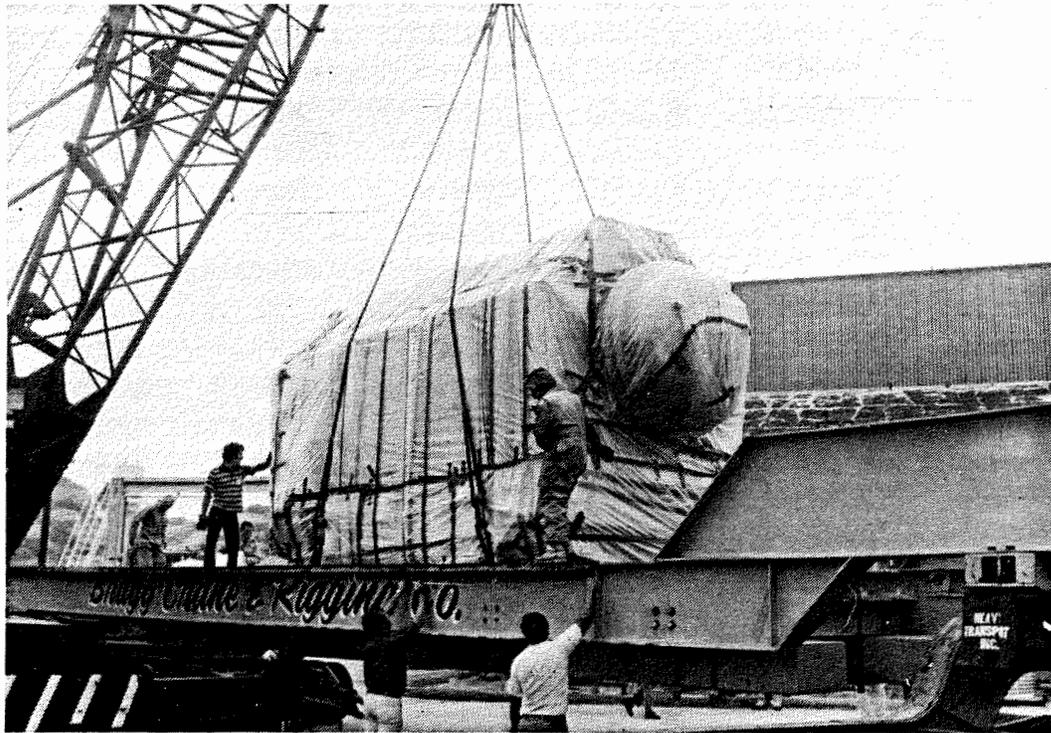
Figure 3-12.   Shield and Vessel Being Loaded on Trailer for Shipment to Disposal Site
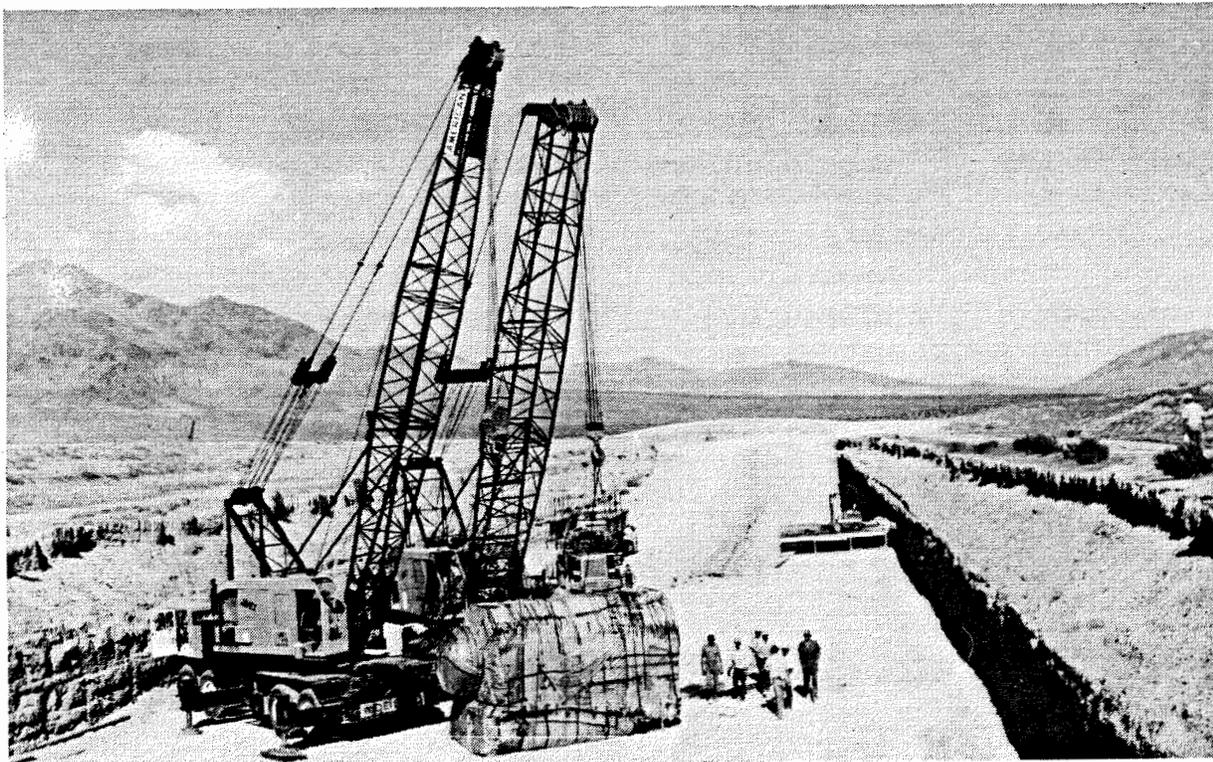
Figure 3-13.   Shield and Vessel at Disposal Site

temporarily stored at Radioactive Materials Disposal Facility (RMDF). It was shipped to the licensed disposal site at Beatty, Nevada for disposal in July 1978.

2. Neutron absorption by lithium–6 present in the soil surrounding the biological shield

Since the concrete biological shield was relatively thin, a significant fraction of the neutrons (0.0003) that were produced in the reactor reached the adjacent soil. Assuming the lithium content of the soil to be 27.13 ppm, as measured, it is estimated that about 0.003 Ci of tritium was produced during the SER operating period, and about 0.036 Ci was produced by S8ER operation. Most of this soil remained at the facility following removal of the biological shield.

3. Ternary fission

This was calculated to be 0.2 Ci at the end of SER operation and 2.6 Ci at the end of S8ER operation.

4. Neutron absorption by lithium–6 present as an impurity in NaK coolant

Because of the much smaller volume of NaK in the cores of the SER and S8ER, compared to sodium in the SRE, production of tritium in this manner would have been much less than the proportionate reduction from the SRE production based on energy released. Assuming a lithium content of 0.002 ppm in the coolant, this has been estimated to be negligible for SER and 0.0001 Ci for S8ER.

No other potential sources of tritium production at T010 were identified.

Summarizing the estimated tritium production at T010:

1.	Concrete shield: maximum	7.6 Ci, April 1965
2.	Soil: maximum	0.038 Ci, April 1965
3a.	Ternary fission, SER at shutdown	0.2 Ci, December 1960
3b.	Ternary fission, S8ER at shutdown	2.6 Ci, April 1965
4.	NaK Coolant:	0.0001 Ci, April 1965

3.3 PRODUCTION OF TRITIUM AT T059

Building T059 was initially constructed in 1962/63 and then modified in 1964 to provide a simulated space environment for testing SNAP-8 reactors. Significant power operations began with S8DR in May 1968. A cutaway sketch and the facility layout is shown in Figure 3–14. An elevation view (N–S) through the reactor pit is shown in Figure 3–15. A floor plan of the reactor test vault area is shown in Figure 3–16. The floor of the reactor pit is 51 ft below grade level, at an elevation of 1,763.7 feet above mean sea level.
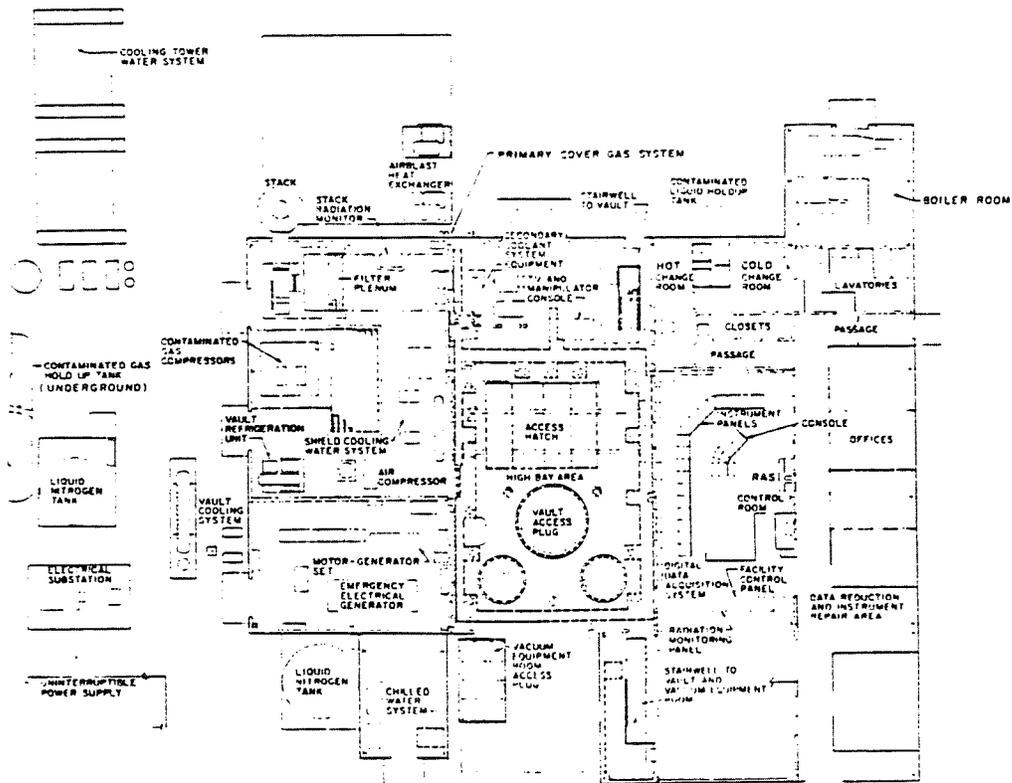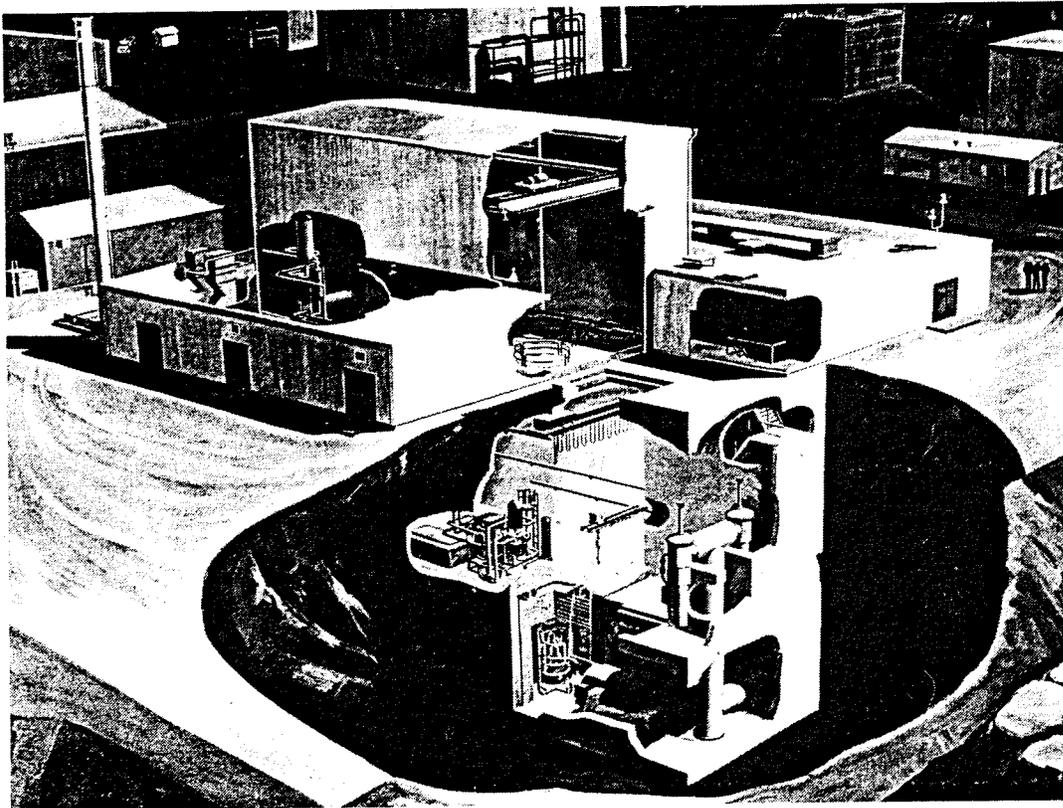
Figure 3–14. Cutaway View and Facility Arrangement, T059
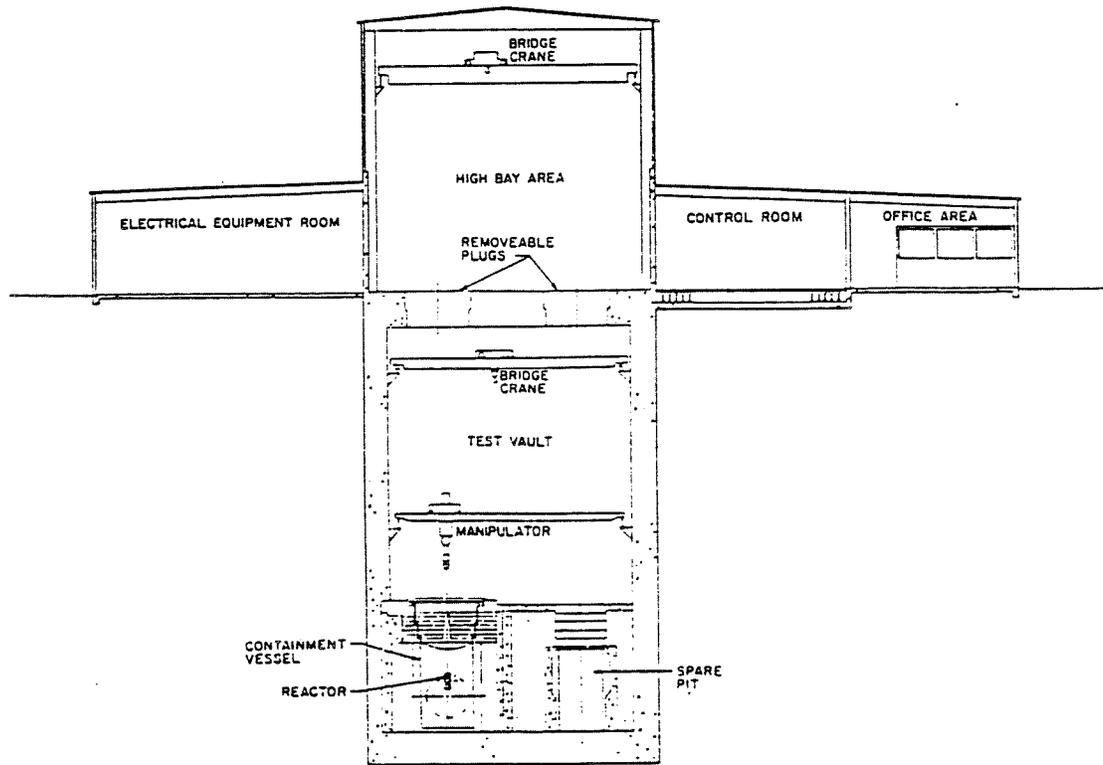
Figure 3–15.   Elevation of T059, Looking East
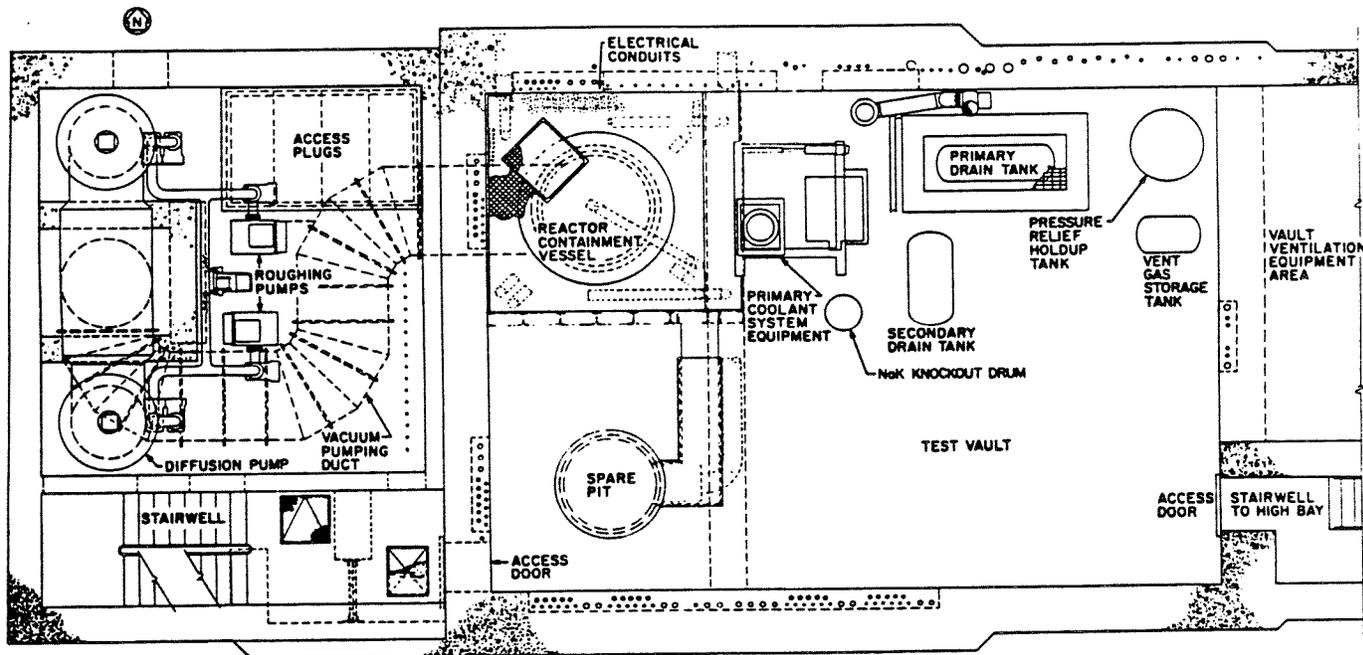
Figure 3–16.   Floor Plan of T059 Basement

The reactor was operated in a vacuum vessel to simulate the space environment, and this is shown in Figure 3-17. Special shielding for this reactor is shown in Figures 3-18 and 3-19.

The S8DR reactor is shown in Figure 3-20 and was similar to the S8ER that was operated in Building T010. This figure also shows schematically the arrangement for testing a lithium hydride shadow-shield for satellite applications. This shield was contained in a separate vacuum enclosure within the overall vacuum vessel.

The reactor contained 211 fuel/moderator elements of $(Zr-U) H_x$ with beryllium reflector segments that fully filled the core vessel. The fuel/moderator elements were clad in Type 316 stainless steel.

Heat from operation of S8DR was transferred to a secondary (nonradioactive) NaK coolant system through an IHX. Heat was discharged to the atmosphere from the secondary system by means of two Air-Blast Heat Exchangers. Each was adequate to discharge 600 kW_t by itself.

Both the primary and secondary coolant systems used cold traps to maintain coolant purity. The minimum residence time was 30 minutes, at 0.5 gpm to assure precipitation of impurities, including oxides and hydrides. It could be operated at 1.5 gpm, with a minimum residence time of 10 minutes to provide rapid coolant cleanup. The coolant system contained void volumes in tanks but not in the reactor itself. Helium was used as a cover gas in the coolant systems.
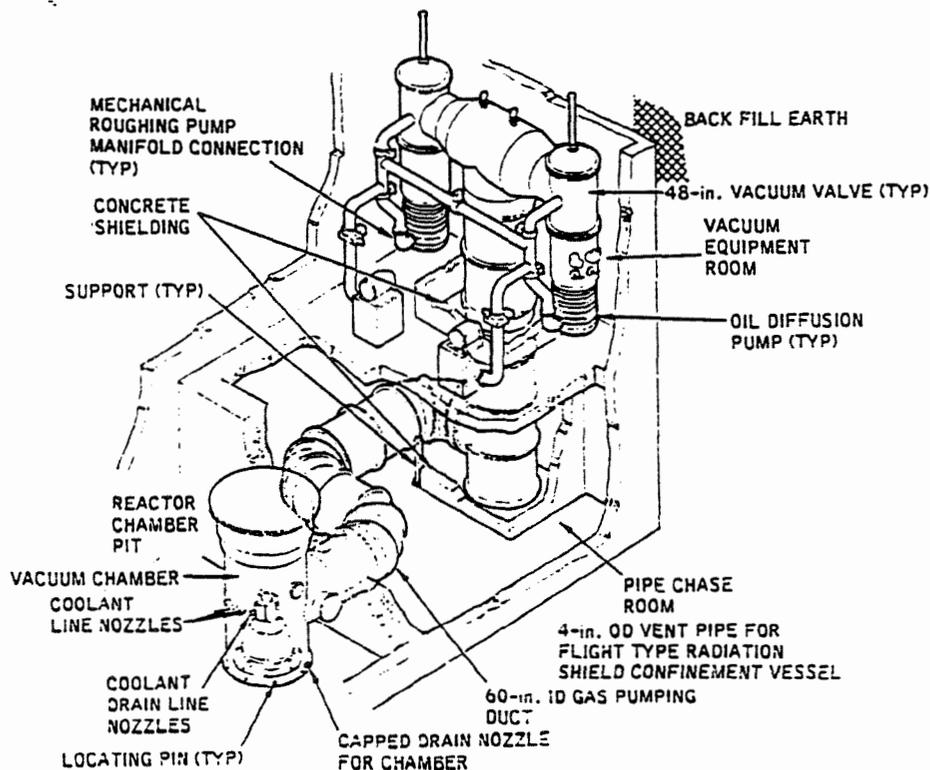
Figure 3-17. Test Arrangement for S8DR
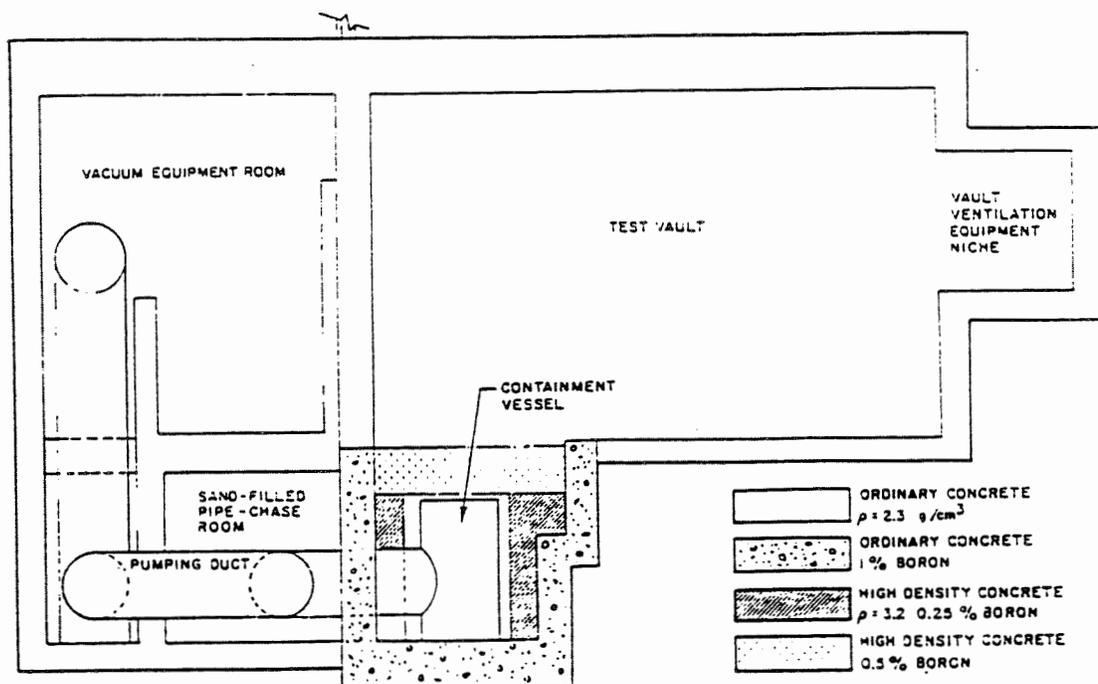
Figure 3-18. Shielding Arrangement for S8DR Test

The S8DR was mounted on a prototype SNAP-8 shield consisting of lithium hydride, and was contained in a vacuum chamber that was continuously evacuated to simulate conditions in space. The shield was contained in its own vacuum confinement vessel to assure isolation from the reactor test. A gamma shield, 0.875 in. of tungsten alloy, was attached to the reactor, above the lithium hydride shield. To minimize nuclear heating in the shield material due to neutron absorption by lithium-6, the shield was composed of lithium-7 hydride, with the lithium-6 content reduced from the natural value of 7.4% to possibly 0.1%, based on current specifications from ORNL. (Material with a lithium-6 content as low as 0.06% might have been available at the time.) This also significantly reduced the tritium production due to neutron absorption in lithium-6.

The shield was clad in stainless steel and contained within a stainless steel confinement vessel, as shown in Figure 3-20. The outside of the shield cladding and both inside and outside of the confinement vessel were coated with chromic oxide to achieve a high thermal emissivity. (This coating would also significantly reduce hydrogen and tritium permeation). Thus, tritium produced in the lithium hydride shield material was isolated from the environment by double containment in oxide-coated stainless steel. For test purposes, the shield and the confinement vessel could be pressurized with helium or evacuated. Gas from these regions was pumped to the gas holdup tanks and then, after delay for any needed decay of radioactive argon and xenon, discharged to the atmosphere through the facility stack.
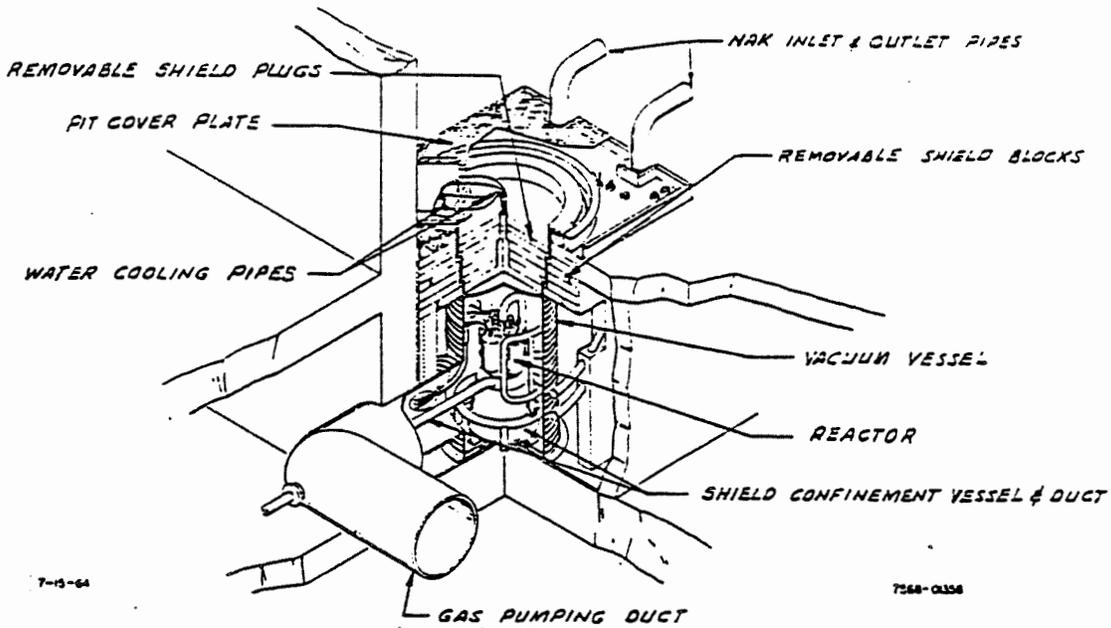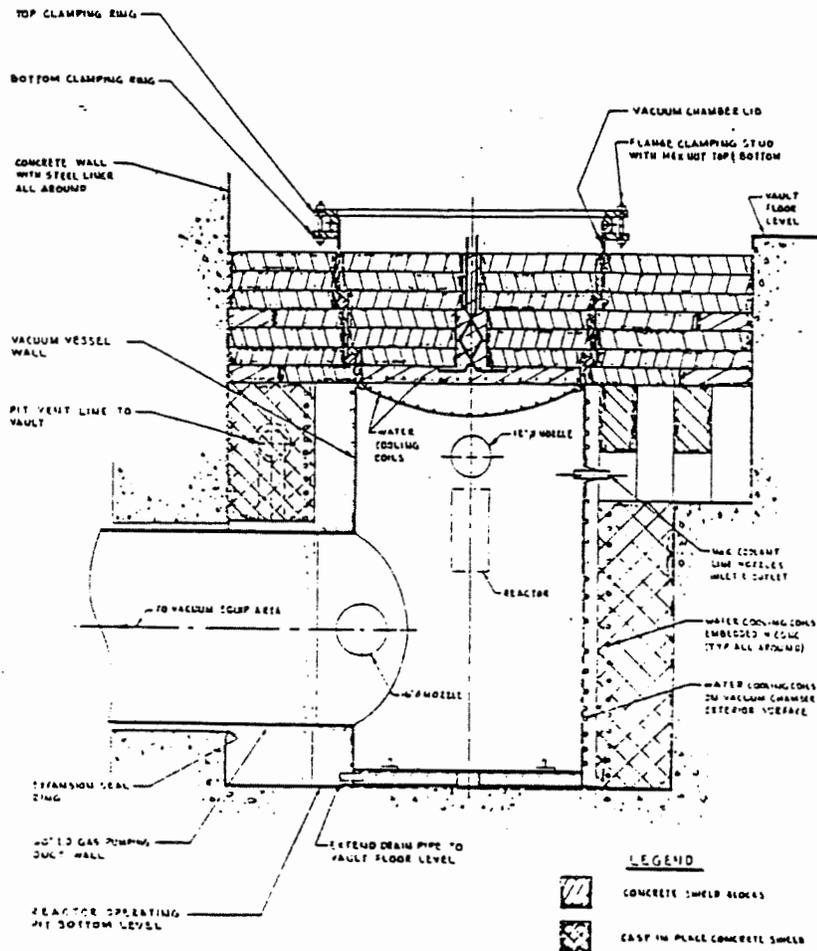
Figure 3–19. Cutaway and Cross–Section Views of S8DR Reactor and Shield Test Vacuum Vessel and Shielding
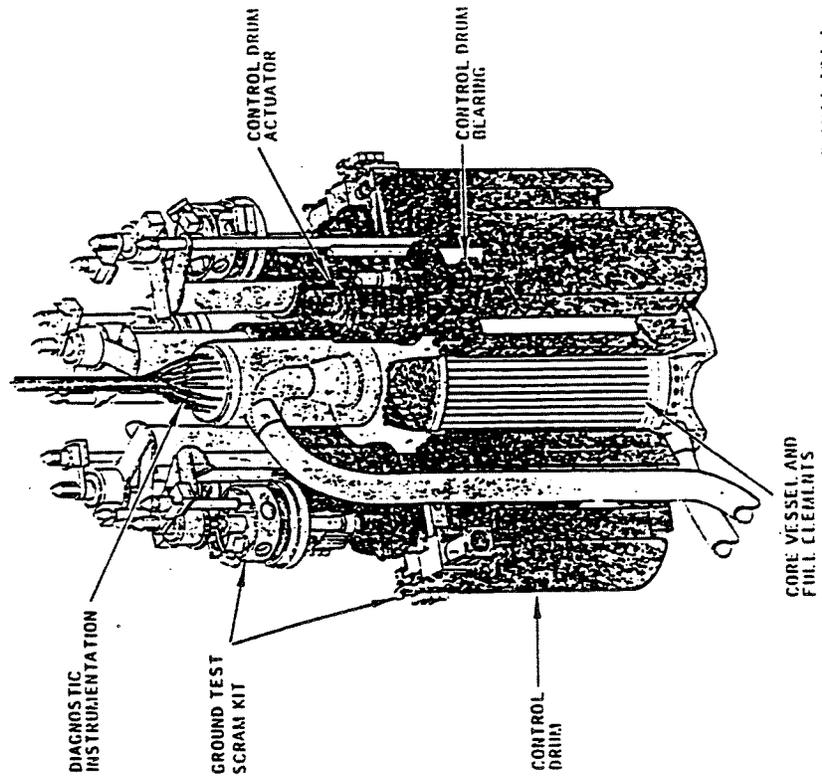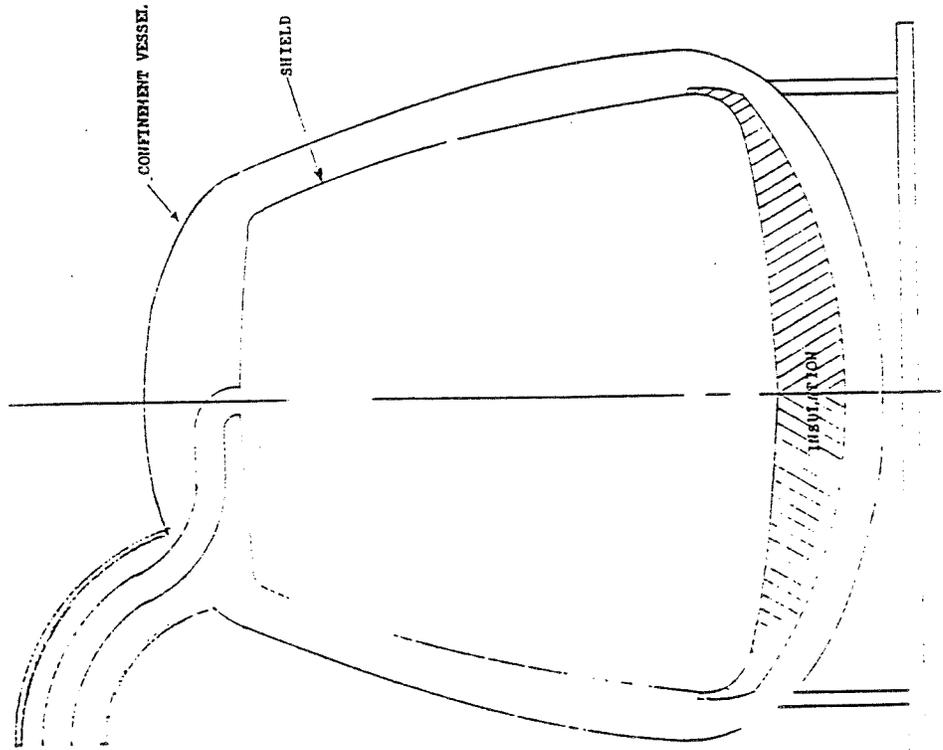
Figure 3-20. S8DR Reactor and Shield Test Arrangement

The reactor vacuum vessel was surrounded by high-density concrete shielding within the steel-lined vacuum pit (Figure 3-19). The pit walls and floor were constructed of ordinary concrete with 1% natural boron added. This served to reduce the thermal flux in the concrete and thereby reduce activation.

The Vacuum Equipment Room and the Pipe Chase Room were added to T059 in 1964 (before any reactor testing had been done), to provide the vacuum conditions to simulate operation of the S8DR in space. As part of this addition, a french drain was provided around the three external sides, below the foundations, to collect and remove any infiltrating groundwater. Tritium in groundwater at SSFL was first discovered in water from this french drain, in 1989.

Production of tritium in the S8DR operation at T059 occurred predominantly in the following ways:

1a. Neutron absorption by lithium-6 present in the concrete biological shield

This was calculated in the planning for D&D of T059 by using the neutron flux at the surface of the ordinary concrete at the bottom of the pit. A lithium concentration of 41 ppm in the concrete was assumed. Adjustment of the calculated value for current cross-section and half-life values, and for the observed lithium concentration of 6.87 ppm, gives a tritium activity of 0.9 Ci at shutdown (12/01/69). (It is this concrete shield that appears to have produced elevated tritium concentrations in adjacent groundwater in the french drain and in the groundwater monitor well RD-28.)

1b. Neutron absorption by lithium-6 present in the high-density concrete shield

This was calculated by using the neutron flux at the surface of the high-density concrete around the sides of the vacuum vessel. The lithium concentration was assumed to be the same as in the ordinary concrete. The tritium production is estimated to have been 62 Ci at shutdown.

1c. Neutron absorption by lithium-6 present in the sand fill of the Pipe Chase Room

The maximum specific activity was calculated for the sand surrounding the vacuum duct. Assuming a density equal to half that of solid concrete, and the lithium concentration measured in the concrete (6.87 ppm), an exposed length of duct of 12 ft and a flux of 1.8×10^{10} n/cm^2/sec, the tritium activity at shutdown is estimated to have been 9 Ci.

2. Neutron absorption by lithium-6 present in the soil surrounding the biological shield

The high-density concrete and ordinary concrete biological shields absorbed nearly all the neutrons that escaped from the reactor (approximately 60 neutrons in a million that were produced by the reactor passed into the soil). Assuming the lithium content of the soil to be 27.13 ppm (as measured for nearby soil), it is estimated that only about 0.009 Ci was produced during operation of S8DR.

3. Ternary fission

   This was calculated to be 3.0 Ci at the end of operation of S8DR at 1 December 1969.

4. Neutron absorption by lithium-6 present as an impurity in the NaK coolant.

   Because of the much smaller volume of NaK in the core of the S8DR, compared to sodium in the SRE, production of tritium in this manner would have been much less than the proportionate reduction from the SRE production based on energy released. Assuming a lithium content of 0.002 ppm in the coolant, this is estimated to be 0.0001 Ci.

5. Neutron absorption by lithium-6 in the lithium hydride shield

   Assuming a concentration of 99.9% Li-7 in the lithium, a neutron flux of 2.6×10^{10} $n/cm^2/sec$, and a neutron relaxation length in LiH of 13.5 cm, the tritium produced in this manner is estimated to have been 334 Ci at 12/01/69.

No other potential sources of tritium production at T059 were identified.

Summarizing the estimated tritium production at T059, at the time of reactor shutdown:

1a.	Ordinary concrete shield:	0.9 Ci, December 1969
1b.	High-density concrete shield:	62 Ci, December 1969
1c.	Sand in Pipe Chase Room:	9 Ci, December 1969
2.	Soil:	0.009, December 1969
3.	Ternary fission:	3.0 Ci, December 1969
4.	NaK coolant: maximum	0.0001 Ci, December 1969
5.	SNAP-8 shield:	334 Ci, December 1969

3.4 PRODUCTION OF TRITIUM AT T028

Building T028 was a facility for testing space-reactor shields and contained a swimming-pool reactor, which supplied thermal neutrons to a uranium fission-plate to serve as a source of fission-energy neutrons. This reactor was initially rated at 50 kW$_t$ and was designated the Shield Test Reactor (STR). In 1964, it was modified by replacing the TRIGA-type fuel elements with MTR-type elements and by addition of a recirculating water cooling-tower heat exchanger to permit operation at 1 MW$_t$, and was renamed the Shield Test and Irradiation Reactor, STIR. A sketch of the facility is shown in Figure 3-21, with the floor plan. Vertical sections are shown in Figure 3-22. The STR reactor is shown in Figure 3-23. The tank was filled with water, and was surrounded by ordinary concrete except toward the test area, where high-density concrete was used. The space around the concrete was filled with natural soil.
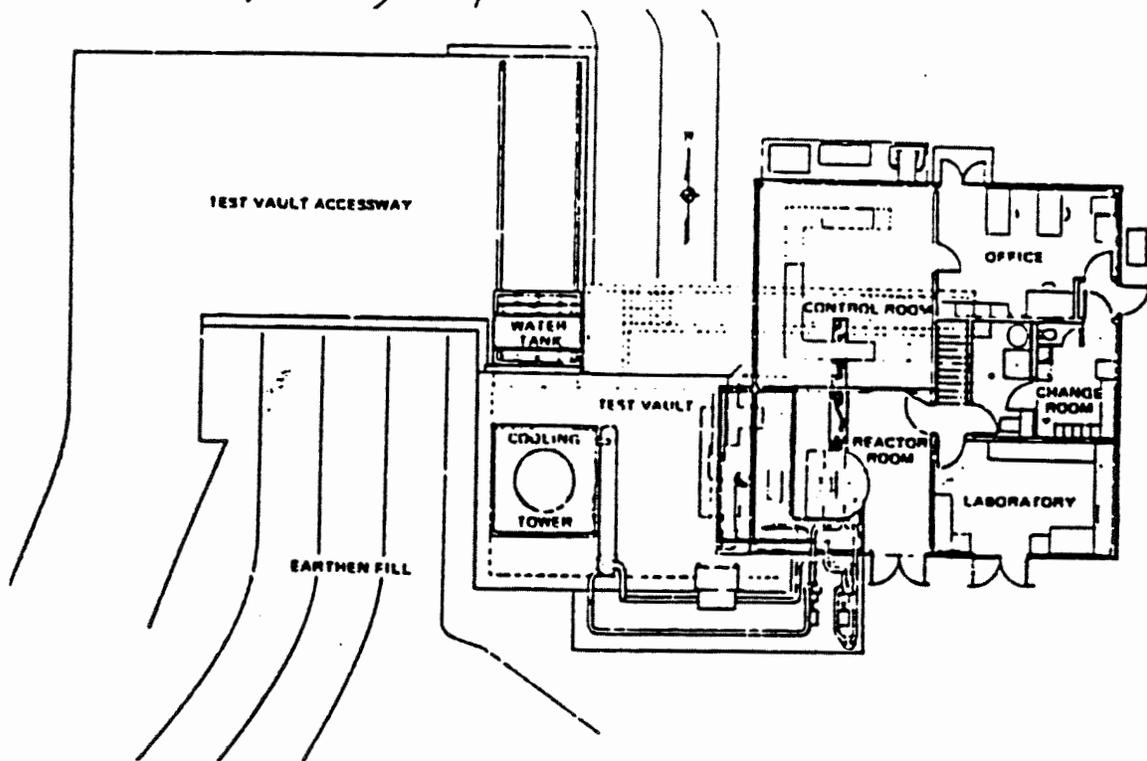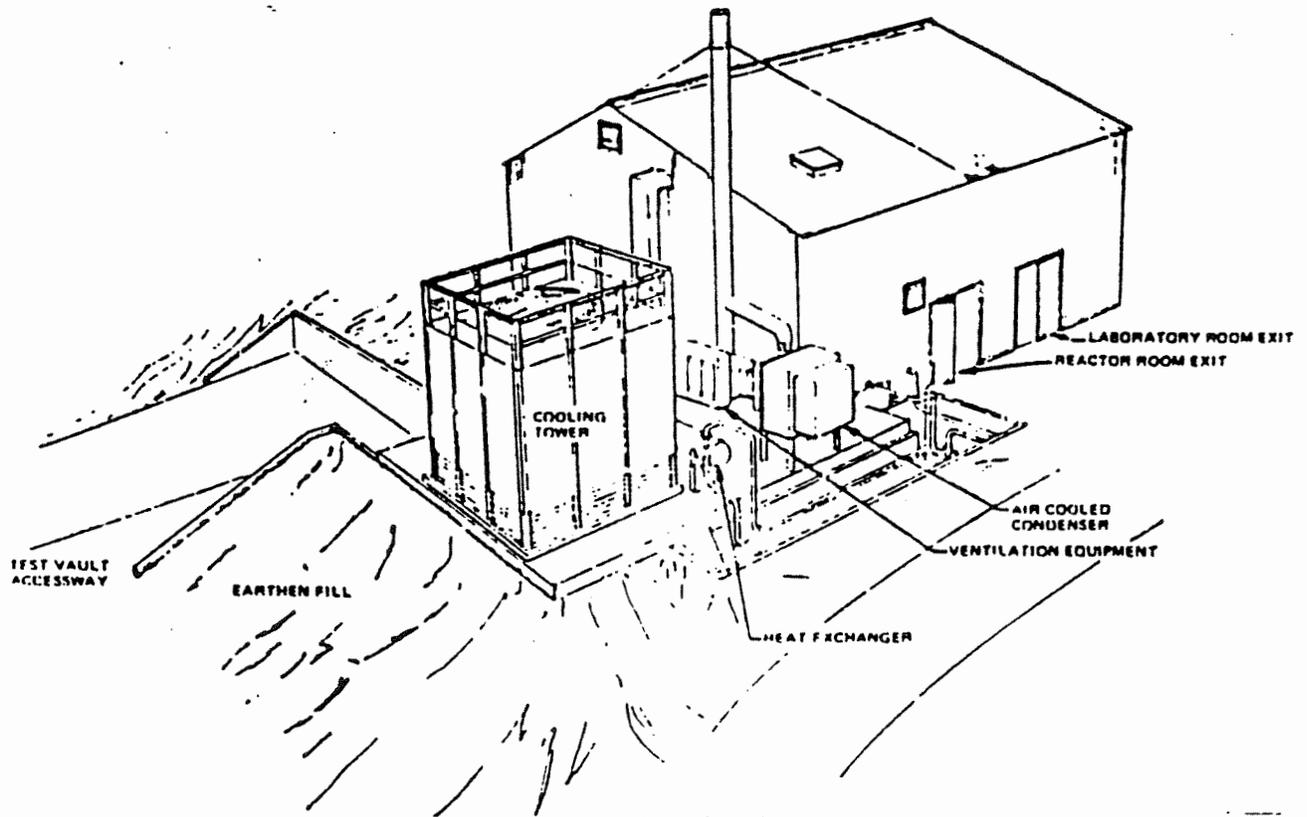
Figure 3–21. Perspective View and Facility Layout, T028
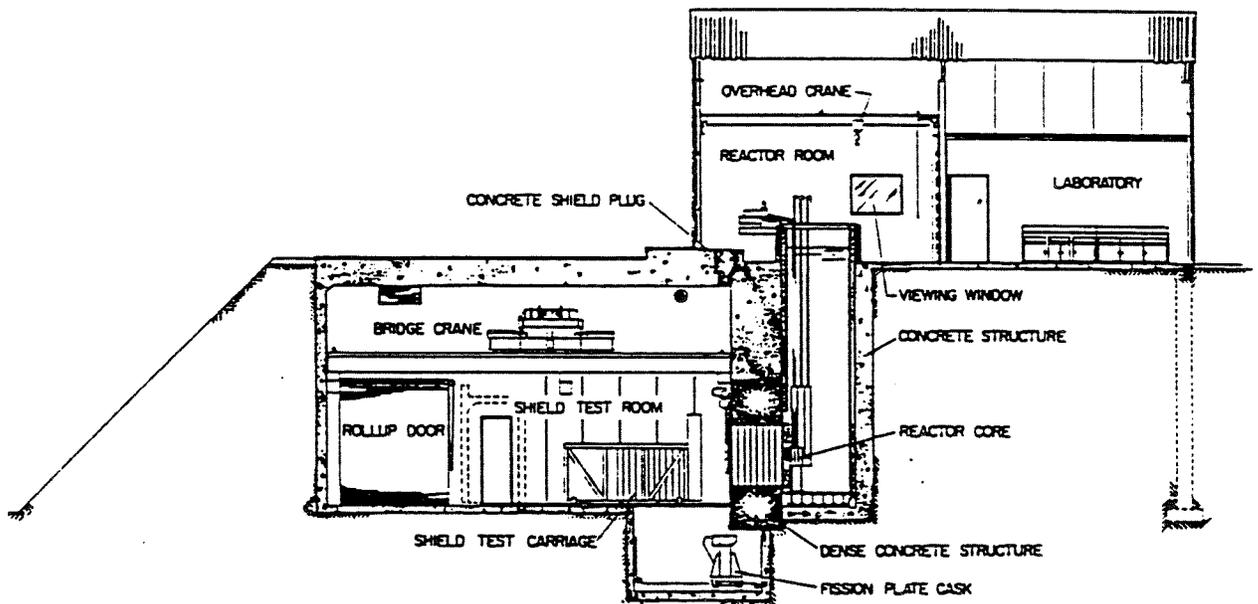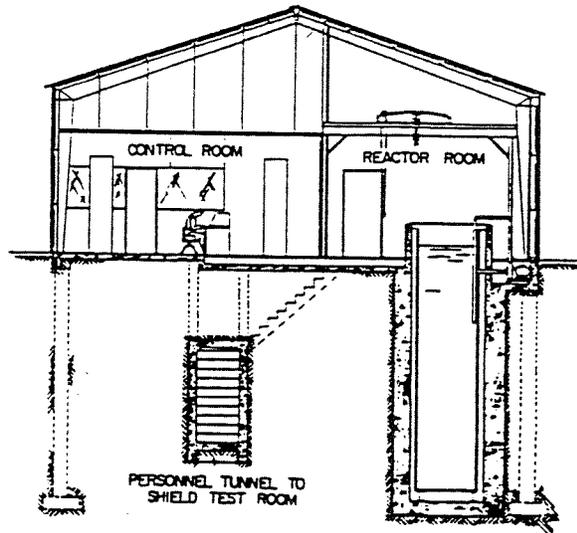
Figure 3–22. Sectional Views of T028
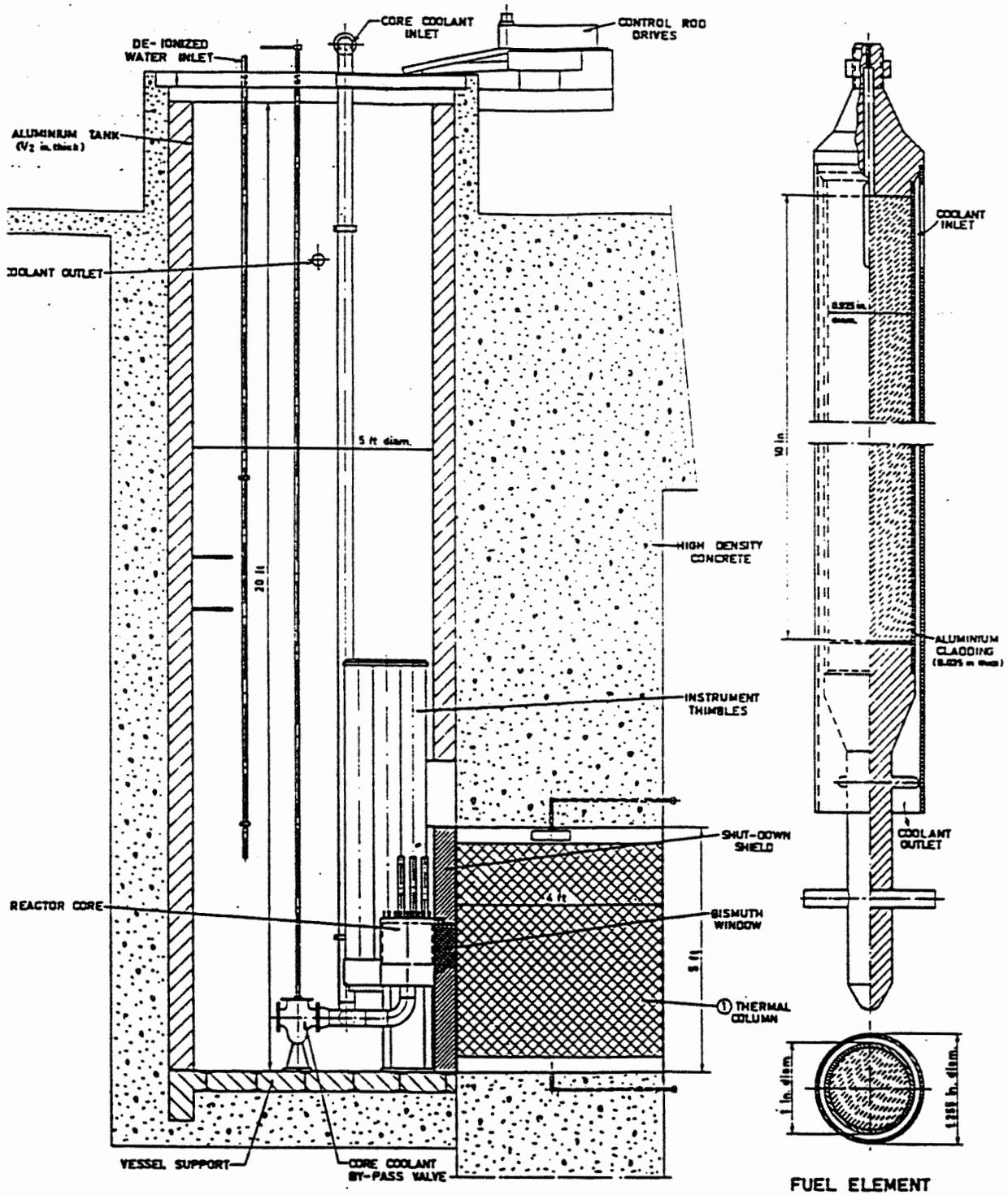
Figure 3–23. Shield Test Reactor (STR) and Fuel Element

Heat from reactor operation was discharged by means of a refrigerated cooler for operation at 50 kW$_t$ or by use of a cooling–tower heat exchanger at 1 MW$_t$.

Production of tritium during operation of STR and STIR at T028 occurred predominantly in the following ways:

1a. Neutron absorption by lithium–6 present in the concrete biological shield

This was estimated from an approximation for the flux at the concrete and the lithium content measured in the SRE and T059 concrete, and was found to be less than 0.2 microcuries. This is negligible.

1b. Neutron absorption by lithium–6 present in the high–density concrete shield

This was estimated for the region surrounding the thermal column, assuming a flux of 2 x 10^{11} n/cm^2/sec at 50 kW$_t$, a neutron relaxation length of 11 cm, and the same lithium content (per unit volume) as the ordinary concrete. This process produced less than 25 Ci at February 1974.

2. Neutron absorption by lithium–6 present in soil surrounding the biological shield

The thick water reflector of the STR and Shield Test and Irradiation Reactor (STIR) reactors absorbed nearly all the neutrons that escaped from the reactor, so that extremely few neutrons (less than 1 neutron per billion produced by the reactor) passed into the adjacent soil.

Production of tritium in this manner was negligible.

3. Ternary fission

This is estimated to be less than 0.4 Ci at shutdown of STIR in February 1974, including tritium produced in the fission plate.

4. Neutron absorption by lithium–6 present as an impurity in the graphite thermal column

Assuming a concentration of 1.2 ppm lithium in the reactor–grade graphite used for the thermal column and a flux of about 6 x 10^{11} n/cm^2/sec at 50 kW$_t$, this is estimated to have produced 1.2 Ci at July 1964. Operation of STIR would have resulted in an additional 28 Ci at February 1974, for a total of 29 Ci at that time.

5. Neutron absorption by deuterium present in water

This was estimated for a volume roughly equal to the reactor core volume, 4×10^4 cm^3, using the natural abundance of deuterium (1.5×10^{-4}), an absorption cross-section of 0.46 millibarn, and an average flux of 8.6×10^{11} n/cm^2/sec at 50 kW$_t$. The activity produced is estimated to be less than 0.0003 Ci, which is negligible.

6. Fast-neutron reactions with boron-10 in control rods

The STR/STIR used a boron carbide control rod, but because of the well-moderated flux and the small amount of reactivity controlled (0.6% Δk), tritium production by fast neutrons absorbed by boron-10 (0.0007 Ci) was negligible.

7. Neutron absorption by lithium-6 in lithium hydride shields

Occasionally, lithium hydride shields were tested at T028, using fission neutrons from the fission plate. Assuming that natural lithium was used, with 7.42% lithium-6, and by comparison with the estimated tritium production in the T024 operation of S10FS3, it is estimated that less than 600 Ci of tritium was produced in the various shields by February 1974.

The production of tritium by STR and STIR operation at T028 is summarized:

1a.	Ordinary concrete shield: maximum	negligible
1b.	High-density concrete shield: maximum	25 Ci, February 1974
2.	Soil: maximum	negligible
3.	Ternary fission:	0.4 Ci, February 1974
4.	Graphite thermal column:	29 Ci, February 1974
5.	Deuterium in water:	negligible
6.	Boron in control rods:	negligible
7.	Lithium hydride shield tests:	600 Ci, February 1974

3.5 PRODUCTION OF TRITIUM AT T024

Building T024 was built with two underground test vaults connected by a transfer cell, and was used to test S2DR in Vault 1 (1961–1962) and S10FS3 (SNAP-10 Flight System-3) in Vault 2 (1965–1966) at full power for greater than 10,000 hours. The layout of the building is shown in Figure 3–24, with the floor plans shown in more detail in Figure 3–25. The S10FS3 reactor was mounted on a lithium hydride shield and the power conversion system (Figure 3–26), within a vacuum vessel. The equipment arrangement for the S10FS3 test is also shown in Figure 3–25. The shield is shown in Figure 3–27.
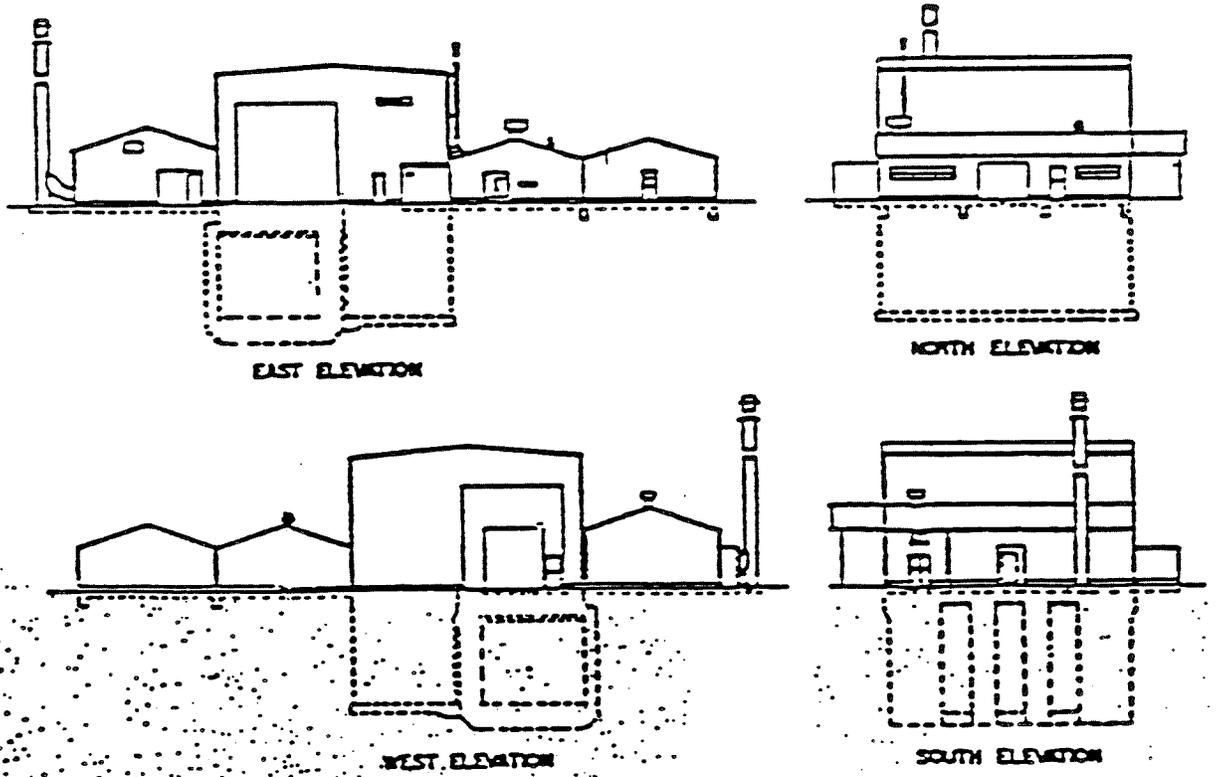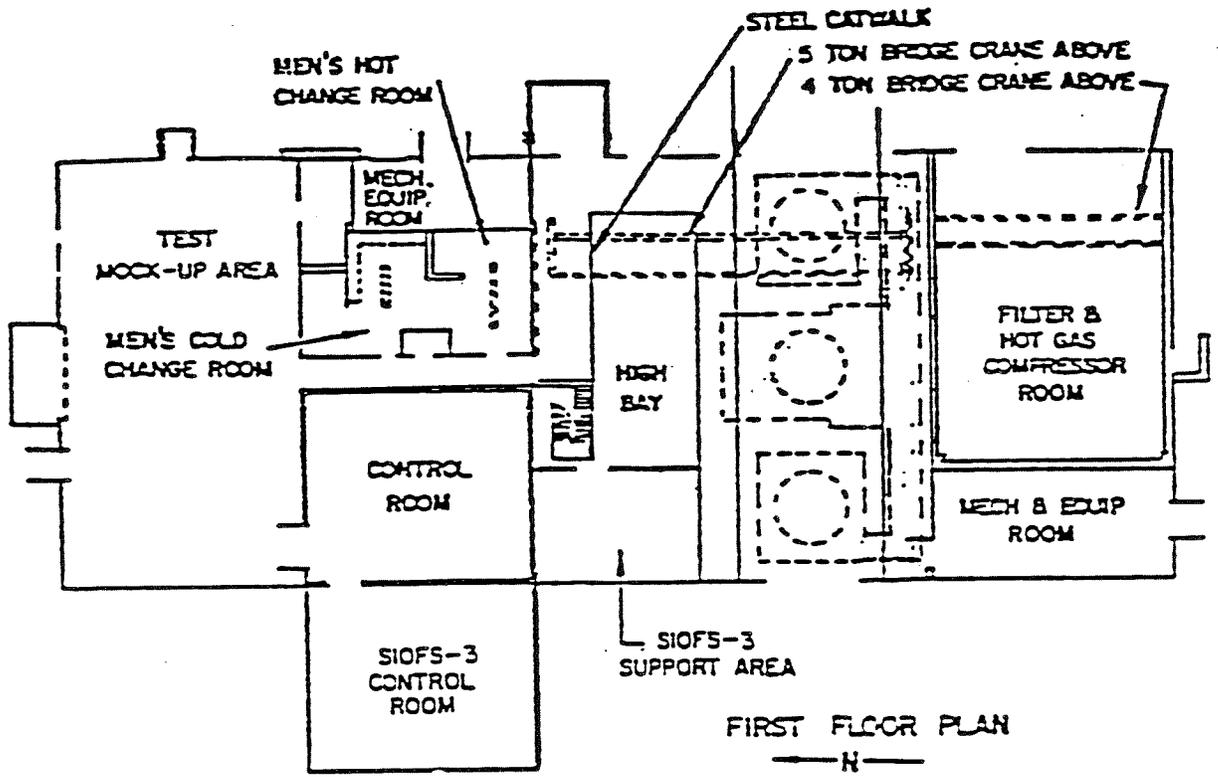
Figure 3-24. Layout of SETF, T024

INTERMEDIATE
FLOOR PLAN

BASEMENT PLAN

Figure 3–25. Test Area Arrangement, T024
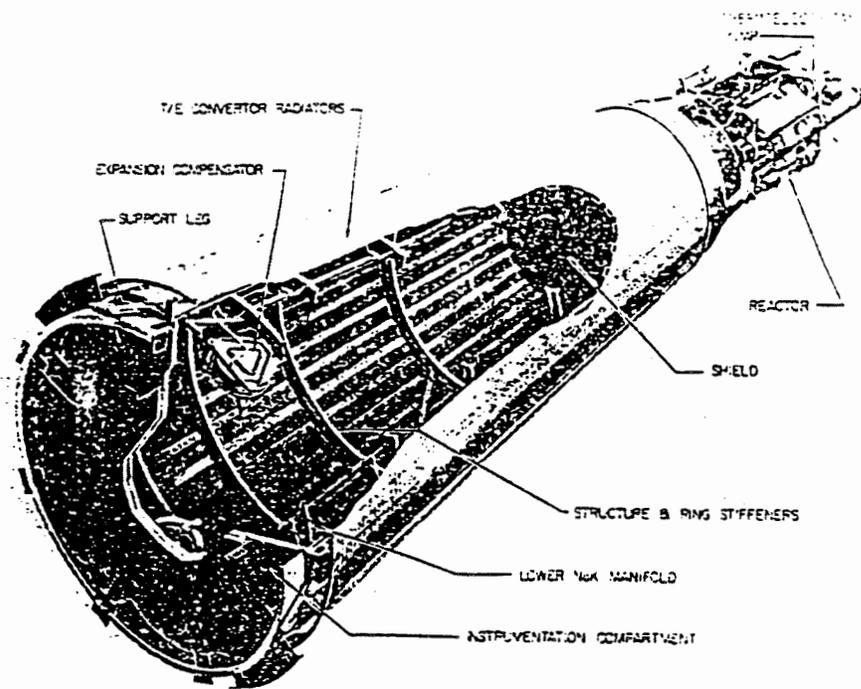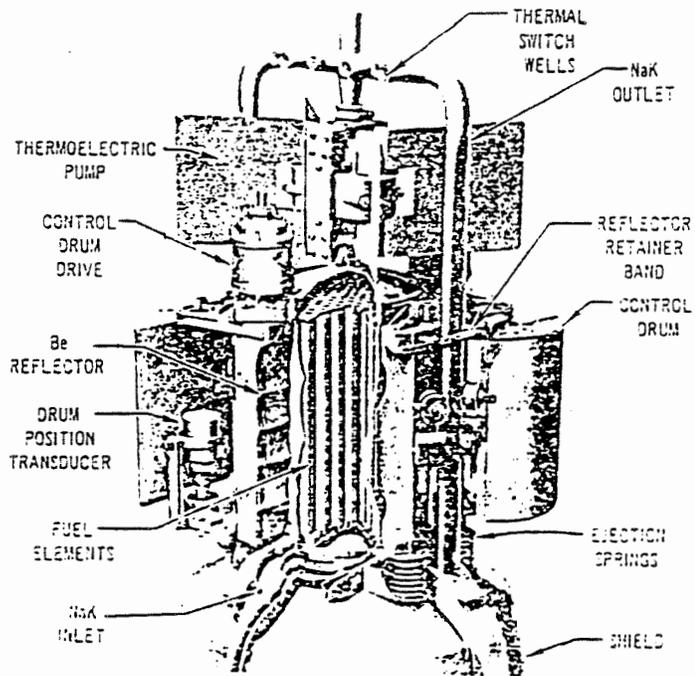
Figure 3-26. SNAP 10A Reactor and Integrated Power System
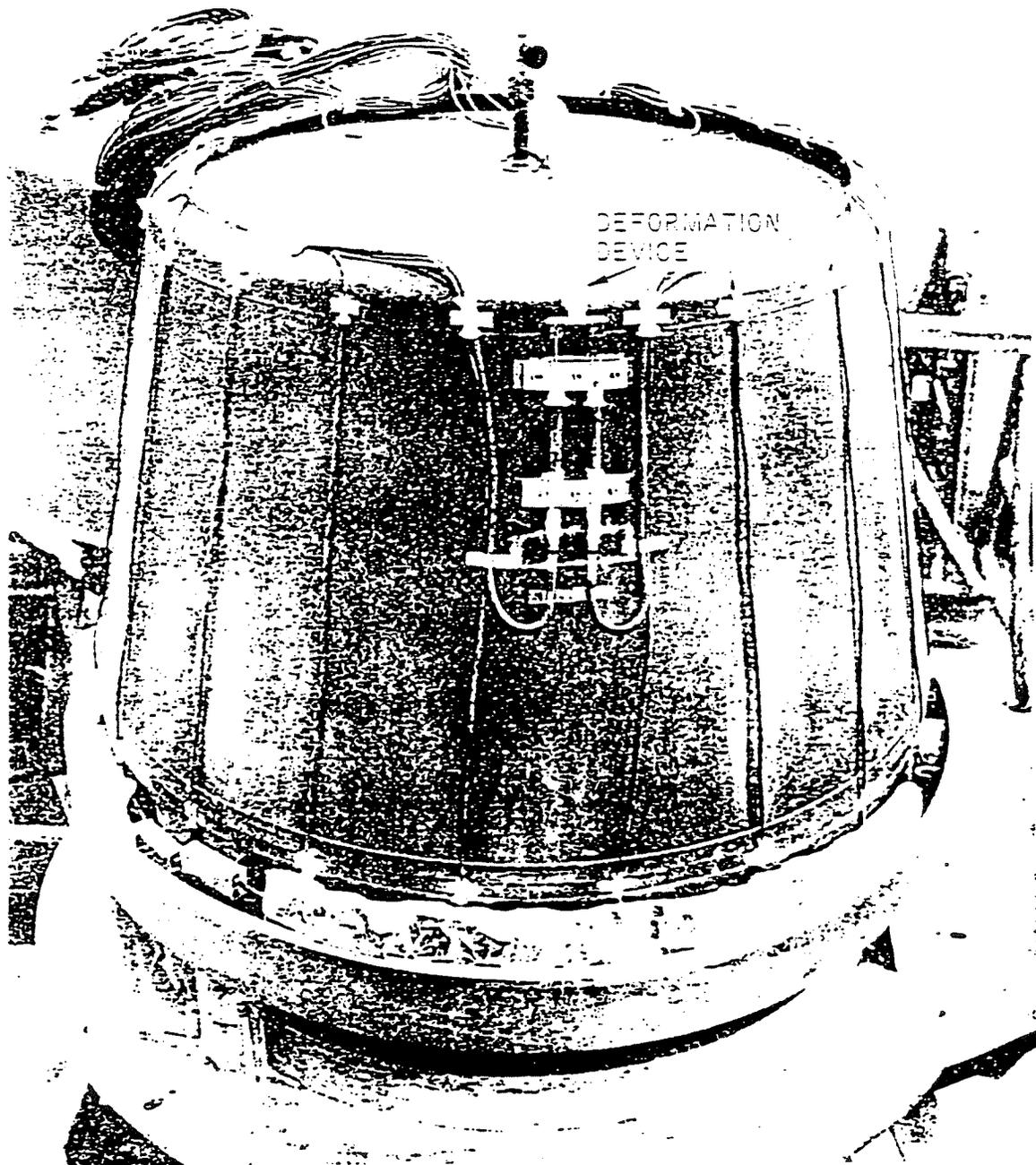
Figure 3–27. Shield and Confinement Vessel Base

The concrete shielding was composed of an inner region (2 feet thick) consisting of limestone aggregate, white cement, and boron frit, providing about 0.5% boron. The balance of the shield was made with ordinary concrete. Exterior walls that were exposed to significant neutron flux were composed of borated concrete throughout. Cooling water was circulated through tubes contained in the concrete, approximately 5 inches from the inner surface. The outer surface of the concrete is in contact with the ground.

As in all the SNAP reactors, the fuel was zirconium–uranium hydride, and the coolant was NaK. A cold trap was provided to maintain purity of the coolant. The fuel was clad with Hasteloy–N with a ceramic hydrogen barrier coating to reduce loss of hydrogen (and similarly tritium) from the fuel. The flight system shield was composed of lithium hydride encased in stainless steel. While the LiH shield used for the S8DR test in T059 had been fabricated from lithium that was depleted in lithium–6, to minimize heating of the shield due to the energy released in the $^6Li(n,t)^4He$ reaction, no record was found referring to such special considerations for any other shields. Scrap LiH from several shields was analyzed in 1992 and found to have the isotopic composition of natural lithium. Since the power levels of the other tests involving these shields were lower by a factor of ten than in the T–59 test, nuclear heating would not have been such a problem. Therefore, it seems likely that only the LiH shield used for the S8DR test in T059 utilized lithium that was depleted in Li–6, and therefore, the T024 shield is assumed to have been made with natural lithium. Like the shield tested in T059, the shield container may have been oxide–coated, but this was not evident from the documents reviewed.

Production of tritium by operation of S2DR and S10FS3 at T024 occurred predominantly in the following ways:

1. Neutron absorption by lithium–6 present in the concrete biological shield

   The concrete used for the inner 2 feet of the shield differed from the concrete usually used at SSFL in that limestone aggregate was substituted for the normally used granitic gravel. The natural lithium content of this concrete was measured in 1992 to be 6.2 ppm. Using this concentration and an average flux of 1.2×10^8 n/cm^2/sec at 37 kW$_t$ for the first 11 cm of concrete (and considering that this flux was uniform over all the walls, floor, and ceiling of the vault) the activity of tritium produced in the concrete of Vault 1 by operation of S2DR is estimated to be 0.16 Ci at shutdown in December 1962. In a similar manner, adjusting for power level and operating time, the tritium activity produced by operation of S10FS3 in Vault 2 is estimated to be 0.20 Ci at March 1966.

2. Neutron absorption by lithium–6 present in soil surrounding the biological shield

   The thick concrete biological shield with the added boron content absorbed nearly all the neutrons that escaped from the reactor (less than 1 neutron per million produced by the reactor reached the soil), and so the production of tritium in the soil was negligible.

3. Ternary fission

This was calculated to produce 0.21 Ci in the fuel of S2DR by December 1962, and 0.26 Ci in S10FS3 by March 1966.

4. Neutron absorption by lithium–6 present as an impurity in NaK coolant

Tritium produced in this manner was negligible for both reactors.

5. Neutron absorption by lithium–6 in the lithium hydride shield

This activity was estimated from the calculation for S8DR at T059, adjusting for power and time, for the S10FS3 operation. The natural isotopic fraction of 7.42 atom percent was used to estimate a production of 2,200 Ci in the S10FS3 shield.

Summarizing tritium production at T024:

1a.	Concrete shield, Vault 1, at S2DR shutdown	0.16 Ci, December 1962
1b.	Concrete shield, Vault 2, at S10FS3 shutdown	0.20 Ci, March 1966
2.	Soil: maximum	negligible
3a.	Ternary fission, S2DR, at shutdown	0.21 Ci, December 1962
3b.	Ternary fission, S10FS3, at shutdown	0.26 Ci, March 1966
4a.	NaK Coolant, S2DR:	negligible
4b.	NaK Coolant, S10FS3:	negligible
5.	S10FS3 shield:	2200 Ci, March 1966

3.6 SUMMARY OF TRITIUM PRODUCTION

The maximum tritium production for the SSFL reactors is shown in Table 3–2. This table summarizes the tritium production by the material in which it was produced.

Thus, in a period of roughly 11 years, the SSFL reactors produced about the same amount of tritium as a typical power reactor does in 9 months, or as much as is produced by cosmic rays in the atmosphere in about 1 day. Nearly all of this (95%), was produced in well–contained materials (graphite moderator logs clad in zirconium, lithium hydride clad in stainless steel) that minimized releases to the environment. Nearly all these materials have been removed from the SSFL facilities and disposed of at authorized radioactive disposal sites.

The different conditions under which tritium was produced can be grouped into four categories:

1. Contained – in which the tritium is produced within closed components, such as metallic fuel elements, clad moderator, test shields.

Table 3-2. Tritium Production in Reactor Materials (curies)*
(Combined maximum production at each facility, at dates of shutdown)

Material	Facility					Totals
	T143	T010	T059	T028	T024	
Ordinary concrete	5.6	7.8	0.94	0.000	0.37	15.0
High-density concrete	–	–	62.0	25.0	–	87.0
Soil	0.000	0.039	0.009	0.000	0.000	0.048
Sand			8.9			8.9
Fuel	111.0	2.8	3.0	0.38	0.47	120.0
Graphite	9,200.0	–	–	30.0	–	9,200.0
Coolant	0.040	0.000	0.000	0.000	0.000	0.040
Boron	69.0	–	–	0.001	–	69.0
LiH shields	–	–	330.0	590.0	2,200.0	3,100.0
Total	9,400.0	11.0	410.0	650.0	2,200.0	
Grand Total						13,000.0

* Values are rounded-off to two significant figures.
Totals may differ from the sum of the individual entries.
Values less than 0.001 are shown as 0.000.

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2. Confined – in which tritium is produced in reactor coolant that is confined by the coolant system.

3. Isolated – in which the tritium is produced in materials that are separated from the environment, such as separate (interior) high-density concrete shielding.

4. Exposed – where the material containing the tritium is in direct contact with air, water, or ground.

The first category, contained, includes tritium produced in fuel, the SRE moderator and control rods, and the SNAP (Systems for Nuclear Auxiliary Power) reactor flight shields. Confined tritium is produced in the sodium, NaK, or water coolants. Isolated tritium is that produced in shielding materials that are separated from the environment, such as the high-density concrete at T059 and at T028. Exposed refers to those situations where tritium is produced in concrete shielding that is in contact with the ground, or in the soil itself. The release of the tritium produced in these different materials can vary greatly and is discussed in the next section.

In many cases, the material containing the tritium was removed and disposed of as radioactive waste after production of the tritium. Table 3-3 summarizes the maximum inventory of tritium by the four categories discussed above, listed according to facility. This is somewhat less than that produced, as shown in Table 3-2, because of radioactive decay.

Table 3-3.   Maximum Tritium Content of SSFL Reactor Facilities (curies)*

Condition	Facility					Totals
	T143	T010	T059	T028	T024	
Contained	8,200.0	2.6	340.0	620.0	2,200.0	11,000.0
Confined	61.0	0.000	0.000	0.001	0.000	61.0
Isolated	–	–	71.0	25.0	–	96.0
Exposed	5.0	7.7	0.95	0.000	0.34	14.0
Maximum Inventory	8,200.0	10.0	410.0	650.0	2,200.0	11,000.0
Remaining on 5/1/92		0.008	0.27		0.078	0.36

* Values are rounded-off to two significant figures.
  Totals may differ from the sum of the individual entries.
  Values less than 0.001 are shown as 0.000.

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The tritium activity shown as remaining is in unexcavated soil at the site of T010, in concrete and soil still in place at T059, and in the concrete shielding of T024. All material that could have contained any significant amount of tritium at T143 and T028 was removed during the decommissioning of these facilities.

D641-0036/tab

4.0 RELEASES OF TRITIUM

Tritium produced in the various materials at SSFL reactor facilities may find its way to the environment with greater or lesser difficulty, depending on the material it is produced in and the barriers preventing its release. Hot, unoxidized stainless steel does not provide much of a barrier, while cold, oxidized stainless steel does. Concrete exposed to groundwater provides a relatively easy path, while graphite encased in zirconium does not. Each of these conditions will be considered in estimating releases of tritium to the environment from the SSFL reactors. Reasonable values have been assumed for the parameters used in these estimates. An attempt has been made to estimate realistic releases, rather than maximum conceivable releases.

Release of tritium from the SSFL reactors was primarily via two separate routes: releases to groundwater from irradiated concrete and soil, and releases to surface water from the reactor by way of the heat–transfer system. The groundwater releases will be considered for all reactors. The surface water release will be discussed in detail for the SRE. Diffusion of tritium produced in the concrete shield and the adjacent soil provides a pathway for entry of tritium into the groundwater.

It is assumed that the concentration of tritium in the concrete that is accessible to the groundwater (that is, at the outer surface of the shield, in contact with the soil) can be represented as the average concentration with an exponential approach in time, governed by a diffusion constant. The production of tritium in the concrete, as discussed in the previous section is a function of the neutron flux escaping from the reactor containment into the concrete, the concentration of lithium in the concrete, and the time during which this occurs. As the reactor is operated, the amount of tritium produced in the concrete increases. Some of this tritium changes into He–3 by radioactive decay, and is of no further interest. At the same time, individual tritium atoms exchange with (ordinary) hydrogen atoms in the water of hydration of the concrete and become part of this bound water. These tritium atoms then gradually diffuse from regions of high tritium content to regions of low (or zero) tritium content. (While the tritium atoms can (and do) move in any direction, more atoms will be leaving a region of high concentration than will be entering it, thus causing a net movement away from the reactor.) This effect is modeled as an exponentially asymptotic approach to a uniform concentration of tritium in the concrete. This concentration depends on the amount of tritium produced in the shield, the thickness of the shield (the distance through which the tritium atoms must diffuse), the time since the tritium was produced, and the rate of diffusion. The concentration of tritium at the outer surface of the shield is assumed to be immediately available to groundwater in the surrounding soil. If this groundwater is moving past the concrete shield, the tritium concentration in the groundwater will remain relatively constant and low. If the groundwater is not moving, the concentration, in a thin region of soil surrounding the shield, will gradually build up to the level in the concrete. Tritium will leave this region by slow diffusion in the groundwater.

Similarly, tritium produced by neutron absorption by the lithium in the grains of soil, diffuses from the soil grains to the surrounding groundwater. (In this analysis, it is assumed that the soil (and the reactor shield concrete) are in contact with groundwater).

4.1 RELEASE OF TRITIUM FROM THE SRE

In many regards, the SRE and the Experimental Breeder Reactor-II (EBR-II), operated at the Idaho National Engineering Laboratory, are quite similar. Both used (the EBR-II is still in operation) metallic fuel in stainless steel cladding with molten sodium as a heat transfer material (coolant). EBR-II operates at a somewhat higher power, rated at 62.5 MW_t compared to the 20 MW_t rating for SRE, and the actual SRE average of 9.2 MW_t. The SRE was a "thermal" rather than "fast" reactor, using graphite in zirconium cans to moderate the fission neutrons. The EBR-II has no moderator. The EBR-II differs from the SRE by being a "pool" reactor, with the IHX and the primary pump contained, with the reactor itself, in a large, sodium-filled vessel, while the SRE was a "loop" reactor, with the primary coolant (liquid sodium) flowing in a piping loop outside the reactor vessel, through the IHX and the primary pump. This difference in arrangement would affect the relative releases to the air and the steam generator. In the SRE arrangement, the coolant boundary (pipe walls, pump and valve bodies, the IHX shell) is more exposed to the atmosphere than in the EBR-II arrangement. This would result in a somewhat greater fractional release of tritium (predominantly as elemental HT rather than as HTO) to the air, with a consequently lower fractional release to the water system. Thus, calculation of releases from the SRE reactor coolant system to the water/steam system, based on EBR-II experience, would tend to overestimate the actual releases. However, considerable information has been obtained on tritium released from EBR-II and this can be directly scaled by the ratio of powers to estimate releases from the SRE. The SRE was used to generate electricity effectively one-half of the time it was at power. During this time, there was the potential for release of tritium, produced in various ways within the reactor system, to surface water by way of the heat transfer system, via discharges ("blowdown") from the cooling tower. This pathway can be estimated by use of the EBR-II monitoring data. There was no monitoring for tritium during operation of the SRE (or any of the SSFL reactors) since the quantities produced and released by such low-power reactors would not be significant.

Decommissioning of the facility was started in 1974, with the radioactive sodium systems removed in 1976, the reactor vessel in 1977, and the biological shield and surrounding soil in 1979. Decommissioning was completed in 1982.

(The similarity between the SRE and EBR-II has been used to estimate tritium releases to surface water from the SNAP reactors, as well. The calculations of releases to surface water assume steady operation with the tritium inventory at shutdown for each reactor. This overestimates releases and concentrations by about a factor of 2.)

The model used to calculate tritium releases to surface-water by way of the cooling system, based upon observed released from EBR-II:

Transfer from reactor to surface water

$$= \text{(EBR-II fractional release)} \left(\frac{\text{release to water}}{\text{release to water and air}} \right)$$

$$X \left[\text{(tritium in fuel)} \left(\frac{\text{1/SRE fuel diameter}}{\text{1/EBR-II reactor fuel diameter}} \right) \right.$$

$$+ \text{(tritium in graphite) (release fraction to coolant)}$$

$$+ \text{(tritium in boron) (release fraction to coolant)}$$

$$\left. + \text{(tritium in coolant)} \right]$$

A tritium atom produced by ternary fission in the fuel, regardless of its point of origin, must migrate a greater distance to reach the surface of a larger fuel rod than a smaller fuel rod. Tritium migrates slowly in uranium metal (and presumably in thorium metal, SRE Core 2, as well) and so the amount of tritium reaching the surface will vary roughly inversely with the migration distance. This distance is proportional to the fuel rod diameter and so the ratio of the inverse fuel diameters has been used to adjust the EBR-II release rate to an estimated value for the SRE.

The tritium concentration in the discharged water is calculated by dividing the transferred (released) activity by the volume of makeup water supplied to the cooling tower.

4.1.1 Tritium Produced Within the Reactor

The observed tritium release from EBR-II to the environment was found to be about 0.2% of the production rate in the reactor. For the SRE, the annual production rate in the fuel during operation was 55 Ci/year. This would result in a release of 0.11 Ci/year, with about 0.01 Ci/year in the water system and 0.1 Ci/year to the atmosphere, based on EBR-II experience. However, adjusted to account for the more massive fuel used by the SRE, and including tritium production in the graphite moderator and the boron in the control rods, the actual release rate is estimated to be 0.0024 Ci/year to surface water. These releases would have occurred only during times of operation at significant temperature to promote diffusion of the tritium through the various barriers.

A somewhat greater rate of diffusion from the fuel to the coolant could have occurred for the overheated fuel assemblies that were involved in the fuel cladding failure incident in 1959. (In this incident, 13 of the 43 installed fuel assemblies became overheated due to blockage of the coolant channels while operating at full power, resulting in melting of the upper ends of the fuel rods and the stainless steel cladding.) This damage occurred progressively and was not disruptive, and so there is no reason to expect that the normal chemical processes (tritium reaction with the sodium and tritium absorption by the zirconium cladding of the moderator) would not have continued to be effective in minimizing releases of tritium from the fuel to the environment.

Tritium in the graphite moderator would have been released to the environment at a much lower rate than that produced in the fuel for several reasons: the volume–to–surface ratio for the graphite moderator is over 7 times that of the fuel, essentially meaning that the tritium produced in the graphite must diffuse, on average, 7 times as far to escape from the graphite as from the fuel. The graphite was about 250°C cooler than the fuel, making the tritium much less mobile. The diffusion of tritium in graphite may be similar to diffusion in boron – this is estimated to be about 0.01 cm per year at a temperature of 390°C. Finally, the zirconium cladding of the graphite forms a stronger chemical bond with tritium than does the sodium coolant. It is estimated that approximately 0.001 of the tritium produced in the graphite diffused to the zirconium, and about 1/3 of that amount passed into the sodium.

A significant difference between EBR–II and the SRE is the presence of a large amount of zirconium cladding on the graphite moderator logs of the SRE. The great chemical affinity of zirconium for hydrogen (and so, for tritium) would probably deplete the tritium content of the sodium coolant in the SRE, perhaps by a factor of about 3. This reduction in tritium concentration in the sodium has been used to reduce the estimate of environmental releases that are shown here, and so these values are reasonable estimates of the releases that might have occurred.

Using the release rate of tritium through the steam system, calculated by the equation discussed above, and the water makeup rate, the equilibrium tritium concentration in the cooling tower blowdown water would have been roughly 24 pCi/L at the end of 1 year operation. During this time the concentration of tritium in surface water, due mainly to weapons testing, averaged several thousand pCi/L.

The releases to the surface water would have been approximately 0.0024 Ci for Core I and the same for Core II. Surface water at the SRE was collected in the SRE pond, where the tritium would have been diluted in about 7,000 gallons of water (containing several thousand pCi/L of tritium from weapons testing). Water from this pond was pumped, as necessary, to surface drainage for transfer to Pond R–2A, where any excess (artificial) tritium would have been further diluted in about 2.5 million gallons of water. Since much of the water in Pond R–2A came from deep pumped wells, this water would have contained very little of the weapons–testing tritium, and so releases from this pond would have had tritium concentrations far less than normal surface water at the time.

4.1.2 Tritium Produced in the Concrete Biological Shield and Adjacent Soil

The SRE concrete shield was approximately 4 ft thick and remained in place, in contact with the ground and any groundwater that was present after terminating all reactor operation. It was removed in February 1979, and sent to Beatty, Nevada, for disposal. During operation of the reactor and the coincidental production of tritium and after shutdown of the reactor, the tritium would have slowly diffused in the concrete. As the tritium concentration at the surface of the concrete increased, more tritium would have been available for transfer into the groundwater. Tritium in the groundwater would have diffused away from the shield if the water was stationary, or would have moved with the groundwater if the groundwater was passing by.

The combined release rate, from the concrete and the soil, to the groundwater, is shown as a function of time in Figure 4–1. This shows a peak release rate of 22 μCi/year (0.000022 Ci/year) in 1964 primarily as a result of the rapid release of tritium produced in the soil during operation of the reactor. The abrupt end of the release after 1979 corresponds to the removal of the concrete shield and the adjacent soil.

Figure 4–1.   Estimated Tritium Release Rate from SRE to Groundwater

4.2   RELEASE OF TRITIUM FROM T010

4.2.1   Tritium Produced in the Fuel

In estimating the release of tritium from the fuel, the absence of cold traps and the presence of zirconium were considered in adjusting the EBR–II release rates. Cold traps in EBR–II were up to 50% efficient (compared to the theoretical removal rate). The zirconium is estimated to holdup approximately 2/3 of the tritium produced. In addition, a vitreous coating was used on the inside of the SER fuel cladding, and a ceramic coating was used for S8ER, to reduce hydrogen (and therefore tritium) loss from the fuel to an overall loss of approximately 3%.

Because the SNAP reactors were very small, with a core roughly 9 inches in diameter and 17 inches high for S8DR for example (while LWRs have dimensions that are typically 10 times these), the reactivity was very sensitive to the hydrogen (moderator) content. Hydrogen lost from the fuel during operation of S8DR for 6,688 hours amounted to about 2.4% and caused a loss of reactivity of 2.4%. Since an excess reactivity of 10% could be build into a SNAP reactor, loss of 10% of the hydrogen

would shut down the reactor. The other SNAP reactors lost less of their hydrogen (and fission–produced tritium) because of lower operating powers and shorter test runs. The vitreous and ceramic coatings were intended to impede the loss of hydrogen, so as to prolong the operational life of the reactor. Cracks and defects in these coatings (and in the fuel cladding) would have led to a more rapid release of hydrogen than desired, but the amount released (of hydrogen and tritium) could not have exceeded a few percent. (This release was determined to be 2.4% for the S8DR operated in T059 and would have been similar for the S8ER in T010.)

There was no release of this tritium to surface water.

4.2.2 Tritium Produced in the Concrete Biological Shield and Adjacent Soil

The shield at T010 was roughly 2 feet thick and it remained in place after the end of reactor operations, until its removal, intact, in February 1978. It was in contact with backfill soil and any groundwater that might have been present. The french drain around the base of this shield was equipped with a pump to automatically transfer any groundwater that was collected in the drain to a holdup tank for sampling, analysis, and disposal. No record was found of this tank ever needing to be sampled, during operation of the facility. Therefore, it is thought that the T010 ground was generally dry. However, two leaks in cooling lines, once below the floor and once in the ground around the containment vessel, are known to have occurred and could have saturated the adjacent ground, promoting release of tritium.

The combined tritium release rate, from the concrete shield and the adjacent soil to the groundwater (assumed to be present) is shown in Figure 4–2. This shows a peak release rate of 0.0047 Ci/year in 1965 due to release of tritium produced in the soil during operation of the reactor, with a nearly complete end of the release after 1978, following removal of the shield.

An additional potential for release of tritium to surface water occurred on removal of the biological shield from the ground, for disposal as radioactive waste. The shield was excavated and removed on 16 February 1978, and stored outside at RMDF. On or before 22 May 1978, the shield was thoroughly wrapped in plastic sheet, essentially isolating it from the environment. During the time it was exposed, rain fell on the shield, and could have extracted a small amount of tritium from the concrete. Rainfall runoff from the RMDF is collected in a sump and pumped to surface drainage, to ultimately flow to Pond R–2A. The concentration of tritium in the runoff water would have been less than the possible groundwater concentration when the shield had been in the ground, further reduced by dilution by the ratio of rain falling on the shield relative to that falling on the RMDF drainage area.

The rainfall runoff concentration estimated in this manner is roughly 80 pCi/L. Environmental tritium is estimated to have been about 100 pCi/L in surface water at the time, and 100 to 200 pCi/L in rainfall.
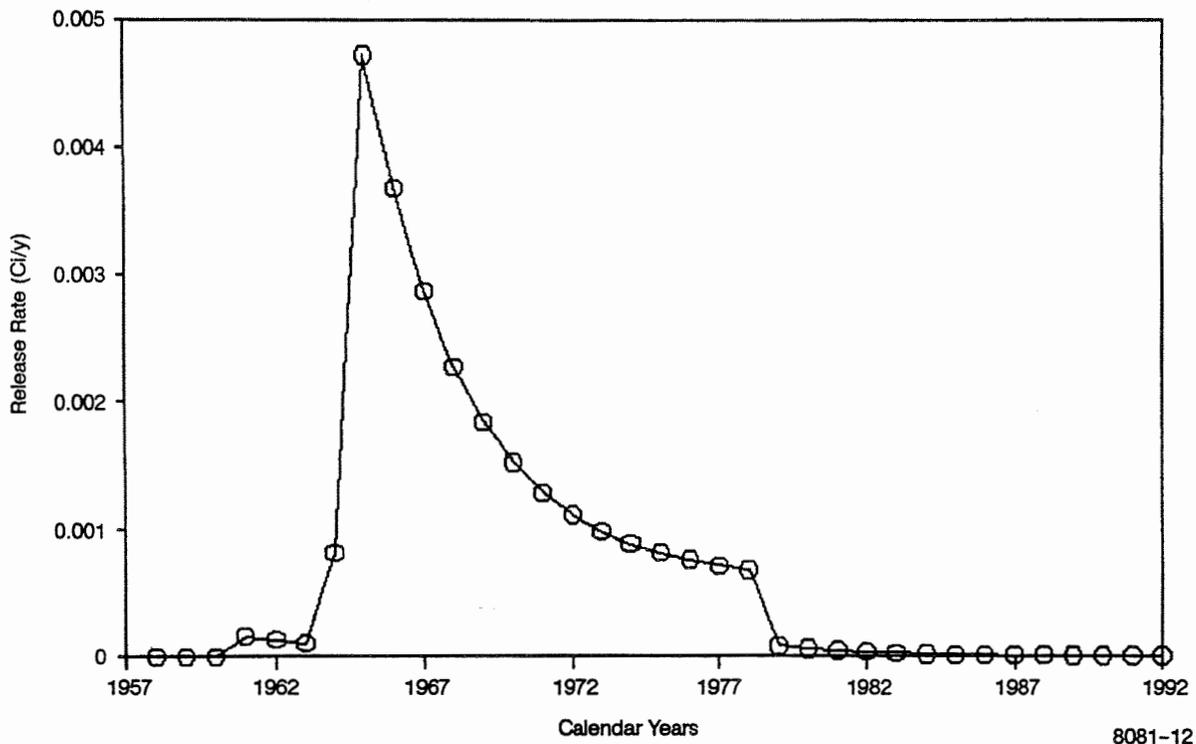
Figure 4–2. Estimated Tritium Release Rate from T010 to Groundwater

4.3 RELEASE OF TRITIUM FROM T059

It was at this facility, in water from the french drain around the west end of the basement, that tritium was first detected. Considerations for release of tritium are closely similar to those for Building T010.

4.3.1 Tritium Produced in the Fuel

Release of the tritium produced in the fuel was minimized by means of a very effective ceramic barrier on the inside of the fuel element cladding, limiting hydrogen (and tritium) loss to 2.4%. As discussed above, the physical necessity to maintain the hydrogen content of the fuel elements, for the purpose of neutron moderation, means that no more than a few percent of the hydrogen (or the tritium produced in the fuel) could be released. Flow–through cold traps in both primary and secondary coolant circuits captured most of the tritium (and hydrogen) released to the coolant.

There were no releases of this tritium to groundwater or surface water.

4.3.2 Tritium Produced in the Lithium Hydride Shield

Considering the relatively low temperature of the shield and the oxidized surface of the stainless steel of the shield container, release of this tritium is estimated to be less than 1 Ci. This would have been released to the vacuum system and would have become bound with the hydrocarbon lubricant in the pumps. Some small amount would have been discharged to the atmosphere. This release is estimated at 0.01%, based on the observed partition of radon gas between water and air. Chemical

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bonding of tritium to hydrocarbon molecules would be expected to further reduce this estimated release.

4.3.3 Tritium Produced in the Concrete Biological Shield and Adjacent Soil

The outer, ordinary concrete biological shield at T059 is still in place in the ground and is in contact with groundwater. Monitoring wells around the facility show an average tritium concentration of approximately 500 pCi/L.

The combined tritium release rate, from the ordinary concrete and the adjacent soil to the groundwater is shown in Figure 4–3.
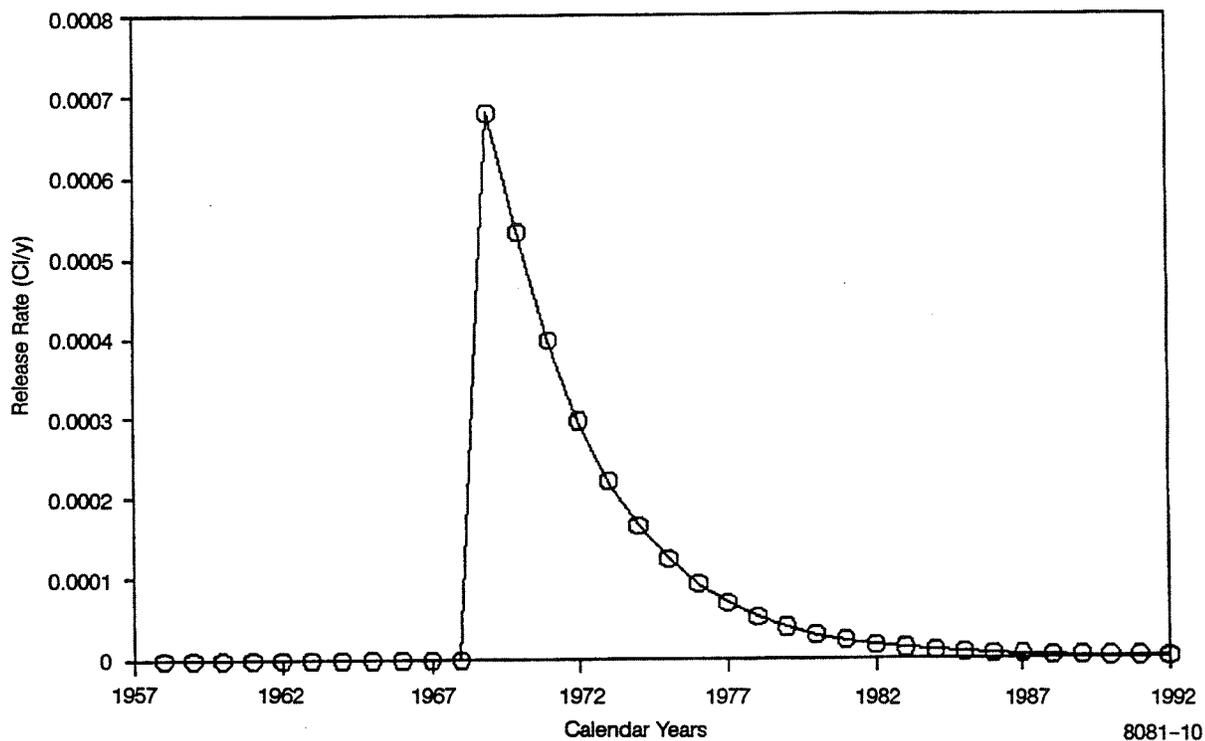
Figure 4–3. Estimated Tritium Release Rate from T059 to Groundwater

4.3.4 Tritium Produced in the High–Density Concrete Shield

Since this concrete is separated from the environment by the mild steel pit liner, and has not had significant contact with groundwater, it is assumed that there has been no release of tritium to the environment.

4.3.5 Tritium Produced in the Sand Fill

While the sand was saturated with water from some time before June 1983 until the sand was completely removed in August 1988, the water was managed so that none flowed back to the local groundwater. Most of this water was evaporated at the RMDF, while some (after purification by ion

exchange) was discharged to surface water, with an average tritium concentration below the limit of detection of the EPA drinking water analytical procedure (500 pCi/L) and far below the regulatory limit for release to the environment (3,000,000 pCi/L) in 1989.

4.4 RELEASE OF TRITIUM FROM T028

The amount of tritium produced in parts of the T028 STR/STIR facility with potential for release to the environment was negligible. The diffusion of tritium from the graphite thermal column is so slow that no significant release would have occurred. The graphite, the high–density concrete, and a major amount of the ordinary concrete were disposed of as radioactive waste.

4.4.1 Tritium Produced Within the Reactor

The upgrade from STR (50 kW_t) to STIR (1 MW_t) included installation of a recirculating–water cooling tower. Blowdown of this cooling–tower could have released minor amounts of tritium to surface water. While STIR differs greatly from EBR–II, the same release model has been used to provide a rough estimate of the release from this reactor as was used for the other reactors. The surface water release is estimated to be 0.0002 Ci, at a maximum concentration of 2 pCi/L.

4.4.2 Tritium Produced in the Concrete Shield

Diffusion of tritium in the ordinary concrete could have reached the groundwater. The calculated release rate is shown in Figure 4–4.

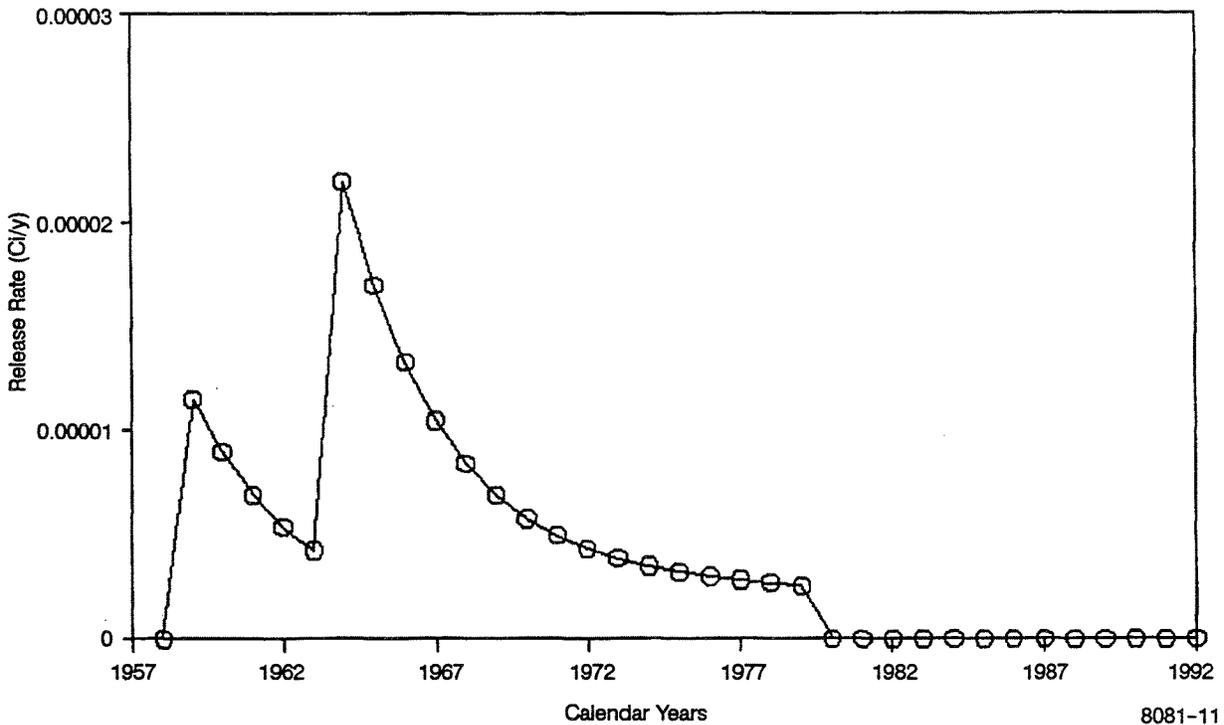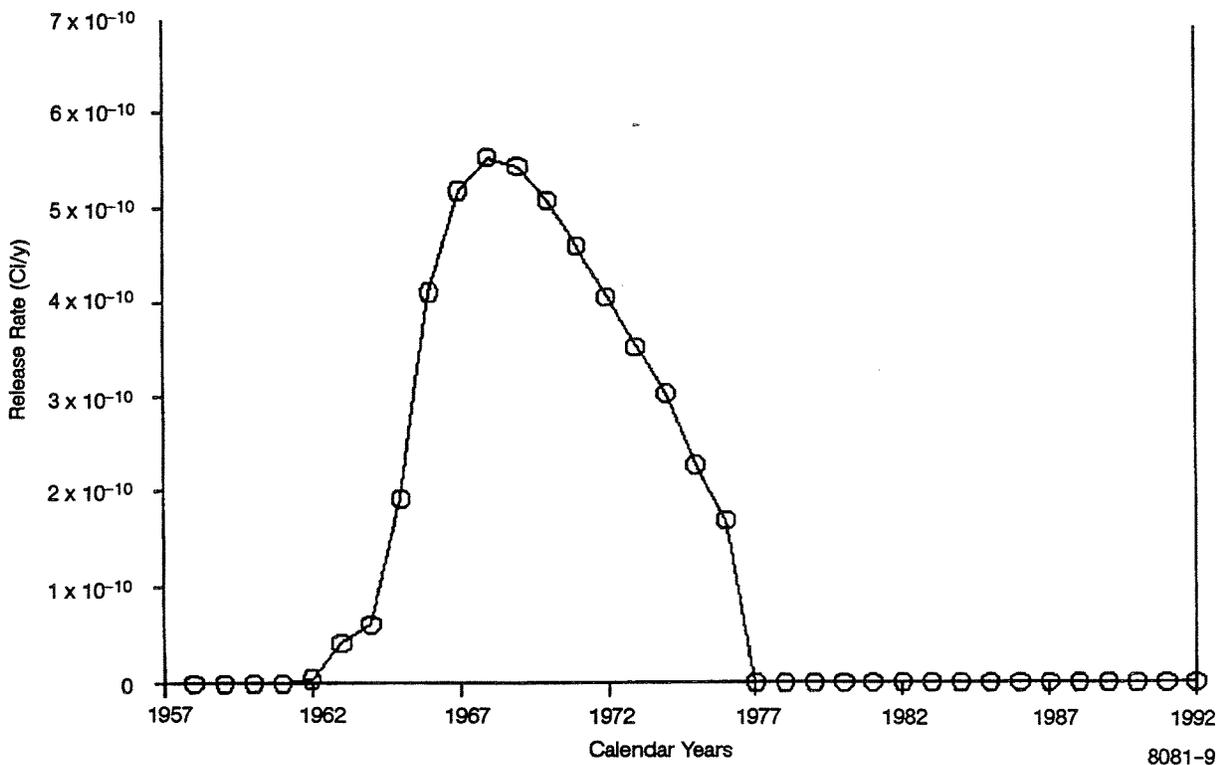Figure 4–4. Estimated Tritium Release Rate from T028 to Groundwater

4.5 RELEASE OF TRITIUM FROM T024

4.5.1 Tritium Produced in the Fuel

By comparison with the release calculated for S8DR in T059, approximately 0.0004 Ci was released from S2DR in Vault 1 and 0.0005 Ci was released from S10FS3 in Vault 2. None of this was to water, but would have been released as HT or HTO to the atmosphere or been trapped in the vacuum–pump oil.

4.5.2 Tritium Produced in the Coolant

This was negligible.

4.5.3 Tritium Produced in the Lithium Hydride Shield

Considering the relatively low temperature of the shield and the oxidized surface of the stainless steel of the shield container, release of this tritium is estimated to be less than 1 Ci. As in the test of a shield with S8DR in T059, this tritium would have been released to the vacuum system and become bound with the hydrocarbon lubricant in the pumps. Some small amount would have been discharged to the atmosphere. This release is estimated at 0.01%, based on the observed partition of radon gas between water and air. Chemical bonding of tritium atoms to hydrocarbon molecules would be expected to further reduce the estimated release.

4.5.4 Tritium Produced in the Concrete Biological Shield

The shield walls at T024 were approximately 8 ft thick, but only two walls for each cell were exposed to the outside ground, the other two being interior walls. The combined tritium release rate, from the concrete and the adjacent soil, is shown in Figure 4–5.

4.6 INADVERTENT DISPOSAL OF RADIOACTIVE MATERIALS

While there was no known deliberate disposal of radioactive waste to the environment at SSFL, at least two inadvertent releases with a potential for containing tritium did occur. These releases were identified by the much later discovery of mixed fission product activity in the vicinity of the sanitary leach field at RMDF in one case, and in the cleaning pit and basin at T886, the Sodium Disposal Facility, in the other.

4.6.1 RMDF Leach Field

The RMDF leach field became contaminated as a result of releasing about 5,000 gallons of water from the RMDF radioactive water system in 1962. It is assumed that the major source of radioactive water at that time was cleanup of the T020 hot cells after examination of fuel assemblies from the SRE Core I, or from wash water used at the SRE. It is difficult to judge if this water had any significant amount of tritium. While the source of the radioactivity is reasonably assumed to be sodium on fuel elements and equipment from the SRE following the 1959 fuel cladding damage, which is estimated to have contained approximately 0.087 μCi tritium/g sodium, it is uncertain how much
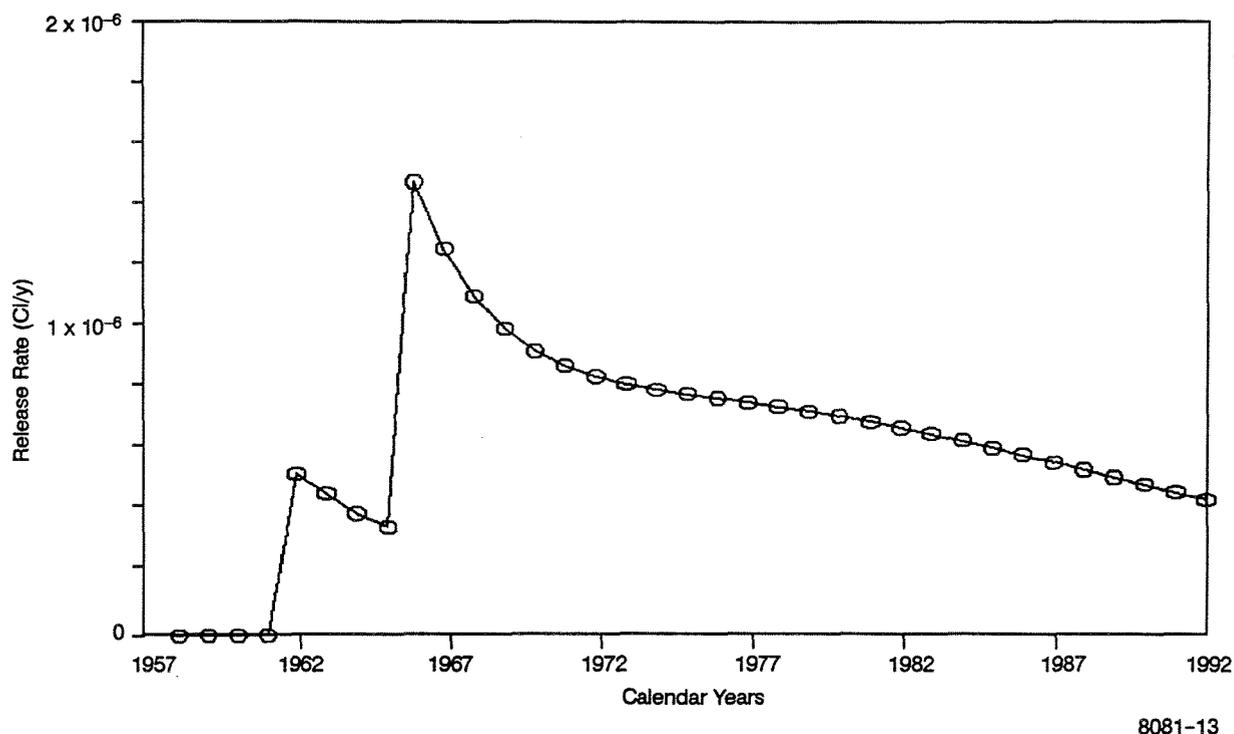
Figure 4–5. Estimated Tritium Release Rate from T024 to Groundwater

of this tritium would have been retained during the various processes of sodium passivation, dissolution, drying, and cell washdown. It seems likely that most of it would have been lost in the form of solidified radioactive waste and evaporation to the atmosphere. Therefore, the relative activity has been reduced by a somewhat arbitrary factor of 10.

Based on a reasonable estimate of the Cs–137 activity released to the leach field (0.1 Ci), the tritium (reduced by the factor of 10 to account for losses) is estimated to be 0.017 Ci. Since this is at the upper range of the estimated releases from the reactors, the leachfield should be considered to be a possible source of tritium in groundwater.

4.6.2 Sodium Disposal Facility

Since inadvertent contamination of the Sodium Disposal Facility (T886) with mixed fission products has been observed, the potential for this facility to be the source of a significant quantity of tritium has been investigated. The Sodium Disposal Facility appears unlikely to be a source of tritium in the groundwater since:

1. Wells at and down–gradient from the Sodium Disposal Facility have not shown tritium above the standard detection limit.

2. No irradiated concrete or soil, or lithium hydride shield materials, or similar materials that might have contained tritium (with the possible exception of primary sodium) have been found during cleanup operations at the Sodium Disposal Facility. Monitoring of wells near the Sodium Disposal Facility is a continuing part of the environmental program.

3. While it cannot be categorically ruled out that sodium (or NaK) containing tritium was inadvertantly disposed of here, the amounts would have been small.

Monitoring of wells near the Sodium Disposal Facility is a continuing part of the environmental program.

4.6.3 RIHL

In 1971, some of the primary coolant from the S8DR caught fire during preparations in the RIHL to dispose of the NaK. About 25 gallons of the NaK (liquid at room temperature) burned, releasing approximately 15 microcuries of fission product activity, primarily Cs–137, to the atmosphere. In chemical form, this release would have contained oxides, hydroxides, and carbonates of sodium and potassium. Since these compounds are hygroscopic and combine with moisture in air, it is possible that any tritium contained in this release could have been deposited on the ground soon after being released and before being fully dispersed in the atmosphere. It is estimated that about 10 microcuries of tritium could have been released during this fire. If that activity were deposited on a relatively small area of 10 m wide by 100 m downwind, and a rainfall of 1 cm followed, the tritium concentration would have been 1,000 pCi/L. This is approximately equal to the concentration in rainfall at that time.

4.6.4 SRE Water Leaks

Releases of contaminated water bearing mixed fission products at the SRE, prior to and during the decommissioning operations were reviewed and were found to pose little potential for significant releases of tritium.

Water with mixed fission product (Cs–137 with Sr–90 and Pm–147) activity was found leaking into the Wash Cell Valve Pit, in 1968, indicating a release of contamination to the ground in this area. This may have come from the adjacent wash cells, which were used for a short period of time to clean sodium from fuel elements after they were removed from the reactor. Because of the small volume of the wash cells and their limited use, only a small amount of radioactivity would have escaped to the ground. This radioactivity may or may not have included tritium from the reactor coolant, but the amount would have been too small to contribute to the elevated levels found in Well RD–34A.

During the process of cutting out the reactor vessel and internal structure, a water–filled storage pit was used for temporary storage of the radioactive metal pieces, prior to shipment to a disposal site. (The major radioactivity in this metal was Co–60, due to neutron activation of natural cobalt contained in the stainless steel.) During use, the storage pit developed a leak and approximately 4,000 gal of water was lost to the adjacent ground. Since the radioactivity in this water had been carried over from operations in the reactor vessel, the potential for transfer of any tritium remaining in the primary sodium coolant was investigated. This potential is very small for the following reasons:

1. Nearly all sodium had been transferred to the Primary Fill Tank for subsequent loading into drums for shipment to Hanford.

2. The residual sodium was reacted with alcohol and water that was then pumped out, solidified, and shipped to a disposal site.

3. The reactor vessel was filled with a large volume of water for use as radiation shielding during the cutting operation. Any remaining tritium would have been in a very dilute concentration.

4. As cut metal pieces were removed from the reactor vessel, they were rinsed with a water spray to minimize the transfer of radioactivity to the storage pit water.

5. The storage pit was filled with a large volume of water, further diluting any tritium.

6. The volume of water lost from the storage pit was not large.

Therefore, it is concluded that this release did not introduce any significant amount of tritium to the groundwater.

4.7 SUMMARY OF TRITIUM RELEASES

Maximum releases estimated for all reactor facilities are listed in Table 4–1. This table shows the sum of releases as they occurred, and therefore does not account for radioactive decay to the present time.

Table 4–1. Maximum Cumulative Releases to Environmental Water from SSFL Reactors (curies)

	T143	T010	T059	T028	T024
Surface water	0.0048	nil	nil	0.0002	nil
Groundwater	0.0002	0.026	0.003	nil	0.00002
Total	0.0050	0.026	0.003	0.0002	0.00002

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4.7.1 Groundwater Releases

The groundwater release rates for each facility, shown in the preceding plots, have been combined, using a logarithmic scale to permit better comparison of these estimates. The combined plots are shown in Figure 4–6. (On this scale, each major division, represents a decrease by a factor of 1,000 from the next higher division.) Here, it is clear that the release from T010 is most significant, followed closely by T059, but SRE and T024 are nearly a factor of a thousand lower. Releases from T028 are trivially small.

Many groundwater–monitoring wells have been drilled in Area IV and the water sampled and analyzed. These have been tested various times, as described in detail in Sections 6 through 9 of this report. The standard analytical method for tritium in drinking water (EPA Method 906.0) has reliably shown detectable tritium in water only from Wells RD–34A and B, RD–28, from the french drain at T059, and in cooling water confined to Building T024. These results are discussed in detail in Sections 6 through 9.

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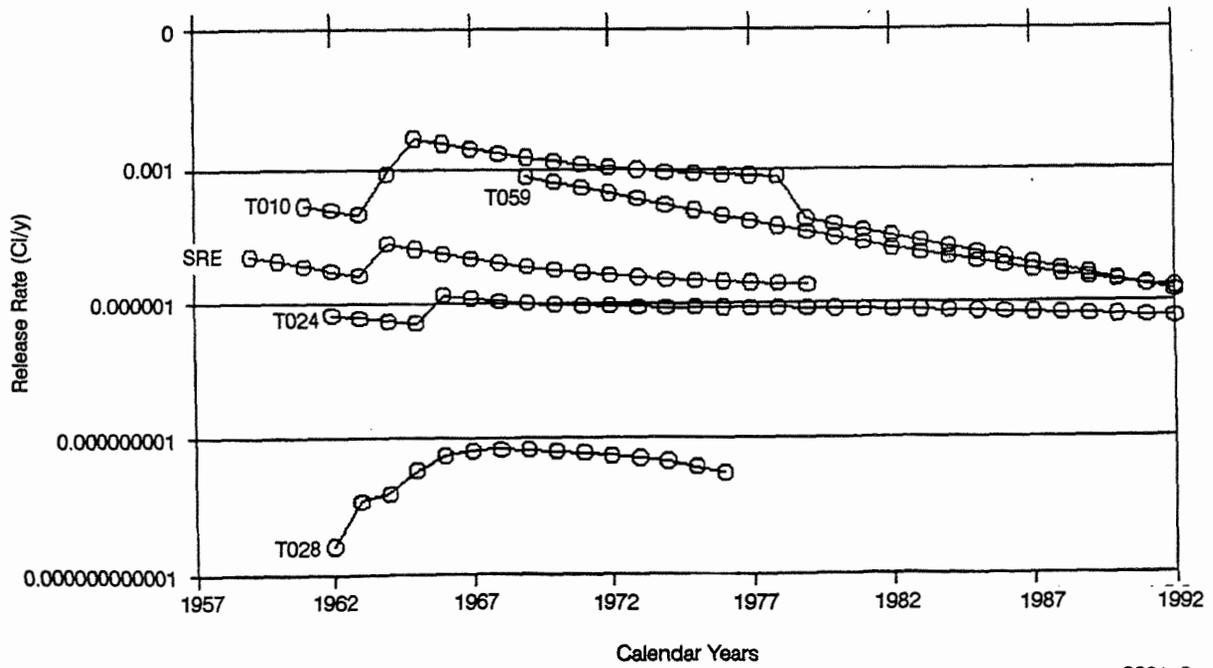
Figure 4–6. Tritium Release Rates to Groundwater Estimated
for Reactor Facilities

5.0 MODELING OF TRITIUM DISPERSION IN GROUNDWATER

As tritium is released to the groundwater, since it is bound in a molecule (HTO) that is essentially identical to the ordinary water molecules (H_2O), the tritium will go with the groundwater. That is, if the groundwater is flowing, the tritium that is released from a facility will follow the flow and will migrate from the neighborhood of the facility. If the groundwater is stationary, unmoving, the tritium will accumulate in the water and spread only slowly by diffusion. The more rapid the groundwater flow, the farther the tritium will move, but the lower the concentration, because of the influx of fresh water. Slow flow will result in higher concentrations but the migration of tritium from the facility will be less. As the groundwater flows, water with higher concentrations of tritium mixes and blends with lower concentrations, and lower concentrations fade into fresh water. This results in a spreading of the tritium, with a consequent dilution of the concentration. This is dispersion, and computer programs have been developed to model this effect.

While some ground water situations may be uniform and well defined, and permit detailed predictions to be made by using computer models, the situation of SSFL is one of fractured, irregular strata with significant variations in flow characteristics. In such a case, the calculations cannot provide a credible representation of the general distribution of the tritium concentrations. For this reason, estimates of the concentration and distribution of tritium in the groundwater must rely on sample analysis data.

6.0 SAMPLING AND ANALYSIS

To seek further information on the concentration of tritium in the groundwater at T059 where it was first found, and on its occurrence in the environment, the Rocketdyne Radiation and Nuclear Safety Group (R&NS) (now Radiation Protection and Health Physics Services) took 17 samples in the summer and fall of 1989 of different sources of water for analysis for tritium. Subsequently, many additional samples, from groundwater, ponds, surface runoff, and facility water, have been collected and analyzed for tritium, using several methods. Since environmental levels of tritium in water range from essentially zero to about a hundred pCi/L at present, and concentrations of tritium at SSFL could be expected to be in the range of a few hundred to a few thousand pCi/L, it is important to carefully review analytical results for reporting errors or analytical artifacts. This is done by comparing results for previous and similar samples and confirming that both the results and assigned uncertainties are consistent. Suspect values are identified or rejected, and repeat analyses, when practical, are requested to redetermine the concentration.

To avoid the measurement problems associated with the expected low levels of tritium in this survey, a laboratory was chosen that was capable of performing an electrolytic enrichment of the tritium prior to use of the normal liquid scintillation counting for determination of the tritium in the electrolyzed water. The sensitivity quoted for this procedure is 10 pCi/L. To provide further information and investigate questions raised by the initial survey, additional samples were taken in winter, spring, and summer of 1990. These were also analyzed by use of the electrolytic enriched process. Many more samples have been taken since then, most of them analyzed using the standard EPA Method 906.0.

The electrolytic enrichment technique used is taken from an EPA/LV procedure (EMSL–LV–0539–17, "Radiochemical Analytical Procedures for Analysis of Environmental Samples," edited by F. Johns, P. B. Hahn, D. J. Thome, and E. W. Bretthaver, Environmental Monitoring and Support Laboratory, Las Vegas, Nevada, March 1979, Procedure entitled "Determination of Low–Level Tritium in Water [Alkaline Electrolytic Enrichment]," pp 81–85). After the enrichment process, the water is analyzed for tritium by use of the standard liquid scintillation counting procedure generally used.

This procedure recommends against its use for water with tritium concentrations above 250 pCi/L because of cross–contamination of equipment. Marshall Lane at UST (ITAS) has studied this limitation and finds that washing and thoroughly drying the equipment eliminates this problem.

Concurrently with this special R&NS survey for tritium in water, water samples taken routinely from various sources by Groundwater Resources Consultants, Inc. (GRC) and the Rocketdyne Environmental Protection Unit were sent for tritium analyses, to BC Laboratories, Combustion Engineering, TMA/Norcal, ITAS, and CEP. Groundwater Resources is under contract to Rocketdyne's Environmental Protection Unit. In some cases, these laboratories then sent the samples to UST for tritium analysis. (These labs used only the standard liquid scintillation method, without the

preprocessing by electrolytic enrichment, and sensitivity quoted is 500–1000 pCi/L.) The locations of the groundwater wells sampled for these surveys are shown in Figure 6-1.

Recently (March 1992), many soil and surface water samples were collected and analyzed for tritium as part of the McLaren/Hart Survey of the Brandeis–Bardin Institute and Santa Monica Mountains conservancy properties adjoining SSFL. These results are not reported here since not all analysis results have been received. Results will be presented in the McLaren/Hart final report, expected in late 1992.
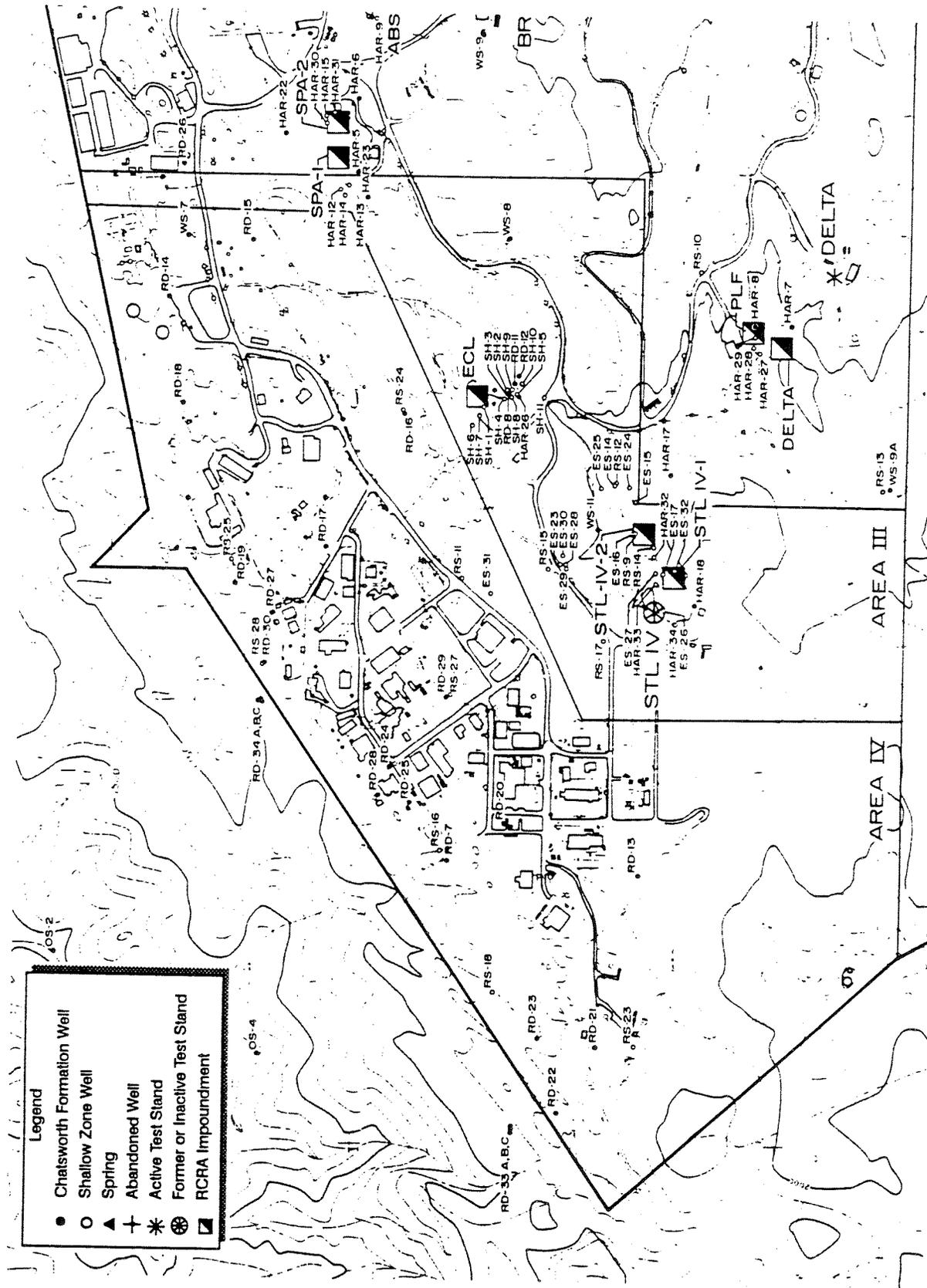
Figure 6–1. Location of Wells at SSFL Sampled for Tritium in Water

7.0 RESULTS AND INTERPRETATION

Table 7-1 shows a complete summary of the results of these measurements. The results have been sorted according to the sample location. The highest tritium concentrations reported are for the offsite well RD-34A. In this table, many results are shown as negative concentrations. This is a consequence of the very low levels of tritium being analyzed. Because of the random nature of counting radioactivity and particularly when the activity is close to the analytical background, occasionally the observed countrate will be less than the average background countrate. In those cases, subtraction of the background will produce a negative result for the analysis.

The results reported as values that are suitable for statistical review (excluding values that are reported as less than the detection limit) are displayed in Figure 7-1, as a cumulative probability plot. (A Gaussian or "normal" distribution of values would show as a diagonal straight line on this plot.) The concentration value is plotted by the small circle, with the (+ /-) 2-sigma uncertainties shown by the error bars extending above and below each point. This plot shows a merging of the set of UST/EE results (those with very small or no error bars) and those of other analyses (those with moderate to large error bars). The distribution clearly deviates from a Gaussian distribution at about 83% cumulative probability, and at about 96%. The behavior of this statistical distribution provides an identification of natural tritium concentrations and artificial tritium concentrations.
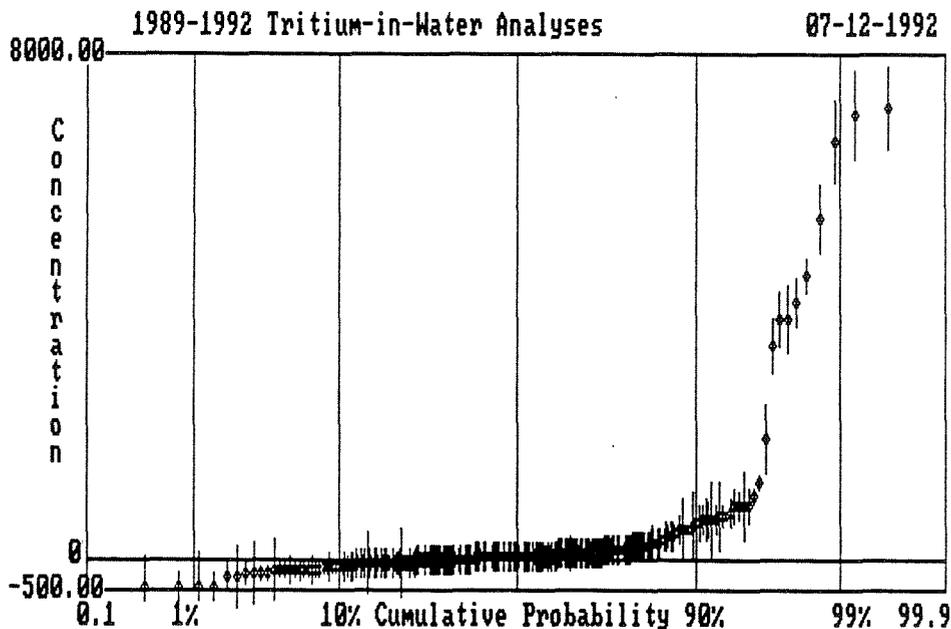
Figure 7-1. Cumulative Probability Plot of Results of Tritium-in-Water Analyses

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Table 7-1. Summary of Tritium-in-Water Analyses
(Sheet 1 of 9)

September 12, 1992

LOCATION	DATE	TRITIUM pCi/L	+/-2 SIGMA	SAMPLE ID	SAMPLER	LABORATORY
17TH&G STS	9/17/89	42.90	9.07	T917C	R&NS	UST/EE
ARROWHEAD	9/19/89	26.10	7.69	T919C	R&NS	UST/EE
BELL CREEK	9/18/89	30.00	8.12	T918B	R&NS	UST/EE
CALLEGUAS	12/14/90	117.00	230.00		GRC	ITAS
CALLEGUAS	3/10/91	48.10	134.00		GRC	ITAS
CALLEGUAS	3/12/92	-179.00	500.00		GRC	CEP
CANADAGLAC	3/15/90	15.40	8.31	T0315	R&NS	ITAS/EE
CHATS SWPL	10/02/89	36.70	9.50	T1002A	R&NS	UST/EE
DEAD WATER	9/26/89	0.53	3.03	I21801B	UST	UST/EE
DEAD WATER	9/26/89	4.34	3.27	I21665X	UST	UST/EE
DEAD WATER	9/26/89	3.33	3.13	I21800B	UST	UST/EE
DEAD WATER	9/26/89	1.99	3.15	I21664X	UST	UST/EE
DEAD WATER	9/ /90	1.50	7.78		ITAS	ITAS/EE
DEAD WATER	9/ /90	8.86	7.97		ITAS	ITAS/EE
DISTRICT17	9/11/89	-154.00	116.00	7169-5	GRC	BCL/UST
ES-23B	3/04/91	154.00	197.00		GRC	ITAS
ES-24	9/10/89	<1000	not stated		GRC	TMA/Norcal
ES-24	9/10/89	-58.00	126.00		GRC	UST
ES-24	9/10/89	-62.70	124.00		GRC	UST
ES-24	9/10/89	<1000	not stated	8294-9	GRC	TMA/Norcal
ES-24	9/10/89	<1000	not stated	8294-8	GRC	TMA/Norcal
ES-31	12/10/90	49.90	196.00		GRC	ITAS
ES-31	3/04/91	159.00	197.00		GRC	ITAS
ES-31	3/04/91	590.00	221.00		GRC	ITAS
ES-31	6/03/91	7.70	194.00		GRC	ITAS
ES-31	3/05/92	-730.00	494.00		GRC	CEP
ES-31	9/07/91	-48.10	196.00		GRC	ITAS
ES-31	12/07/91	-89.60	206.00		GRC	CEP
ETEC PRPK	9/17/89	28.80	8.51	T917F	R&NS	UST/EE
GOLDNWILDR	6/01/90	-3.07	7.32	T0601	R&NS	ITAS/EE
HAR- 3	9/11/89	<1000	not stated	8294-28	GRC	TMA/Norcal
HAR- 3	9/11/89	-4.78	121.00	7169-2	GRC	BCL/UST
HAR- 4	9/11/89	<1000	not stated	8294-30	GRC	TMA/Norcal
HAR- 4	9/11/89	<1000	not stated	8294-29	GRC	TMA/Norcal
HAR- 4	9/11/89	-185.00	115.00	7169-3	GRC	BCL/UST
HAR- 6	9/14/89	45.90	133.00	7276-17	GRC	BCL/UST
HAR- 6	9/14/89	<1000	not stated	8297-18	GRC	TMA/Norcal
HAR- 7	9/09/89	-88.90	128.00		GRC	BCL/UST
HAR- 7	no date	<1000	not stated	8294-19	GRC	TMA/Norcal
HAR-14	9/12/89	-22.90	124.00	7235-5	GRC	BCL/UST
HAR-14 FLT	9/12/89	<1000	not stated	8296-13	GRC	TMA/Norcal
HAR-14UNFT	9/12/89	<1000	not stated	8296-12	GRC	TMA/Norcal
HAR-16	9/09/89	-57.40	126.00		GRC	BCL/UST
HAR-16	9/09/89	<1000	not stated	8294-20	GRC	TMA/Norcal
HAR-16	6/28/90	58.30	11.30	HAR-16	GRC	ITAS/EE
HAR-18	9/11/89	<1000	not stated	8297-6	GRC	TMA/Norcal
HAR-18	9/11/89	-68.40	133.00	7276-6	GRC	BCL/UST
HAR-19	9/09/89	<1000	not stated	8294-21	GRC	TMA/Norcal
HAR-19	9/09/89	329.00	137.00		GRC	BCL/UST
HAR-19	6/28/90	12.90	212.00		GRC	ITAS
HAR-20	9/09/89	-65.00	125.00		GRC	BCL/UST
HAR-20	9/09/89	<1000	not stated	8294-22	GRC	TMA/Norcal

Table 7–1. Summary of Tritium–in–Water Analyses
(Sheet 2 of 9)

HAR-21	9/09/89	<1000	not stated	8294-23	GRC	TMA/Norcal
HAR-21	9/09/89	-39.20	121.00		GRC	BCL/UST
HAR-30	9/12/89	-45.00	129.00	7276-7	GRC	BCL/UST
HAR-30	9/12/89	<1000	not stated	8297-7	GRC	TMA/Norcal
LADWP TAP	6/18/90	29.70	10.10	T0618A	R&NS	ITAS/EE
LADWP TAP	9/19/89	35.80	8.54	T919F	R&NS	UST/EE
MOORPARK	6/20/90	16.60	8.52	T0620A	R&NS	ITAS/EE
OS- 1	9/13/89	<1000	not stated	8297-8	GRC	TMA/Norcal
OS- 1	9/13/89	-227.00	121.00	7276-8	GRC	BCL/UST
OS- 1	12/11/90	-17.50	207.00		GRC	ITAS
OS- 1	3/09/91	-109.00	185.00		GRC	ITAS
OS- 1	9/09/91	-2.38	210.00		DHS	DHS
OS- 1	9/09/91	63.80	201.00		GRC	ITAS
OS- 1	12/09/91	-49.00	209.00		GRC	ITAS
OS- 1	6/09/92	-129.00	489.00		GRC	CEP
OS- 2	9/13/89	<1000	not stated	8297-9	GRC	TMA/Norcal
OS- 2	9/13/89	-90.80	128.00	7276-9	GRC	BCL/UST
OS- 2	12/11/90	-39.70	206.00		GRC	ITAS
OS- 2	3/08/91	-80.40	186.00		GRC	ITAS
OS- 2	3/08/91	86.50	186.00		GRC	ITAS
OS- 2	9/09/91	-105.00	208.00		DHS	DHS
OS- 2	6/09/92	348.00	493.00		GRC	CEP
OS- 3	9/13/89	<1000	not stated	8297-10	GRC	TMA/Norcal
OS- 3	9/13/89	7.49	132.00	7276-10	GRC	BCL/UST
OS- 3	12/11/90	-35.10	207.00		GRC	ITAS
OS- 3	3/08/91	44.40	192.00		GRC	ITAS
OS- 3	12/09/91	-9.42	211.00		GRC	ITAS
OS- 3	6/09/92	-223.00	485.00		GRC	CEP
OS- 4	9/13/89	71.20	135.00	7276-11	GRC	BCL/UST
OS- 4	9/13/89	<1000	not stated	8297-12	GRC	TMA/Norcal
OS- 4	12/11/90	-26.80	208.00		GRC	ITAS
OS- 4	6/09/92	169.00	488.00		GRC	CEP
OS- 5	9/13/89	-52.40	129.00	7276-12	GRC	BCL/UST
OS- 5	9/13/89	<1000	not stated	8297-13	GRC	TMA/Norcal
OS- 5	12/11/90	-80.30	205.00		GRC	ITAS
OS- 5	3/08/91	-162.00	182.00		GRC	ITAS
OS- 5	9/09/91	-102.00	208.00		DHS	DHS
OS- 5	9/09/91	-102.00	208.00		DHS	DHS
OS- 5	9/09/91	129.00	204.00		GRC	ITAS
OS- 5	12/09/91	61.90	214.00		GRC	ITAS
OS- 5	6/09/92	91.00	492.00		GRC	CEP
OS- 8	9/13/89	<1000	not stated	8297-14	GRC	TMA/Norcal
OS- 8	9/13/89	101.00	140.00	7276-13	GRC	BCL/UST
OS- 8	6/09/92	-172.00	490.00		GRC	CEP
OS-10	9/13/89	<1000	not stated	8297-15	GRC	TMA/Norcal
OS-10	9/13/89	-121.00	126.00	7276-14	GRC	BCL/UST
OS-10	12/09/91	-120.00	205.00		GRC	ITAS
OS-15	12/10/91	127.00	224.00		GRC	ITAS
OS-16	9/14/89	<1000	not stated	8297-16	GRC	TMA/Norcal
OS-16	9/14/89	-100.00	127.00	7276-15	GRC	BCL/UST
OS-16	9/09/91	-131.00	208.00		DHS	DHS
OS-16	9/09/91	-93.30	193.00		GRC	ITAS
OS-16	12/10/91	148.00	226.00		GRC	ITAS
OS-16	3/12/92	226.00	400.00		GRC	CEP
OS-17	9/13/89	<1000	not stated	8297-17	GRC	TMA/Norcal
OS-17	9/13/89	37.50	132.00	7276-16	GRC	BCL/UST
OS-17	9/12/91	306.00	230.00		GRC	ITAS

Table 7-1.   Summary of Tritium-in-Water Analyses
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OS-17	12/10/91	31.70	219.00		GRC	ITAS
OS-17	3/12/92	-193.00	500.00		GRC	CEP
OS-21	9/09/89	<1000	not stated	8294-18	GRC	TMA/Norcal
OS-21	9/09/89	-160.00	121.00		GRC	BCL/UST
OS-21	3/09/91	-38.80	188.00		GRC	ITAS
OS-21	12/10/91	-165.00	209.00		GRC	ITAS
OS-21	3/12/92	209.00	500.00		GRC	CEP
PERRIER	5/10/90	96.30	14.20	T0510	R&NS	ITAS/EE
RAINFALL	9/17/89	15.80	7.14	T917A	R&NS	UST/EE
RD- 1	9/11/89	123.00	137.00	7276-1	GRC	BCL/UST
RD- 1	9/11/89	<1000	not stated		GRC	TMA/Norcal
RD- 3	9/12/89	-129.00	117.00	7235-1	GRC	BCL/UST
RD- 3	9/10/89	-155.00	122.00		GRC	UST
RD- 3 FLT	no date	<1000	not stated	8296-2	GRC	TMA/Norcal
RD- 3 FLT	9/11/89	<1000	not stated		GRC	TMA/Norcal
RD- 3UNFLT	9/10/89	<1000	not stated	8294-10	GRC	TMA/Norcal
RD- 3UNFLT	no date	<1000	not stated	8296-1	GRC	TMA/Norcal
RD- 5	9/10/89	-10.30	128.00		GRC	UST
RD- 5	9/10/89	<1000	not stated	8294-11	GRC	TMA/Norcal
RD- 5	9/10/91	144.00	202.00		GRC	ITAS
RD- 5	9/09/91	-114.00	208.00		DHS	DHS
RD- 6	9/10/89	<1000	not stated	8294-12	GRC	TMA/Norcal
RD- 6	9/10/89	-44.00	126.00		GRC	UST
RD- 6	3/06/91	83.10	193.00		GRC	ITAS
RD- 6	9/09/91	-183.00	206.00		DHS	DHS
RD- 6	9/10/91	58.60	197.00		GRC	ITAS
RD- 6	3/10/92	477.00	500.00		GRC	CEP
RD- 7	9/11/89	-101.00	128.00	7276-2	GRC	BCL/UST
RD- 7	9/11/89	<1000	not stated	8297-2	GRC	TMA/Norcal
RD- 7	6/27/90	10.00	7.96	RD-7	GRC	ITAS/EE
RD- 7	12/05/90	-8.63	201.00		GRC	ITAS
RD- 7	3/09/91	32.30	192.00		GRC	ITAS
RD- 7	3/06/92	-432.00	500.00		GRC	CEP
RD- 8	9/11/89	-136.00	126.00	7276-3	GRC	BCL/UST
RD- 8	9/11/89	<1000	not stated	8297-3	GRC	TMA/Norcal
RD-10	9/10/89	<1000	not stated	8294-13	GRC	TMA/Norcal
RD-10	9/10/89	-72.10	125.00		GRC	UST
RD-10	3/06/91	21.20	190.00		GRC	ITAS
RD-10	3/07/92	-14.00	500.00		GRC	CEP
RD-13 FLT	no date	<1000	not stated	8296-4	GRC	TMA/Norcal
RD-13UNFLT	9/10/89	<1000	not stated	8296-3	GRC	TMA/Norcal
RD-13UNFLT	9/12/89	<1000	not stated		GRC	TMA/Norcal
RD-13	9/12/89	-167.00	115.00	7235-2	GRC	BCL/UST
RD-13	10/17/89	-88.10	229.00		GRC	BCL/UST
RD-13	12/06/90	-28.80	197.00		GRC	ITAS
RD-13	3/08/91	-33.30	189.00		GRC	ITAS
RD-13	3/12/92	68.00	500.00		GRC	CEP
RD-14	10/18/89	-157.00	226.00		GRC	BCL/UST
RD-14	10/18/89	161.00	202.00		GRC	BCL/UST
RD-14	12/07/90	2.77	195.00		GRC	ITAS
RD-14	3/09/91	26.80	191.00		GRC	ITAS
RD-14	12/06/91	-90.60	206.00		GRC	ITAS
RD-14	3/05/92	-480.00	498.00		GRC	CEP
RD-15	10/19/89	-12.20	192.00		GRC	BCL/UST
RD-15	12/07/90	4.99	198.00		GRC	ITAS
RD-15	3/10/91	85.50	186.00		GRC	ITAS
RD-15	3/11/92	92.00	500.00		GRC	CEP

Table 7-1. Summary of Tritium-in-Water Analyses
(Sheet 4 of 9)

RD-15		3/11/92	<100	not stated		GRC	Teledyne
RD-16		10/25/89	176.00	222.00		GRC	BCL/UST
RD-16		12/07/90	56.30	198.00		GRC	ITAS
RD-16		3/09/91	98.10	187.00		GRC	ITAS
RD-16		12/05/91	67.40	219.00		GRC	ITAS
RD-16		6/06/92	564.00	529.00	REANALYSIS	GRC	CEP
RD-17		10/18/89	77.80	243.00		GRC	BCL/UST
RD-17		10/18/89	14.10	194.00		GRC	BCL/UST
RD-17		12/04/90	108.00	199.00		GRC	ITAS
RD-17		3/05/91	1.85	189.00		GRC	ITAS
RD-17		12/07/91	-44.40	209.00		GRC	CEP
RD-17		3/04/92	-98.00	498.00		GRC	CEP
RD-18		10/26/89	53.60	215.00		GRC	BCL/UST
RD-18		7/01/90	11.30	8.08	RD-18	GRC	ITAS/EE
RD-18		12/08/90	26.80	195.00		GRC	ITAS
RD-18		3/09/91	201.00	192.00		GRC	ITAS
RD-18		3/12/92	314.00	414.00		GRC	CEP
RD-19		10/26/89	27.30	214.00		GRC	BCL/UST
RD-19		12/08/90	-20.30	193.00		GRC	ITAS
RD-19		3/08/91	225.00	193.00		GRC	ITAS
RD-19		3/08/91	11.50	182.00		GRC	ITAS
RD-19		12/11/91	-22.10	217.00		GRC	ITAS
RD-19		3/12/92	330.00	500.00		GRC	CEP
RD-20		10/17/89	-72.10	230.00		GRC	BCL/UST
RD-20		12/07/90	49.90	197.00		GRC	ITAS
RD-20		12/10/90	-26.80	192.00		GRC	ITAS
RD-20		3/05/91	132.00	196.00		GRC	ITAS
RD-20		12/10/91	20.20	219.00		GRC	ITAS
RD-20		3/04/92	-274.00	486.00		GRC	CEP
RD-21		10/20/89	-100.00	229.00		GRC	BCL/UST
RD-21		10/20/89	35.70	194.00		GRC	BCL/UST
RD-21		6/21/90	5.84	7.72	RD-21	GRC	ITAS/EE
RD-21		12/03/90	182.00	202.00		GRC	ITAS
RD-21		3/08/91	119.00	188.00		GRC	ITAS
RD-21		12/05/91	184.00	225.00		GRC	ITAS
RD-21		3/04/92	-256.00	497.00		GRC	CEP
RD-22		10/19/89	-47.90	189.00		GRC	BCL/UST
RD-22		7/01/90	-0.58	7.37	RD-22	GRC	ITAS/EE
RD-22		12/04/90	116.00	198.00		GRC	ITAS
RD-22		12/04/90	41.30	195.00		GRC	ITAS
RD-22		3/11/91	-90.50	186.00		GRC	ITAS
RD-22		12/06/91	-26.80	210.00		GRC	ITAS
RD-22		6/05/92	75.00	517.00	REANALYSIS	GRC	CEP
RD-23		10/20/89	589.00	267.00		GRC	BCL/UST
RD-23		3/11/91	64.70	193.00		GRC	ITAS
RD-23		6/29/90	243.00	26.20	RD-23	GRC	ITAS/EE
RD-23	FLT	6/29/90	129.00	218.00		GRC	BCL/UST
RD-23		12/05/90	88.30	206.00		GRC	ITAS
RD-23		3/11/91	106.00	195.00		GRC	ITAS
RD-23		12/05/91	256.00	229.00		GRC	ITAS
RD-23		3/04/92	-66.00	517.00		GRC	CEP
RD-24		9/12/89	-22.00	122.00	7235-3	GRC	BCL/UST
RD-24	FLT	9/12/89	<1000	not stated	8296-6	GRC	TMA/Norcal
RD-24	UNFLT	9/12/89	<1000	not stated	8296-5	GRC	TMA/Norcal
RD-24		9/20/89	147.00	17.20	T920A	R&NS	UST/EE
RD-24		10/17/89	-89.00	229.00		GRC	BCL/UST
RD-24		6/30/90	91.60	13.80	RD-24	GRC	ITAS/EE

Table 7-1. Summary of Tritium-in-Water Analyses
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RD-24	12/05/90	37.40	204.00		GRC	ITAS
RD-24	3/06/91	158.00	197.00		GRC	ITAS
RD-24	12/11/91	-33.70	216.00		GRC	ITAS
RD-24	3/06/92	122.00	500.00		GRC	CEP
RD-25	7/01/90	8.73	7.88	RD-25	GRC	ITAS/EE
RD-25	9/12/89	-162.00	116.00	7235-4	GRC	BCL/UST
RD-25 FLT	9/12/89	<1000	not stated	8296-11	GRC	TMA/Norcal
RD-25 FLT	9/12/89	<1000	not stated	8296-8	GRC	TMA/Norcal
RD-25UNFLT	9/12/89	<1000	not stated	8296-10	GRC	TMA/Norcal
RD-25UNFLT	9/12/89	<1000	not stated	8296-7	GRC	TMA/Norcal
RD-25	9/19/89	3.24	6.93	T919E	R&NS	UST/EE
RD-25	10/20/89	-99.30	229.00		GRC	BCL/UST
RD-25	12/05/90	17.30	207.00		GRC	ITAS
RD-25	3/06/91	-45.30	187.00		GRC	ITAS
RD-25	12/10/91	93.30	222.00		GRC	ITAS
RD-25	3/06/92	482.00	500.00		GRC	CEP
RD-26	10/20/89	45.90	237.00		GRC	BCL/UST
RD-26	12/04/90	209.00	204.00		GRC	ITAS
RD-26	3/07/91	110.00	187.00		GRC	ITAS
RD-26	3/11/92	459.00	416.00		GRC	CEP
RD-27	12/04/90	90.20	197.00		GRC	ITAS
RD-27	3/07/91	27.90	183.00		GRC	ITAS
RD-27	10/19/89	2.82	193.00		GRC	BCL/UST
RD-27	3/09/92	-199.00	500.00		GRC	CEP
RD-28 FLT	9/13/89	<1000	not stated	8296-15	GRC	TMA/Norcal
RD-28UNFLT	9/13/89	<1000	not stated	8296-14	GRC	TMA/Norcal
RD-28	9/13/89	665.00	149.00		GRC	BCL/UST
RD-28	9/26/89	669.00	61.30	T1004A	R&NS	UST/EE
RD-28	10/19/89	699.00	234.00		GRC	BCL/UST
RD-28	3/27/90	819.00	236.00		GRC	ITAS
RD-28	7/01/90	612.00	244.00	RD-28	GRC	ITAS
RD-28	9/16/90	814.00	242.00		GRC	ITAS
RD-28	9/16/90	839.00	242.00		GRC	ITAS
RD-28	12/05/90	567.00	232.00		GRC	ITAS
RD-28	3/06/91	638.00	223.00		GRC	ITAS
RD-28	6/10/91	431.00	227.00		GRC	ITAS
RD-28	9/11/91	620.00	247.00		GRC	ITAS
RD-28	12/10/91	575.00	500.00		GRC	CEP
RD-28	12/10/91	433.00	250.00		GRC	ITAS
RD-28	3/06/92	420.00	110.00		GRC	Teledyne
RD-28	3/06/92	441.00	500.00		GRC	CEP
RD-28	6/10/92	1070.00	541.00	REANALYSIS	GRC	CEP
RD-28	6/10/92	1025.00	505.00		GRC	CEP
RD-29	10/18/89	-101.00	230.00		GRC	BCL/UST
RD-29	12/06/90	55.70	201.00		GRC	ITAS
RD-29	3/05/91	105.00	194.00		GRC	ITAS
RD-29	12/10/91	89.50	222.00		GRC	ITAS
RD-29	12/10/91	200.00	500.00		GRC	CEP
RD-29	3/03/92	-447.00	520.00		GRC	CEP
RD-30	10/19/89	108.00	199.00		GRC	BCL/UST
RD-30	12/06/90	34.60	200.00		GRC	ITAS
RD-30	3/09/91	89.60	195.00		GRC	ITAS
RD-30	9/09/91	20.30	199.00		GRC	ITAS
RD-30	12/06/91	28.70	213.00		GRC	ITAS
RD-30	6/03/92	<200	not stated		GRC	Teledyne
RD-30	6/03/92	-76.00	518.00	REANALYSIS	GRC	CEP
RD-31	10/24/89	188.00	227.00		GRC	BCL/UST

Table 7-1. Summary of Tritium-in-Water Analyses
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RD-31	12/05/90	-56.60	198.00		GRC	ITAS
RD-31	3/10/91	182.00	191.00		GRC	ITAS
RD-31	3/05/92	-652.00	493.00		GRC	CEP
RD-33A	10/01/91	<500	not stated		GRC	CEP
RD-33A	12/05/91	97.20	221.00		GRC	ITAS
RD-33A	12/12/91	-14.40	214.00		GRC	ITAS
RD-33A	12/12/91	-450.00	227.00		GRC	DHS/S&RL
RD-33A	12/12/91	138.00	500.00		GRC	CEP
RD-33A	6/08/92	335.00	515.00	REANALYSIS	GRC	CEP
RD-33B	10/01/91	<500	not stated		GRC	CEP
RD-33B	12/12/91	51.90	218.00		GRC	ITAS
RD-33B	12/12/91	-164.00	500.00		GRC	CEP
RD-33B	12/12/91	-306.00	230.00		GRC	DHS/S&RL
RD-33B	6/24/92	-219.00	492.00		GRC	CEP
RD-33C	10/01/91	<500	not stated		GRC	CEP
RD-33C	12/05/91	68.30	219.00		GRC	ITAS
RD-33C	12/12/91	-21.10	214.00		GRC	ITAS
RD-33C	12/12/91	100.00	500.00		GRC	CEP
RD-33C	12/12/91	-432.00	227.00		GRC	DHS/S&RL
RD-33C	6/08/92	368.00	513.00	REANALYSIS	GRC	CEP
RD-33D	8/22/91	7.71	195.00	487-511	GRC	ITAS
RD-33D	8/22/91	5.78	195.00	592-678	GRC	ITAS
RD-33D	8/22/91	69.40	198.00	592-616	GRC	ITAS
RD-33D	8/23/91	<500	not stated		GRC	CEP
RD-33D	8/23/91	86.70	199.00	258-301	GRC	ITAS
RD-33D	8/23/91	98.20	200.00	403-427	GRC	ITAS
RD-33D	8/23/91	173.00	204.00	400	GRC	ITAS
RD-33D	8/23/91	23.00	196.00	340-364	GRC	ITAS
RD-34A	8/12/91	3788.00	557.00		GRC	CEP
RD-34A	8/12/91	4510.00	287.00		DHS	DHS
RD-34A	8/12/91	3380.00	406.00		GRC	ITAS
RD-34A	8/14/91	3780.00	437.00		GRC	ITAS
RD-34A	8/16/91	5410.00	547.00		GRC	ITAS
RD-34A	10/01/91	6628.00	642.00		GRC	CEP
RD-34A	12/05/91	7040.00	685.00		GRC	ITAS
RD-34A	12/05/91	7155.00	632.00		GRC	CEP
RD-34A	3/10/92	6700.00	200.00		GRC	Teledyne
RD-34A	6/08/92	2529.00	548.00	REANALYSIS	GRC	CEP
RD-34A	3/10/92	7069.00	598.00		GRC	CEP
RD-34B	8/13/91	326.00	210.00		GRC	ITAS
RD-34B	8/13/91	359.00	213.00		DHS	DHS
RD-34B	8/15/91	252.00	208.00		GRC	ITAS
RD-34B	10/01/91	637.00	570.00		GRC	CEP
RD-34B	12/05/91	336.00	234.00		GRC	ITAS
RD-34B	12/11/91	236.00	230.00		GRC	ITAS
RD-34B	12/11/91	820.00	538.00		GRC	CEP
RD-34B	3/10/92	390.00	100.00		GRC	Teledyne
RD-34B	6/08/92	534.00	520.00	REANALYSIS	GRC	CEP
RD-34B	3/10/92	336.00	500.00		GRC	CEP
RD-34C	8/14/91	108.00	197.00		GRC	ITAS
RD-34C	8/14/91	172.00	201.00		GRC	ITAS
RD-34C	8/14/91	15.20	205.30		DHS	DHS
RD-34C	8/15/91	60.40	198.00		GRC	ITAS
RD-34C	8/15/91	108.00	197.00		GRC	ITAS
RD-34C	10/01/91	<500	not stated		GRC	CEP
RD-34C	12/06/91	71.20	215.00		GRC	ITAS
RD-34C	12/12/91	247.00	500.00		GRC	CEP

Table 7-1. Summary of Tritium-in-Water Analyses
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RD-34C	12/12/91	30.80	217.00		GRC	ITAS
RD-34C	3/10/92	498.00	500.00		GRC	CEP
RD-34C	3/10/92	<100	not stated		GRC	Teledyne
RD-34C	6/08/92	455.00	519.00	REANALYSIS	GRC	CEP
RD-51B	8/15/91	119.00	201.00		GRC	ITAS
RD-51C	7/10/91	243.00	252.00		GRC	ITAS
RD-51C	12/14/91	32.70	219.00		GRC	ITAS
RD-51C	3/06/92	83.00	500.00		GRC	CEP
RMDF POND	9/17/89	17.00	7.30	T917B	R&NS	UST/EE
RMDF POND	12/11/91	<500	not stated		RP&HPS	CEP
RS- 7	9/11/89	-74.60	120.00	7169-4	GRC	BCL/UST
RS- 7	no date	<1000	not stated	8294-27	GRC	TMA/Norcal
RS-11	12/06/90	43.20	200.00		GRC	ITAS
RS-11	3/04/91	58.20	192.00		GRC	ITAS
RS-11	12/07/91	12.00	212.00		GRC	ITAS
RS-11	3/05/92	-378.00	500.00		GRC	CEP
RS-13	no date	<1000	not stated	8294-5	GRC	TMA/Norcal
RS-13	9/09/89	-148.00	121.00		GRC	UST
RS-14	9/10/89	-39.30	129.00		GRC	UST
RS-14	9/10/89	-116.00	122.00		GRC	UST
RS-14	9/10/89	<1000	not stated	8294-6	GRC	TMA/Norcal
RS-14	9/10/89	<1000	not stated	8294-7	GRC	TMA/Norcal
RS-16	3/09/92	25.00	500.00		GRC	CEP
RS-17	12/10/90	61.00	197.00		GRC	ITAS
RS-17	12/07/91	-5.54	211.00		GRC	ITAS
RS-18	3/10/91	75.70	194.00		GRC	ITAS
RS-18	3/10/91	102.00	195.00		GRC	ITAS
RS-18	3/04/92	-200.00	496.00		GRC	CEP
RS-27	3/04/92	-472.00	498.00		GRC	CEP
RS-28	10/19/89	47.00	195.00		GRC	BCL/UST
RS-28	12/06/90	-25.00	197.00		GRC	ITAS
RS-28	3/09/91	198.00	192.00		GRC	ITAS
RS-28	12/06/91	89.90	216.00		GRC	ITAS
RS-28	3/06/92	250.00	500.00		GRC	CEP
RUNOFF SRE	11/26/89	0.00	1000.00	8359-4	EU	TMA/Norcal
RUNOFF SRE	1/17/90	0.00	1000.00	8397-6	EU	TMA/Norcal
RUNOFF SRE	1/17/90	0.00	1000.00	8397-2	EU	TMA/Norcal
RUNOFF SRE	2/05/90	-52.70	173.00	G91510	EU	CE/UST
RUNOFF SRE	2/05/90	0.00	1000.00	8397-6	EU	TMA/Norcal
RUNOFF SRE	2/05/90	0.00	1000.00	8397-2	EU	TMA/Norcal
RUNOFF SRE	2/17/90	-95.10	206.00	G92190	EU	CE/UST
RUNOFF SRE	12/29/91	<500	not stated		EU	CEP
RUNOFF SRE	1/07/92	<500	not stated		EU	CEP
RUNOFF SRE	1/06/92	<500	not stated		EU	CEP
RUNOFFB100	11/26/89	0.00	1000.00	8359-2	EU	TMA/Norcal
RUNOFFB100	1/17/90	0.00	1000.00	8397-7	EU	TMA/Norcal
RUNOFFB100	2/05/90	0.00	1000.00	8397-7	EU	TMA/Norcal
RUNOFFB100	2/05/90	50.80	178.00	G91470	EU	CE/UST
RUNOFFB100	2/17/90	1.92	211.00	G92340	EU	CE/UST
RUNOFFB100	3/20/91	<500	not stated		EU	CEP
RUNOFFB100	12/29/91	<500	not stated		EU	CEP
RUNOFFB100	1/06/92	<500	not stated		EU	CEP
RUNOFFB100	1/07/92	<500	not stated		EU	CEP
RUNOFFRMDF	11/26/89	0.00	1000.00	8359-3	EU	TMA/Norcal
RUNOFFRMDF	1/17/90	0.00	1000.00	8397-3	EU	TMA/Norcal
RUNOFFRMDF	2/05/90	-2.87	175.00	G91490	EU	CE/UST
RUNOFFRMDF	2/05/90	0.00	1000.00	8397-3	EU	TMA/Norcal

Table 7-1. Summary of Tritium-in-Water Analyses
(Sheet 8 of 9)

RUNOFFRMDF	2/17/90	-5.76	211.00	G92310	EU	CE/UST
RUNOFFRMDF	12/29/91	<500	not stated		EU	CEP
RUNOFFRMDF	1/06/92	<500	not stated		EU	CEP
RUNOFFRMDF	1/07/92	<500	not stated		EU	CEP
RUNOFFSBP1	11/26/89	0.00	1000.00	8359-1	EU	TMA/Norcal
RUNOFFSBP1	1/17/90	0.00	1000.00	8397-5	EU	TMA/Norcal
RUNOFFSBP1	2/17/90	15.40	212.00	G92370	EU	CE/UST
RUNOFFSBP1	2/05/90	0.00	1000.00	8397-5	EU	TMA/Norcal
RUNOFFSBP1	2/05/90	89.10	109.00	G91450	EU	CE/UST
RUNOFFSBP1	3/20/91	<500	not stated		EU	CEP
RUNOFFSBP1	1/07/92	<500	not stated		EU	CEP
RUNOFFSBP2	1/17/90	0.00	1000.00	8397-4	EU	TMA/Norcal
RUNOFFSBP2	2/05/90	0.00	1000.00	8397-4	EU	TMA/Norcal
RUNOFFSBP2	2/17/90	-50.90	208.00	G92280	EU	CE/UST
RUNOFFSBP2	12/29/91	<500	not stated		EU	CEP
RUNOFFSBP2	3/20/91	<500	not stated		EU	CEP
RUNOFFSBP2	1/06/92	<500	not stated		EU	CEP
RUNOFFSBP2	1/06/92	<500	not stated		EU	CEP
RUNOFFSBP2	1/07/92	<500	not stated		EU	CEP
R-1 POND	6/21/90	7.55	7.95	T0621A	R&NS	ITAS/EE
R-2A POND	9/18/89	34.00	8.40	T918A	R&NS	UST/EE
R-2A POND	1/13/90	0.00	1000.00	8397-1	EU	TMA/Norcal
R-2A POND	1/17/90	0.00	1000.00	8397-3	EU	TMA/Norcal
R-2A POND	2/05/90	0.00	1000.00	8397-3	EU	TMA/Norcal
R-2A POND	2/05/90	0.00	1000.00	8397-1	EU	TMA/Norcal
R-2A POND	6/21/90	20.30	8.99	T0621B	R&NS	ITAS/EE
SH- 4	9/09/89	-75.80	124.00		GRC	UST
SH- 4	9/09/89	<1000	not stated		GRC	TMA/Norcal
SH- 5	11/29/89	-202.00	239.00		EU	UST
SH- 6	11/29/89	-12.20	249.00		EU	UST
SH- 7	9/09/89	<1000	not stated		GRC	TMA/Norcal
SH- 7	9/09/89	-80.50	124.00		GRC	UST
SH- 7	11/29/89	-258.00	235.00	8359-4	EU	UST
SH-11	9/09/89	<1000	not stated		GRC	TMA/Norcal
SH-11	9/09/89	-43.10	126.00		GRC	UST
SRE POND	9/17/89	20.10	7.42	T917D	R&NS	UST/EE
SRE POND	6/28/90	19.10	8.77	T0628B	R&NS	ITAS/EE
S-2 TANK	9/19/89	464.00	43.90	T919B	R&NS	UST/EE
S-2 TANK	6/21/90	453.00	44.00	T0621C	R&NS	ITAS/EE
T024 COOLN	1/27/92	4070.00	352.00	20158101	RP&HPS	ITAS/EE
T024 FRNCH	9/17/89	54.00	9.85	T917E	R&NS	UST/EE
T024 FRNCH	6/26/90	64.70	12.10	T0626A	R&NS	ITAS/EE
T056 PIT	9/19/89	145.00	17.00	T919D	R&NS	UST/EE
T056 PIT	6/28/90	130.00	17.20	T0628A	R&NS	ITAS/EE
T059 FRNCH	7/13/89	1890.00	538.00		EPA/LV	CEP
T059 FRNCH	9/01/89	384.00	122.00	B-2	ETEC	RFP
T059 FRNCH	9/11/89	311.00	136.00	7169-1	GRC	BCL/UST
T059 FRNCH	9/19/89	297.00	29.70	T919A	R&NS	UST/EE
T059 FRNCH	2/05/90	972.00	88.00	T0205	R&NS	ITAS/EE
T059 FRNCH	7/13/90	1190.00	107.00	T0713	R&NS	UST/EE
T059 FRNCH	7/17/90	411.00	40.40	T059	R&NS	ITAS/EE
T059 FRNCH	1/27/92	471.00	46.50	20158102	RP&HPS	ITAS/EE
TANK 814	3/10/91	48.10	184.00		GRC	ITAS
WS- 4A	9/09/89	-155.00	125.00		GRC	BCL/UST
WS- 4A	9/09/89	<1000	not stated		GRC	TMA/Norcal
WS- 4A	12/06/90	-67.20	195.00		GRC	ITAS
WS- 5	9/09/89	-216.00	119.00		GRC	BCL/UST

Table 7–1. Summary of Tritium–in–Water Analyses
(Sheet 9 of 9)

|						
WS- 5	9/09/89	<1000	not stated		GRC	TMA/Norcal
WS- 5	6/29/90	12.70	8.37	WS-5	GRC	ITAS/EE
WS- 6	9/11/89	-128.00	125.00	7276-4	GRC	BCL/UST
WS- 6	9/11/89	<1000	not stated	8297-4	GRC	TMA/Norcal
WS- 7	12/06/90	78.00	229.00		GRC	ITAS
WS- 7	12/06/90	187.00	235.00		GRC	ITAS
WS- 7	3/08/91	-70.20	178.00		GRC	ITAS
WS- 7	12/07/91	-48.10	209.00		GRC	ITAS
WS- 7	12/07/91	-457.00	500.00		GRC	CEP
WS- 7	3/25/92	100.00	300.00		GRC	CEP
WS- 8	9/09/89	<1000	not stated		GRC	TMA/Norcal
WS- 8	9/09/89	-258.00	138.00		GRC	BCL/UST
WS- 9A	9/12/89	<1000	not stated	8297-5	GRC	TMA/Norcal
WS- 9A	9/12/89	-53.40	127.00	7276-5	GRC	BCL/UST

8.0 DETERMINATION OF NATURAL TRITIUM

To further investigate the levels of tritium present in natural water in the local area, to permit some judgment as to what is "natural" and what is "artificial" tritium, the results identified by consideration of the previous plot as "natural" are displayed in Figure 8–1, as a cumulative probability plot. In this plot, an estimated Gaussian distribution, determined by a least–squares fit to the data, is shown by the diagonal straight line passing through the points. If the points were perfectly selected from a Gaussian distribution, the points would all be exactly on the line. The observed agreement is quite good.

This natural or background tritium set consists of the following samples and results (in pCi/L ± 2–sigma):

1.	17th and G Streets surface drainage	42.9 ± 9.07
2.	Chatsworth swimming pool	36.7 ± 9.50
3.	Tap water from LADWP, Chatsworth	35.8 ± 8.54
4.	R–2A pond (9/18/89)	34.0 ± 8.40
5.	Bell Creek (9/18/89)	30.0 ± 8.12
6.	Tap water from LADWP, Canoga Park	29.7 ± 10.1
7.	ETEC Power Pak cooling tower water	28.8 ± 8.51
8.	Arrowhead bottled drinking water	26.1 ± 7.69
9.	R–2A pond (6/21/90)	20.8 ± 8.99
10.	SRE pond (9/17/89)	20.1 ± 7.42
11.	SRE pond (6/28/90)	19.1 ± 8.77
12.	RMDF pond	17.0 ± 7.30
13.	Ventura County Waterworks, Moorpark	16.6 ± 8.52
14.	Rainfall (9/17/89)	15.8 ± 7.14
15.	Canadian Glacier bottled drinking water	15.4 ± 8.31
16.	WS–5 (depth to water 405 ft)	12.7 ± 8.37
17.	RD–18 (depth to water 84 ft)	11.3 ± 8.08
18.	RD—7 (depth to water 70 ft)	10.0 ± 7.96
19.	Dead Water (UST)	8.86 ± 7.97
20.	RD–25 (depth to water 50 ft)	8.73 ± 7.88

21.	R–1 pond	7.55 ± 7.95
22.	RD–21 (depth to water 105 ft)	5.84 ± 7.72
23.	Dead Water (UST)	4.34 ± 3.27
24.	Dead Water (UST)	3.33 ± 3.13
25.	RD–25 (depth to water 50 ft)	3.24 ± 6.93
26.	Dead Water (UST)	1.99 ± 3.15
27.	Dead Water (UST)	1.50 ± 7.78
28.	Dead Water (UST)	0.53 ± 3.03
29.	RD–22 (depth to water 303 ft)	–0.58 ± 7.37
30.	Golden Wilderness bottled drinking water	–3.07 ± 7.32
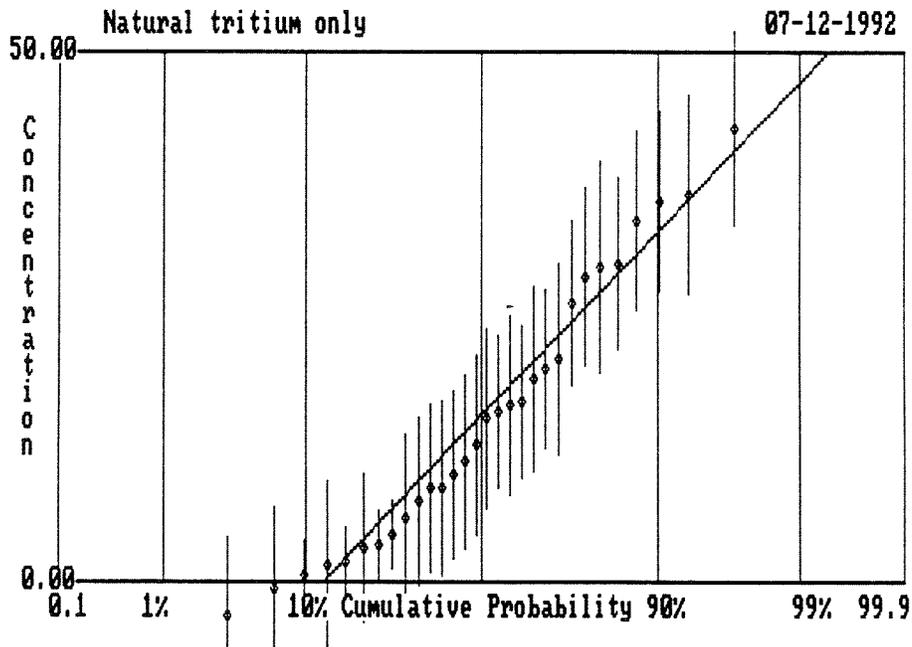
Figure 8–1. Cumulative Probability Plot of Results of Tritium–in–Water Analyses by U.S. Testing, Using Electrolytic Enrichment, for "Natural" Water Samples. The Straight Line Through Most of the Data Points Represents an Approximate Gaussian Distribution

The median value for this set (calculated as the average of the 2 central values) is 14.1 pCi/L. (For a Gaussian distribution, the median and the mean are the same. For a distribution that is nearly Gaussian but is distorted at the high and low ends, the median is a more "robust" estimation of the mean of the distribution than is the arithmetic mean of the data values. That is, this estimate is less likely to be affected by the extreme values.) The uncertainty is approximately ± 6.57 pCi/L (2–sigma = 13.1).

Because of differences between groundwater, surface water, and rain water, it is perhaps more appropriate to state that tritium in water may be expected to be in the range of 0 to 50 pCi/L.

All analytical results discussed in this section were obtained by use of the more sensitive electrolytic enrichment method. Utilizing the less sensitive standard EPA 906.0 method, background concentrations of natural tritium would be expected to produce results of plus or minus several hundred pCi/L, up to "less than 1000."

9.0 DETERMINATION OF ARTIFICIAL TRITIUM

Artificial tritium has been dependably found in water at only three locations at SSFL: Building T059, in the french drain and associated ground monitoring wells; Building T024, in a sample of shield cooling water taken from cooling tubes embedded in concrete; and Wells RD–34A and B, which are down–gradient from Building T010.

9.1 BUILDING T059

The static water levels for the wells (feet above mean sea level) and other water associated with T059, and the average tritium concentrations (pCi/L) are:

|	Feet	pCi/L
T059 french drain	1765	440
RD–28	1763.3	670
T056 pit pond	1779	138
RD–24	1781.0	112
RD–25	1765.7	5.2

These results indicate the disjointed nature of the groundwater system in Area IV: RD–28 and RD–25 are the groundwater wells closest to the T059 reactor test vault (both within about 100 ft) and yet one shows a notably high tritium concentration and the other shows a notably low concentration.

The variation of activity with time observed in the water from the T059 french drain (including water in the S–2 tank) and from the adjacent well, RD–28 is shown in Figure 9–1. There is no clear systematic variation in the water from T059.

9.2 BUILDING T024

Tritium has been clearly found in the shield cooling water in the cooling coils embedded in the concrete shielding of one of the test vaults in T024. The concentration observed, 4,070 ± 352 pCi/L, could have resulted from diffusion of tritium that was produced in the adjacent concrete. This water is of limited volume and is confined within the facility and therefore is not likely to be released to groundwater. In addition, the concentration is far below applicable limits.

Tritium is also shown by use of the extrasensitive enriched tritium method in two samples from the french drain in the basement of T024, where there are test vaults in which SNAP reactors were operated, and where activation of the concrete shielding occurred, as indicated by observations of Co–60 and Eu–152. While lithium (the assumed source of tritium in the concrete shielding at T059) is common in aggregate derived from granite in California, its presence in the limestone aggregate used in T024 (to minimize activation) is uncertain. Lithium in this concrete was determined by analysis to be 6.2 ppm. Therefore, neither the presence nor the absence of artificial tritium in the groundwater at T024 can be established by present information.
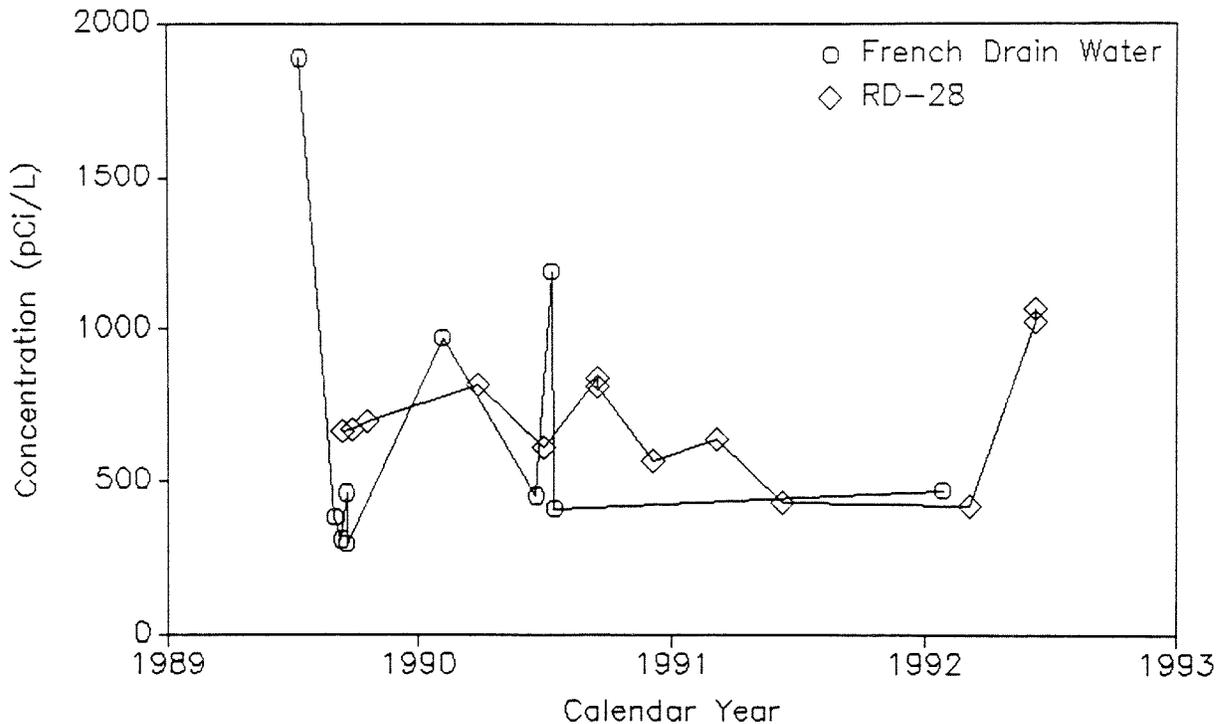
Figure 9-1. Time Variation of Monitoring Data at T059

The results of analysis of the groundwater samples from the T024 French drain are:

Date	Sampler	pCiL
9/17/89	R&NS–UST/EE	54.0 ± 9.85
6/26/90	R&NS–UST/EE	64.7 ± 12.10
	Weighted average	58.3

This indication is barely above the range of natural water at the site of 0 to 50 pCi/L (as determined by the electrolytic enrichment technique).

9.3 WELLS RD–34A AND B

The highest concentrations of tritium in water that have been found to date are from Well RD–34A. This is the shallowest well of a three–well cluster. This cluster is down–gradient of T010 and down–slope from the RMDF leachfield. Lesser concentrations have been found in water from the somewhat deeper well, RD–34B.

As shown in Figure 9–2, the concentration in water samples from RD–34A increased with time since the initial sampling and then decreased to a lower level.
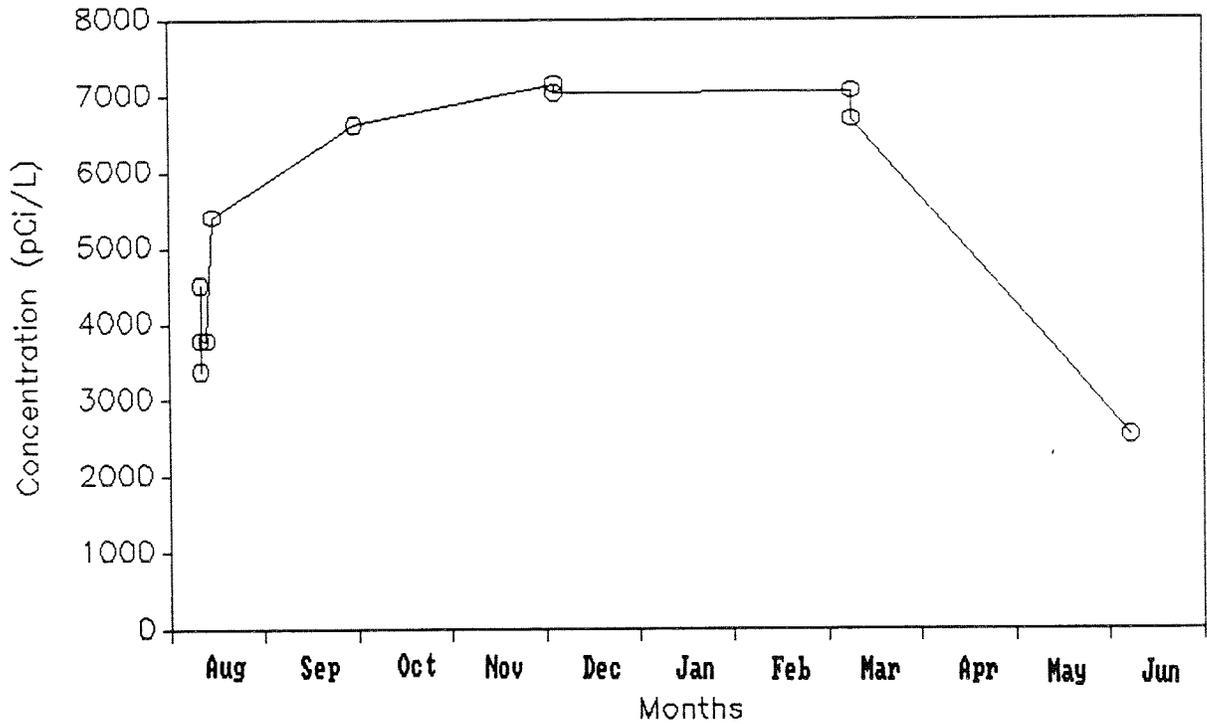
Figure 9–2. Time Variation of Monitoring Data at RD–34A

9.4 OTHERS

Occasional, isolated analyses appear to show detectable tritium. These are reviewed to determine if the result is consistent with prior and subsequent results. In some instance, repeat analyses or duplicate samples are used. Results that have not been confirmed have been omitted from the detailed discussion.

10.0 SUMMARY

This review has found that minor releases of tritium produced in the SSFL reactors may have occurred. Based on extensive sampling of groundwater and detailed theoretical analysis of possible tritium production and release, levels of tritium in the environment at SSFL have not posed a health risk to workers or the public. All estimated releases were well below regulatory limits. Extensive sampling and analysis of water for tritium has confirmed the presence of artificial tritium in groundwater at some locations of SSFL, at low concentrations, far below regulatory limits for radioactivity in the environment, and even well below limits imposed on suppliers of drinking water. Elevated values are found in the vicinity of T010, the RMDF leach field, and T059.

For the purpose of further investigation, the S8ER reactor at T010 appears to have produced the greatest potential concentration of tritium in groundwater, followed by T059 and the SRE. The RMDF leach field also appears to have been a possible location for release of tritium to the groundwater. Continued monitoring of groundwater wells near the Sodium Disposal Facility is recommended.

D641-0036/tab

APPENDIX A
ENGINEERING DATA

Nuclear Data

Tritium half-life	12.346	years.
Tritium yield in fission	1.24E-04	tritons per thermal fission.
Li-6 natural abundance	7.420	atom % (ORNL Stable Isotopes)
Li-6 (n,t) cross-section	940.000	barns. (thermal flux)
Lithium atomic mass	6.941	amu
U-235 fission cross-section	582.600	barns. (thermal flux)
nu-bar (U-235,thermal)	2.425	neutrons per fission
U-235 atomic mass	235.044	amu
B-10 (n,t) cross-section	0.020	barns. (fast-flux average)
B-10 (n,) cross-section	3837.000	barns. (thermal flux)
Deuterium cross-section	0.00046	barns. (thermal flux)
Deuterium abundance	0.01500	atom %
Normalized fission rate	3.3E+10	fissions/sec per watt.
Avogadro's number	6.02E+23	atoms per gram-mole.
Activity conversion	3.7E+10	dps per curie
Days per year	365.250	

Facility:	T143		T010		T059	T028		T024	
Reactor specifications	SRE-I	SRE-II	SER	S8ER	S8DR	STR	STIR	S2DR	S10FS3
U-235 mass (kg)	83.4	81.0	3.0	6.1	7.6	2.5	3.0	6.0	6.0
Flux in fuel (n/cm2/sec)	2.43E+12	2.50E+12	2.10E+11	1.11E+12	9.10E+11	9.49E+09	5.95E+10	7.87E+10	1.39E+11
Nominal power (MWt)	20.000	20.000	0.050	0.600	0.619	0.050	1.000	0.065	0.037
Average power (MWt)	9.178	9.178	0.028	0.307	0.314	0.001	0.008	0.021	0.038
Start date	07/01/58	02/01/63	09/01/59	11/12/63	05/01/68	12/01/61	08/01/64	04/01/61	01/01/65
Shutdown date	07/01/59	02/01/64	12/01/60	04/15/65	12/01/69	07/01/64	02/01/74	12/01/62	03/01/66
MWd actual	3350	3350	13	215	182	1	28	13	16
Graphite moderator mass (kg)	34500	34500				3625	3625		
Lithium content (ppm)	1.20	1.20				1.20	1.20		
Average flux (n/cm2/sec)	1.24E+13	1.24E+13				1.27E+10	9.68E+10		
Coolant volume (cm3)	280416	280416	1849	2157	2593			1849	1849
Coolant mass (kg)	224.3	224.3	1.5	1.7	2.1			1.5	1.5
Lithium content (ppm)	1.00	1.00	1.00	1.00	1.00			1.00	1.00
Control-rod flux (n/cm2/sec)	4.86E+12	5.01E+12	4.19E+11	2.22E+12	1.82E+12			1.57E+11	2.78E+11
Fast flux fraction (>1 MeV)	0.03750	0.03750				0.00625	0.00625		
Thermal flux fraction	0.96250	0.96250				0.99375	0.99375		
delta-k controlled	0.01340	0.01340				0.00025	0.00025		
Flux a 50 kW (IAEA, STR)						8.60E+11	8.60E+11		
Flux a average power (STR, STIR)						1.82E+10	1.39E+11		
Deuterium concentration in light-water moderator(atoms/cm3)						1.00E+19	1.00E+19		
Irradiated moderator volume (cm3)						3.86E+04	3.86E+04		

Facility parameters
Ordinary concrete shield

Concrete density (g/cm3)	2.35	2.35	2.35	2.35	2.35	2.35	2.35	2.35	2.35
Lithium content (ppm)	5.87	5.87	6.37	6.37	6.87	6.37	6.37	6.20	6.20
Date of shield removal	02/01/79	02/01/79	02/16/78	02/16/78	10/01/91	03/01/76	03/01/76	10/01/91	10/01/91
Surface flux (n/cm2/sec)	3.00E+09	3.00E+09	2.88E+09	3.16E+10	6.20E+09	5.48E+00	4.17E+01	6.79E+07	1.20E+08
Thickness (inches)	48	48	27	27	24	15	15	48	48
Inner diameter (inches)	156	156	40	40	110	120	120	216	216

Height (inches)	180	180	180	180	180	120	120	180	180
Volume (cm3)	8.44E+07	8.44E+07	1.06E+07	1.06E+07	2.82E+07	1.39E+07	1.39E+07	1.25E+08	1.25E+08
Irradiated shield area (cm2)	6.92E+05	6.92E+05	1.10E+05	1.10E+05	6.12E+04	3.65E+05	3.65E+05	1.02E+06	1.02E+06
Shield outside area (cm2)	9.55E+05	9.55E+05	2.67E+05	2.67E+05	5.79E+05	4.21E+05	4.21E+05	1.32E+06	1.32E+06
Neutron relaxation length (cm)	11	11	11	11	11	11	11	11	11

Environmental soil

Soil density (g/cm3)	1.40	1.40	1.40	1.40	1.40	1.40	1.40	1.40	1.40
Lithium content (ppm)	27.33	27.33	27.33	27.33	27.33	27.33	27.33	27.33	27.33
Date of soil removal	02/01/79	02/01/79	02/16/78	02/16/78			03/01/76	03/01/76	
Irradiated soil area (cm2)	6.92E+05	6.92E+05	1.10E+05	1.10E+05	6.12E+04	3.65E+05	3.65E+05	1.02E+06	1.02E+06
Neutron relaxation length (cm)	11	11	11	11	11	11	11	11	11
Soil grain size (cm)	0.100	0.100	0.100	0.100	0.100	0.100	0.100	0.100	0.100

H-3 diffusion rate in concrete and soil   0.003537 cm/year

High-Density concrete shield

Date of shield removal	02/01/92	03/01/76	03/01/76
Surface neutron flux (n/cm2/sec)	1.00E+11	4.24E+09	3.23E+10
Thickness (inches)	24	24	24
Inner diameter (inches)	102	61	61
Height (inches)	108	60	60
Volume (cm3)	1.68E+07	5.67E+06	5.67E+06
Irradiated shield area (cm2)	2.76E+05	7.42E+04	7.42E+04
Neutron relaxation length (cm)	10	10	10

Shielding sand

Sand density (g/cm3)	1.20
Lithium content (ppm)	6.37
Date of sand removal	09/25/88
Surface neutron flux (n/cm2/sec)	1.80E+10
Shield inner diameter (inches)	60
Shield height (inches)	144
Shield volume (cm3)	8.84E+15
Irradiated shield area (cm2)	1.93E+05
Neutron relaxation length (cm)	24

Test shields

Reactor "point source" (n/sec)	1.02E+15
Distance to shield (inches)	22
Flux at shield surface (n/cm2/sec)	2.59E+10
Irradiated shield diameter (inches)	40.000
Irradiated shield area (cm2)	8107.3
Neutron relaxation length (cm)	13.5
Effective volume (cm3)	109448.8
LiH density (g/cm3)	0.720