4.1.2.7 Occupational dose

At the L-Reactor, occupational doses would be maintained as low as reasonably achievable. All personnel who work in or enter areas that have radiation-exposure potential receive personal monitoring devices. In addition, a comprehensive bioassay program is maintained for all employees who work in areas where there is a potential for a biological uptake of radioactivity.

Table 4-18 lists the total whole-body dose commitments to workers in the P-, K-, and C-Reactor areas for 1976 through 1980. Based on these data, the total average annual dose commitment to workers in the L-Area would be about 69 person-rem per year. The average work force in each reactor area is about 375 people; thus, the average annual individual dose to workers in the L-Area would be about 185 millirem per year.

Table 4-18. Total doses to workers in P-, K-, and C-Areas

Year	Dose (person-rem)
1976	217.2
1977	231.2
1978	202.0
1979	184.4
1980	203.7
Average	207.7
Average per reactor-year	69.2

The dose commitment to workers during this recent period can be compared to the experience of the 1960-1968 period, during which the annual occupational dose commitment in the P-, K-, C-, and L-Areas averaged 200 person-rem per reactor year (Du Pont, 1982a). A continuing program is maintained to reduce the occupational dose further.

4.1.2.8 *Solid radioactive waste*

About 570 cubic meters of solid radioactive waste would be generated annually at L-Reactor. This waste would be packaged and transported to the SRP low-level waste burial ground. The burial ground is divided into sections to accommodate different levels of radioactivity. The waste is buried in earthen trenches that are about 6 meters deep and 6 meters wide. The exact location of the burial trenches is defined, and accurate records are kept of the contents of each trench. About 40 acres of the burial ground area are available for future use.

The volume of low-level waste added to the burial ground due to L-Reactor operation would occupy about 1 acre of the burial ground area for each 10 years of operation. Offsite radiological effects of burial operations would be negligible.

4.2 ACCIDENTS

TC This section describes the environmental impacts and risks of reactor accidents. It demonstrates that L-Reactor safety systems are designed and would be operated in such a manner that the risk to the public from accidental releases of radioactivity would be extremely small.

4.2.1 Reactor accidents

Radiological protection for the operating staff, the public, and the plantsite would be provided by extensive protective devices and systems at L-Reactor, all designed to ensure that accidents would be prevented, arrested, or accommodated safely. The requirements for these protection systems are based on a spectrum of postulated occurrences and accidents that the plant design must accommodate safely.

The occurrences considered range from relatively minor events such as routine equipment malfunctions to postulated accident situations with a potential for serious consequences. The predominant focus is on prevention of any accidents that could release radioactive material in excess of permissible limits.

Analyses of accidents postulated for the Savannah River Plant reactors are applicable to L-Reactor and used to:

- Ensure that the reactor would operate with acceptably low risk to the public and plant employees and to provide a basis for improved reactor systems that could lower these risks still further.

- Set reactor operating limits for each operating cycle, such that the reactor protective instrumentation and shutdown systems could terminate postulated transients without damaging reactor fuel, the reactor tank, or the radioactivity confinement system.

- Provide assurance that the radioactivity confinement system would operate reliably even in the most serious accidents.

- Specify the offsite emergency response system needed and how the system should be used.

Appendix G describes reactor-accident analyses in more detail.

4.2.1.1 Characteristics of reactor accidents

Accident types

The two types of reactor accidents of primary concern at SRP are release of fission products or other radionuclides from the irradiated reactor fuel and targets, and release of activation tritium from the reactor moderator. The release of fission products is most likely to occur due to fuel or target melting, which might result from either power surges or cooling-system failures. The release of activation tritium from the reactor heavy water is most likely to occur from spills or pipe breaks.

The principal hazard of these accidents is that the released radionuclides become airborne and are carried either to the plant worker onsite or to the offsite population. Radionuclides can also be dispersed by the reactor liquid effluent streams, but the hazards of such dispersal are several orders of magnitude lower than those of airborne dispersal in an accident situation.

If a reactor fuel assembly melts, the materials that can be released to the reactor-room air have been assumed to be:

- 100 percent of the noble gases, primarily krypton and xenon
- 100 percent of the tritium from the lithium-aluminum components |TC
- 50 percent of the halogens, mainly iodine
- 1 percent of the other fuel materials as airborne particulates

If the reactor heavy water (D_2O) is spilled it can evaporate, carrying off any tritium present as DTO vapor. As initially charged, the L-Reactor heavy water would contain trace amounts of tritium, but the tritium in the heavy water could eventually build up to an equilibrium inventory of 5 million curies over a period of 10 years or longer. (The inventory varies with the operating history of the reactor and is now about 3.5 to 3.7 million curies in operating SRP reactors. To be conservative, a higher value of 5 million curies is assumed for accident consequence calculations. This is about 20 percent higher than the highest value ever observed in SRP reactors.) In the event of a spill of the full moderator inventory, about 3 percent of the tritium is assumed to evaporate during the 2-hour period after the spill and then to be released from the stack and dispersed during that period. TC

The SRP reactors, including L, are fitted with a confinement system to remove a large fraction of the radioactivity that might be released to the reactor room. In this confinement system, the reactor room is kept at a negative pressure by use of exhaust fans. The exhaust air is passed through moisture separators and then through high-efficiency particulate air (HEPA) filters and carbon filters to remove more than 99 percent of the particulates and the iodine. The noble gases are not removed by the filters. Airborne tritium is also assumed to be fully released. After filtration, the exhaust air is released through a 61-meter-high stack.

Fission products

Table 4-19 lists the radioactive fission product content for a fully irradiated SRP fuel assembly, the half lives of these fission products, the amounts that might become airborne in a meltdown, and the amounts that might be released through the confinement system.

As seen from the table, the fission products of primary concern from an SRP reactor accident would be noble gases and iodine. Most of these fission products have short half lives and are quite volatile.

Radiation exposures and health effects

The possible pathways by which accidental releases of airborne radioactivity from L-Reactor could result in radiation exposure to the offsite public and to the SRP workers include:

- Exposure to gamma radiation emitted by the radionuclides as they pass overhead (plume shine)

- Immersion in the plume of the release, resulting in inhalation of the radionuclides either with immediate exhalation or with retention in the
A-31 body (depending on the radionuclide biochemistry)

- Immersion in the plume of the release, resulting in a skin contact dose due to tritium

- Exposure to gamma radiation emitted by radionuclides deposited on the ground from the air (ground shine)

- Ingestion of radionuclides in contaminated drinking water and food

Because of the volatile nature of the radionuclides that could be emitted in an L-Reactor accident and their associated short half lives (tritium has a comparatively long radioactive half life, but a short biological half life), the last two pathways would be less important than the first three in the accident analysis.

The radiation doses calculated from the spectrum of postulated accidents associated with L-Reactor (Section 4.2.1.4) are too low to produce any short-term clinical effects or fatalities. The concern, rather, is with possible latent health effects (i.e., cancers or genetic changes).

Extensive studies have been made in relating comparatively low levels of radiation exposure and health effects. The problem is difficult primarily because the effects are statistically so low as to be difficult to measure. For purposes of this analysis, radiation doses were calculated based on dose conversion factors from the International Council on Radiological Protection report ICRP-30.

4.2.1.2 Accident experience and prevention at SRP

Safe operation of the production reactors is implemented by (1) explicit definition of the safe limits of operation, (2) explicit written procedures for normal and abnormal operations, (3) multiple and diverse engineered safety systems and (4) in-depth technical support onsite. This system of operation was in place when the first reactor was started at SRP and has been improved over the years when deficiencies were identified.

For long-term safety, an important function is the ability to spot weaknesses or adverse trends. Each deviation from approved operating procedures is recorded and promptly investigated by onsite technical personnel. If there appears to be a significant question of reactor safety, the reactor is shut down until it can be demonstrated that operation will be within the envelope of acceptable conditions required by the reactor operation and Technical Specifications and Technical Standards, which are established by DOE and the operating contractor, respectively.

Safety considerations override production considerations, and precautionary reactor shutdowns have occurred to investigate possible safety questions. The research at Savannah River Laboratory (SRL) ensures that the latest methods and equipment are evaluated for application to Savannah River Plant. Many important improvements have been made to SRP reactors; in the safety-related areas of thermal analysis, core physics, and monitoring and diagnosis, they equal the current state of the art. These improvements are summarized in Appendix J. Research at SRL includes human factors as well as plant equipment. The incident at Three Mile Island has been studied; lessons learned that are applicable to SRP reactors are being implemented (e.g., an improved reactor training program, the construction of a reactor simulator).

A comprehensive Safety Analysis is the basis for a defense-in-depth safety approach in which possible accident initiators are identified and eliminated to the maximum extent practical, multiple shutdown systems are provided to terminate, without damage, any accidents that do occur, and radioactivity confinement and other systems are installed to minimize the offsite effects of reactor damage if it does happen (Du Pont, 1983a). The emphasis in the Safety Analysis is on accident prevention and mitigation, but it also calculates the consequences of possible occurrences. TC

Provisions for independent safety reviews are required by DOE policy for each level of organization, including contractors, the field offices, and Headquarters. As part of this process, the Atomic Energy Commission's Advisory Committee on Reactor Safeguards served as an independent review body from 1960 to 1974. Numerous reviews by special committees and boards have been conducted periodically, including the Shon Committee in 1971, the Crawford Committee in 1980, and the Ditto Committee in 1981. The process also included the use of consultants. A formal safety consultant review policy was established after 1974. Currently, consultants are used on the Reactor Safety Advisory Committee initiated by the contractor in 1982. Significant steps to strengthen independent reviews were identified and taken as a result of post-TMI-2 reviews. These steps included organizational changes and staff to provide additional independent overview within DOE organizations.

SRP reactors have operated for more than 115 reactor-years with no accidental criticality or abnormal releases to the environment.

The most serious accidents that have occurred at SRP reactors are:

- A sizable moderator spill that occurred during the early stages of operation. At the time of the spill, the moderator contained very little tritium, so the radiation effects of the spill were negligible.

- In 1970, a special source rod melted while it was being held in the discharge machine. The confinement system worked as designed and 99.99 percent of the radioactivity released was trapped and recovered with negligible offsite exposure. This accident was the result of administrative error; appropriate procedural controls have been implemented to prevent a recurrence.

These and other reactor incidents are described in more detail in Appendix G and the Safety Analysis Report (Du Pont, 1983a).

4.2.1.3 Mitigation of accident consequences

Numerous reactor design features provide the ability to reduce the consequences of accidents. The most important of these include the following:

Reactor shutdown systems

Several redundant and diverse systems operate to shut down the reactor rapidly, if necessary.

L-Reactor would have the same defenses against reactivity transients that other SRP reactors have. These defenses would include flow and temperature sensors for each fuel assembly, which are monitored by two sets of redundant computers (control computers and safety computers). The control computer(s) would detect rapidly any reactivity transient that might begin and would cause the normal control-rod system to insert to terminate the transient safely--the first line of defense. If the normal control-rod system fails to terminate the transient, the safety computer(s) would activate the safety-rod drop system that would shut down the reactor within about 1 second--the second line of defense. If the safety rods do not shut down the reactor rapidly, the safety computer(s) would automatically activate the injection of liquid "poison" into the reactor moderator/coolant to accomplish the same safe shutdown--the third line of defense. The few reactivity transients that have occurred have been of a small magnitude, were controlled by the normal control-rod system, and did not require either backup system to operate (safety-rod drop or "poison" injection).

Emergency cooling system

An emergency cooling system (ECS) is provided to protect against the consequences of two postulated accidents: (1) loss of heavy-water coolant and (2) loss of heavy-water circulation.

Emergency cooling of the SRP reactors is accomplished by the addition of light water to the primary reactor cooling system. This water is enhanced in loss-of-coolant accidents by recirculation of the emergency light water by the primary heavy-water circulating pumps.

On activation, the ECS system provides an initial 75,000 liters of borated water for nuclear poisoning by directing all ECS water flow through a large pipe that contains the borated water. The poison solution is forced through the assembly coolant channels and into the moderator. By the time unpoisoned H_2O reaches the coolant channels, sufficient heavy water moderator is displaced with poisoned water to prevent any possible criticality.

Three primary sources and a secondary source of water for the emergency cooling system are provided and include the following:

1. A diesel-driven booster pump that supplies water from the 95-million-liter 186-L basin (primary).

2. A header with a diameter of 107 centimeters pressurized by five pumps drawing water from the 95-million-liter basin (primary).

3. Another header with a diameter of 107 centimeters pressurized by five additional pumps.

4. A line pressurized by the river station pumps. Because the water directly from the river can contain debris that could plug flow channels and orifices in the reactor components, this source is valved off from the ECS and would be used only if all other sources had failed (secondary).

Airborne activity confinement system

The L-Reactor is equipped with an airborne activity confinement system (see Figure G-1). In the event of an accident, an airborne fission product release could occur in the reactor room with the possibility of some release in the heat exchanger bay or pump room. The air from these areas would be exhausted through a set of confinement filters before release to the stack.

During normal operation, the process areas would be closed and maintained at a negative pressure with respect to atmosphere to ensure that all air from the process areas is exhausted through the activity confinement system. Three large centrifugal fans would exhaust the air from the process areas. Two of these fans normally would be online, but only one would be necessary to maintain the negative pressure. The fan motors could be powered by two independent sources of electricity:

- The normal building power, through at least two substations
- The diesel-generated emergency building power

In addition, each online fan has a backup motor; any two fans could be powered by the dedicated diesel generators.

Exhaust filters would remove moisture, particulates, and halogens. The filter banks are enclosed in five separate compartments; three to five of these compartments would be online during operation. Each compartment can be isolated for maintenance and testing; each contains the following filter banks, in the order of air-flow treatment:

- Moisture separators, designed to remove about 99 percent of entrained water (spherical particles measuring 1 to 5 microns) to protect against significant impairment of the particulate filters

- Particulate filters, designed to retain more than 99 percent of all particulates with diameters of 0.3 micron or larger

- Activated carbon beds that use an impregnated carbon to retain halogen activity

As shown in Figures 4-7 and 4-8, L-Reactor is completely surrounded by a massive concrete structure, which in combination with the confinement system forms a barrier of high reliability against the possible release of radioactive material. The confinement system has the capacity to accommodate unexpected gas or energy releases. Hydrogen formed during an accident would be swept from the building by the high ventilation flow before explosive concentrations could be reached. Even with steam or hydrogen explosions for the worst hypothetical accident, the integrity of the structure and confinement system (including filters) would not be breached by rupture. Durant and Brown (1970) present a detailed analysis of a most severe hypothetical accident affecting the confinement system; this analysis specifically addresses the impact of hydrogen and steam explosions. Durant et al. (1966) documents confinement system tests that confirm the confinement system can withstand the severe accident conditions described above with a large margin of safety.

For all reactor accidents, the airborne activity confinement system is assumed to operate. The three exhaust fans described above would provide a high degree of assurance that at least one would remain in operation to maintain the process-area exhaust through the filter system. The probability that all three fans would fail is estimated to be 10^{-4} per year. Such a fan failure happening at the same time as one of the described accidents would be extremely unlikely.

Reactor room spray system

A system of nozzles is provided in the reactor room to spray cooling water on an irradiated assembly accidentally dropped during unloading operations. The spray pattern from these nozzles covers the area traversed by the discharge machine.

Site features

The site feature that would most effectively mitigate the consequences of an accident at L-Reactor is the 9-kilometer distance to the nearest SRP boundary. Although South Carolina Highway 125 is only 5 kilometers from L-Reactor, there are existing procedures for stopping traffic and clearing all personnel off the highway within a short time of any incident on the Savannah River Plant. (For more detail concerning site features, see Section 3.1.)

Emergency planning

Onsite. The L-Reactor operating procedures include an Emergency Response Plan, which includes specific policies and procedures to minimize injuries and property damage caused by accidents, disasters, or deliberate damage in the reactor areas. The plan deals with sheltering or evacuation, nuclear incidents, civil defense readiness, missile or air attack, rescue plan, natural disasters and alerts, bomb threats, off-plant accidents, and forced entry or terrorist attack. (For more detail concerning Onsite Emergency Planning, see Appendix G.)

Offsite. DOE has various service agreements for assistance or special support with Fort Gordon and with Talmadge Hospital in Augusta, Georgia. DOE also has fire-fighting mutual aid agreements with the City of Aiken, South Carolina, and the South Carolina Forestry Commission. Memos of Understanding between DOE and the States of South Carolina and Georgia cover notification and emergency responsibility in the event of a potential or actual radiological emergency at the SRP. (For more detail concerning Offsite Emergency Planning, see Appendix H.) DOE continually reviews and updates its emergency planning procedures for consistency with applicable industrial and regulatory standards.

WIND system. The Weather Information and Display (WIND) System (Garrett et al., 1983) is an automated emergency response system for real-time predictions of the consequences of liquid and atmospheric releases from the Savannah River Plant. Site-specific features of the system include meteorological towers at each production area that are instrumented at the stack height, computer terminals at each production area that can be used to run emergency response codes remotely, codes that use empirical information on atmospheric diffusion and deposition gathered at the Savannah River Plant (Garrett, 1981; Carlson et al., 1982), and stream transport and diffusion codes that have been calibrated with dye tests in the SRP streams (Buckner et al., 1975). (For more detail concerning WIND, see Appendix G.)

4.2.1.4 Accident risk assessment

Accident description

Postulated events considered for safety evaluation of the L-Reactor are discussed in Appendix G and, more comprehensively, in the Safety Analysis Report (Du Pont, 1983a). Among these events are four postulated accidents that cover a spectrum of credible events with probabilities of greater than 10^{-6} per reactor-year that could release radioactive materials into the environment. Accidents with probabilities less than 10^{-6} per site-year are not considered credible.

Use of the probability of 10^{-6} per reactor-year as a threshold for credible reactor accidents has no absolute basis, but it is consistent with normal practice in the nuclear power industry. For example, this value can be derived from both an American National Standards Institute (ANSI) standard and the U.S. Nuclear Regulatory Commission Standard Review Plan. ANSI/ANS-212-1978, Appendix B, uses the value of 10^{-6} per site per year as a cutoff probability, TC

below which combinations of events leading to accidents need not be considered
for design purposes. The cutoff value does not include the probability of the
consequences exceeding 10 CFR 100 dose guidelines, which is included in the NRC
Standard Review Plan (NUREG-0800) acceptance criteria of 10^{-7} per year. The
use of the 10^{-6} per site year value in the ANSI standard for accident proba-
bility is consistent with the NRC Standard Review Plan's value of 10^{-7} per
TC site per year for accident plus consequence probability because the probability
of the consequences exceeding 10 CFR 100 dose guidelines following an accident
are conservatively estimated to be less than 10^{-1}. The SRP use of the 10^{-6}
threshold is not for a so-called uncontrolled release, but for dividing
"treated-as-credible" from "treated-as-noncredible" accidents. Even with esti-
mates of accident probabilities beyond the 10^{-6} per reactor-year threshold,
radioactive releases are limited by the performance of the reactor confinement
system; they are not uncontrolled releases to the environment.

These four accidents are used for consequence and risk calculations. Other
accidents or events are discussed in Appendix G, including the failure of an
irradiated fuel or target component in the disassembly basin and various fuel-
melt accidents. None of the accidents postulated would cause offsite doses that
exceed either those adopted by DOE as safety limits for nuclear facilities (DOE
TC Order 5480.1A) or those adopted by NRC as guidelines for siting for commercial
power reactors (10 CFR 100). The four postulated accidents that cover the
spectrum of credible events and risks are:

Moderator spill. Tritium in the moderator could become airborne and be partially released to the confinement system following ECS actuation or any loss-of-coolant accident. Tritium released into the confinement system is discharged from the stack, because the confinement system has no mechanism for tritium removal.

Five million curies of tritium are assumed to be present in the moderator
of L-Reactor; this is the equilibrium value of tritium in the moderator and is
30 to 40 percent higher than present actual values for operating SRP reactors.
The full moderator inventory of tritium is unlikely to evaporate and discharge
to the atmosphere through the confinement system following any accident because
the moderator would flow first into the 225,000-liter tank and then to the
1,900,000-liter tank of the liquid activity confinement system, unless the acci-
TC dent is a spill in the process room; in that case, most of the moderator would
flow directly to the 1,900,000-liter tank. About 3 percent of the tritium is
assumed to evaporate during the 2-hour period after the postulated accident and
then to be released from the stack and dispersed during that period.

Discharge mishap. One irradiated fuel assembly could melt during a discharge operation under certain adverse (and improbable) conditions and release noble gases, iodine, and particulates. Fifty percent of the iodine and 100 percent of the noble gases available for release are assumed to escape the assembly and become airborne within the confinement system. More than 99 percent of that iodine reaching the carbon filter beds would be removed by the filter (a small fraction would desorb later and be released); 100 percent of the noble gases reaching the filters would pass through the filter. Half of the particulates released to the confinement system would reach the HEPA filters, where 99 percent of these particulates would be retained.

Reloading error leading to criticality. The highly localized damage postulated to occur following this accident would involve less than 3 percent of the core; melting would release iodine and fission products into the moderator. For this analysis, 50 percent of the iodine and all the noble gases were assumed to become airborne. Before the discharge operation began, the fission products would have decayed for a minimum of 14 hours. However, more fission products would be formed during the postulated criticality accident, and it was conservatively assumed that the fission product content of the core would be the equilibrium concentration at full power.

One-percent core melt due to a loss-of-coolant accident (LOCA). This accident is assumed to result from a double-ended pipe break in one of the six primary lines supplying heavy water to the reactor plenum. To compound this accident, the break is assumed to occur in one of the three primary lines having an emergency cooling-water injection line. Furthermore, a second emergency cooling-water addition system is assumed to be disabled. These assumptions of system operability are consistent with the single-failure criteria used on commercial power plants. SRP reactors are operated at power levels that limit core damage to 1 percent with only one of the three ECS operating. If the ECS operates as designed, no melting would occur. The amount of radioactivity available for release would be 1 percent of the noble gases and the iodine inventories in the core at the time of the accident. All released noble gases are assumed to become airborne. Fifty percent of the released iodine is assumed to become airborne. More than 99 percent of the released iodine would be trapped on the carbon filters; a small fraction would desorb later and be released from the stack.

Probability analysis

The following analyses are provided for each of the four hypothetical accidents:

Moderator spill. A 45,000-liter moderator spill (about 20 percent of the moderator inventory) occurred once at the Savannah River Plant during the early stages of operation. This spill was caused by a valving error while the reactor was shut down. Since then, unnecessary valves have been blanked, and moderator inventory procedures, level detection instrumentation, and leak detection instrumentation have been improved significantly. As a result, the Savannah River Plant has experienced more than 100 reactor-years of operation without a significant moderator spill. Today, the most probable scenario leading to a significant moderator spill is an unnecessary actuation of the ECS. The ECS has never activated; only once in 115 reactor-years of operation was there a spurious combination of reactor alarms and procedures that erroneously indicated the need to actuate the ECS. As a result, alarms and procedures were reanalyzed and improved. If inadvertently actuated, the ECS would result in a significant moderator spill only if the reactor is shut down and contains heat generating assemblies with primary (AC) process water pumps shut down (during reactor operation, moderator pressure at ECS injection points exceeds ECS pressure; the ECS source is restrained by check valves), which occurs about 10 percent of the time. Because of extensive reactor instrumentation that provides a comprehensive status of reactor parameters, components, and systems, an estimated 90-percent probability exists that unnecessary actuation of the ECS will be terminated before the majority of the moderator has been expelled from the reactor. Thus, the estimated probability of spilling most of the moderator is equal to or less than 10^{-4} per reactor-year. AY-9

Discharge mishap. The melting of a fuel or target assembly during discharge would require at least two concurrent failures (for example, a failure of the assembly-holding mechanism on the discharge machine resulting in the dropping of a slug-type assembly plus a failure of the reactor room spray-cooling system, or a failure of the discharge machine drive mechanism resulting in the stalling of the machine plus a failure of four independent sources supplying cooling water to the discharge machine; in the latter case, melting would not necessarily result because the reactor room spray-cooling system could be used to provide cooling if the discharge machine stalls and its cooling-water supplies are lost).

In 115 years of reactor operation, no assembly has been dropped during discharge, indicating that the probability of this event is on the order of 0.01 or less per reactor-year. A review of approximately 250 tests of the reactor room spray system indicates four incidents in which less-than-designed flow was obtained. The system consists of 12 valves with 9 nozzles per valve. In each of the four incidents, the area of the process room receiving a less-than-designed flow was small, approximately 10 percent, indicating that the probability of failure to provide adequate spray cooling to a dropped assembly when called on to function is 0.0016.

More than 300,000 fuel and target assemblies have been discharged without a failure of the discharge machine cooling-water system. The probability of melting an assembly due to failures of both the discharge machine drive mechanism and the cooling system has been estimated to be approximately 7×10^{-5} (Nomm, 1977). Improvement to the discharge machine drive and control system that have been or are being implemented will substantially reduce this probability (by one
AY-9 or two orders of magnitude).

By combining the above probabilities, the estimated probability of melting a fuel or target assembly during discharge is estimated to be less than 10^{-4} per reactor-year.

Reloading error leading to criticality. This type of accident has not occurred at Savannah River Plant.

The reloading error most likely to occur that would lead to a large reactivity increase involves removing a target assembly, failing to replace that assembly with a fresh target, and then removing an adjacent target assembly. The probability of criticality occurring from the removal of so much absorbing material depends on three factors: (1) the probability that the reloading error occurs somewhere in the reactor; (2) the fraction of reactor positions for which the reloading error could produce extreme reactivity changes; and (3) the probability that the reactivity effect could be large enough to achieve criticality. (No damage would occur if the reactor were just critical. The reactivity addition would have to be large enough to achieve significant supercriticality. But to be conservative, this analysis only considers the probability of achieving criticality to be more likely than that of achieving supercriticality. The probability of actual damage would be less than that discussed here.)

Each reactor area has a charge/discharge computer system that monitors for target vacancies, checks the validity of steps in the charge and discharge sequence, and imposes interlocks that require extraordinary actions to bypass key steps. Prior to the installation of the charge/discharge computer system, the frequency of a double target vacancy was estimated to be about 0.1 per reactor year. Specific charge analyses indicate that about 4×10^{-5} of the postulated double vacancies could result in sufficient reactivity changes to achieve criticality. Thus, without taking credit for protection provided by the charge/discharge computer system, the probability of a double target vacancy resulting in a criticality is estimated to be 4×10^{-6} per reactor-year (Church, 1983).

Protection provided by the charge/discharge computer system has not been evaluated explicitly but should reduce the probability of occurrence by at least a factor of 10 to a value less than 4×10^{-7} (Church, 1983). This is below the probability considered credible. Until the protection provided by the computer system is evaluated explicitly, this accident is considered to define the spectrum of credible events and risks along with the other three accidents discussed in this section.

<u>One-percent core melt due to a loss-of-coolant accident</u>. This type of accident has not occurred at Savannah River Plant. The results of a literature search on pipe breaks in highly pressurized systems (L-Reactor is not a highly pressurized system) indicate probabilities on the order of 3×10^{-5} per year for massive piping failures. The probability of a partial failure of the Emergency Cooling System has been estimated to be 3×10^{-2}. Thus, the probability of the accident occurring with only one operable ECS is less than 1×10^{-6} per reactor-year. (If two ECS systems are operable, there is no damage.) AY-9

The assembly flow rates are computed for these extreme conditions using methods that are normalized to the results of reactor experiments simulating loss-of-coolant-accident conditions. Based on these flow rates, the damage to the reactor core is computed as a function of preincident reactor power. A maximum upper limit is then set on reactor power such that the reactor damage will not exceed 1 percent in the event of a maximum-leak-rate, loss-of-coolant accident coupled with losses of two of the three ECS systems.

Thus, the probability of a loss-of-coolant accident occurring and causing 1-percent core melting is estimated not to exceed 10^{-6} per reactor-year (Church, 1983).

<u>Radiological consequences of reactor accidents</u>

This section describes the techniques used to calculate offsite doses that result from reactor accidents. Appendix G provides a more detailed (NRC, 1979; Pendergast, 1982a,b) description. The calculations are consistent with NRC guidelines for accident analysis. The methods discussed were used for analysis of all accidents, including the moderator spill and fuel melting accidents.

Three parameters are necessary to compute the maximum offsite dose. First, the radioactive source term must be specified, including the release rate and isotope type. Second, the transport of the isotope by the wind must be computed based on appropriate calculational models and meteorological data. Third, the external and internal doses to an individual assumed to be located at the plant boundary are computed based on a standard man, breathing rates, and several parameters related to absorption of energy from a particular isotope.

The release from the stack is assumed to propagate as a Gaussian plume over a 2-hour period, and the exposure of an individual is treated as a time-integrated calculation. Two-hour duration of the meteorology is assumed, and this implies the subject is irradiated for a 2-hour period. This is very conservative because measurements at the SRP site show that the probability of wind persistence for a 2-hour period is, for some directions, only about 20 percent.

The 2-hour irradiation period begins when radioactive material reaches the plant boundary. Both the noble gas and iodine source terms are assumed to have decayed during transport. Decay during the exposure is not included in the calculation.

The source term for iodine is the amount that would penetrate and desorb from the filters in the first 2 hours following the incident. The average iodine retention efficiency assumed for the carbon is that for carbon aged 19 months. This is intended to be typical of normal operation. Carbon beds are replaced on a staggered schedule, so some beds have relatively fresh carbon, some have carbon of intermediate age, and some have carbon approaching its service limit of 30 months.

The downwind concentration of iodine, tritium, and noble gases was calculated according to an integral technique using the computer code NRC145-2. This code was developed at Savannah River Plant and uses a Gaussian plume model based on NRC Regulatory Guide 1.145 (Pendergast, 1982a).

The meteorological data used in the dose calculations were collected from January 1975 through December 1979. The data were obtained at towers near P-, K-, and C-Reactors. Calculations for L-Reactor used data from the closest tower (K-Area). The meteorological data from each tower were averaged for 2-hour periods and sorted into 16 direction sectors, six wind speeds, and seven stability classes. (Stability classes were based on the deviation of the mean wind direction.)

Median meteorological conditions (50th percentile) were assumed in these calculations. Relative doses could be higher under more extreme meteorological conditions, as indicated in Figure 4-9.

Corrections for the topography and jet rise of the released plume are also applied.

Interpolation between 2-hour doses and annual average doses was used to obtain the dose for an extended exposure period of 120 hours, using a method recommended in the NRC Guidelines, incorporated into NRC145-2 (Pendergast, 1982a), and independently verified.

The thyroid dose and the whole-body dose are composed of an inhalation component from iodine, tritium, and a shine component from the gamma emission of the noble gases. The inhalation component was computed by multiplying the isotopic relative concentration by the source strength and dose conversion factors. The shine component integrated the gamma dose from the entire (finite) radioactive plume.

The moderator spill accident considers the tritium dose when the moderator is displaced from the reactor (e.g., due to actuation of the Emergency Cooling

System). The calculation assumes a release of 0.15 megacurie (3 percent of the assumed 5 megacuries tritium inventory in the moderator) over a 2-hour period. The calculated dose to an individual at the plant boundary is shown in Table 4-20.

Table 4-20. Calculated radiation dose to a person at the SRP site boundary following four specific accidents (median meteorology)

Accident	Calculated dose (rem)		
	Whole-body (2 hr)[a]	Thyroid (2 hr)	Thyroid (120 hr)
D_2O spill	0.006	-	-
Discharge mishap (one fuel assembly melts)	0.003	0.004	0.01
Reloading error (3% core damage)	0.39	0.51	1.5
LOCA (1% core damage)	0.13	0.17	0.50

[a]The 2-hour whole-body dose is essentially the same as the accident-duration whole-body dose.

The discharge mishap accident assumes that an irradiated fuel assembly, having decayed for 14 hours after shutdown, melts while being discharged. The calculated dose to an individual at the plant boundary is shown in Table 4-20.

As discussed above, calculations indicate that the maximum hazard for a reloading accident would involve less than 3 percent of the core inventory of fission products. The fission product content of the core is assumed to be the equilibrium concentration that would be obtained at full power. Table 4-20 lists the calculated dose to an individual at the plant boundary.

The 1-percent core-melt accident assumes that a massive double-ended pipe break occurs. Thus, 1 percent of core fission product inventory as well as heavy-water coolant is released. Table 4-20 lists the calculated dose to an individual at the plant boundary.

In summary, these offsite doses from postulated accidents were calculated in accordance with accepted methods and assumptions. Appendix G describes offsite doses from particulates. These doses do not exceed DOE radiation protection standards (DOE 5480.1a.1, Chapter 11) for normal operation.

BF-9 Releases to ground water and surface water

No significant releases to ground water or surface water would be expected from reactor accidents. In the event of a loss-of-primary-coolant or a loss-of-pumping accident, the reactor scrams and the emergency cooling system forces as much as 53,000 liters of water per minute into the reactor to remove decay heat from the core. This water displaces the heavy water, then continues to flow through the reactor.

Overflow from the reactor is pumped to one of two holding tanks that are part of the confinement system. The first tank has a capacity of 225,000 liters and will retain essentially all of the displaced heavy water and its associated tritium. When this tank is full, any subsequent flow bypasses the tank at an upstream overflow point and flows to a 1.9-million-liter tank located in a 190-million-liter earthen basin.

If ECS flow has to continue until the larger tank is full (e.g., for a large primary coolant leak that cannot be isolated), subsequent flow bypasses the tank at an upstream overflow point and enters the earthen basin.

Air that is displaced as the tanks fill with water passes through vent lines and joins the ventilation air that is exhausted through the confinement filters to the 61-meter stack.

If core damage occurs during these severe accidents (less than 1-percent melting is calculated to occur for a large pipe break with only one of three ECS systems operable), fission products would be released to the emergency coolant flowing through the reactor. Any melting would occur in the first minutes of an accident while the decay heat is high and stable ECS flow is being established.

Volatile fission products would be released into the confinement ventilation system; the remainder of the fission products would be retained in the two tanks, which hold a total of more than 10 times the volume of the primary coolant. Any water flowing to the earthen basin after the tanks are full would have passed through a well-cooled, well-flushed core and would be essentially free of radioactivity. For the highly unlikely case of delayed melting after the tanks are full, the noble gases and radioiodine could be carried to the 190-million-liter basin where they could be released directly to the atmosphere. In this case, the iodine would cause increased offsite thyroid doses. Because of the extremely low probability of delayed core damage, no additional dose risk is attributed to this accident.

Risk considerations

The foregoing descriptions have dealt with both the frequency (or likelihood of occurrence) of accidents and their offsite dose impacts (or consequences). Because the ranges of both factors might be quite broad, it is useful to combine them to obtain average measures of environmental risk. Such averages can be particularly instructive as an aid to the comparison of radiological risks associated with accident releases and with natural sources of radiation.

A common way in which this combination of factors is used to estimate risk is to multiply the probabilities by the consequences. The resultant risk is then expressed as a magnitude of consequences expected per unit of time. Table 4-21 lists the estimated whole-body risks associated with the four postulated accidents described in this section. These risks were calculated by multiplying the calculated whole-body doses in Table 4-20 by the corresponding accident probabilities in Table 4-22; they range from 10^{-4} to 10^{-3} millirem per reactor-year. All risk values are much less than the risk that would be associated with a natural radiation dose of 93 millirem per year.

Table 4-21. Risk evaluation of postulated serious accidents

Accident	Consequence[a] (mrem)	Probability (y^{-1}) per reactor-year	Expected whole-body risk (mrem/reactor-year)
Moderator spill	6	10^{-4}	6×10^{-4}
Discharge mishap	3	10^{-4}	3×10^{-4}
Reloading error	390	4.0×10^{-7}	1.6×10^{-4}
LOCA, resulting in 1% core melt	130	10^{-6}	1.3×10^{-4}

[a]The 2-hour whole-body dose is essentially the same as the accident-duration whole-body dose.

4.2.1.5 Assessment of severe hypothetical accidents

Any accident that results in damage greater than the maximum calculated for the accidents described above (3-percent core melt) is highly improbable. As discussed in more detail in Appendix G and in the Safety Analysis Report (Du Pont, 1983a), analyses of hypothetical SRP reactor accidents indicate that the probability of an accident of a higher consequence than a 3-percent core melt is extremely low. The estimated probability of accident sequences that would result in melting as much as 100 percent of the reactor core is on the order of
TC 10^{-8} per reactor-year. For this analysis, the Airborne Activity Confinement System is expected to continue to function properly because it is already online before the accident, includes redundant primary components and diverse backup
TC power supplies, and has a high tolerance to severe accidents (Du Pont, 1983a). As an added safety measure, a Confinement Heat Removal System has been installed to reduce the possibility of confinement failure in the extremely unlikely event of a full core-melt accident. However, to assess the consequences of core melting for a highly improbable sequence of events, a 10-percent melt accident is postulated. Based on the discussion for the accidents with lesser consequences, the probability of a 10-percent core melt would be between 10^{-6} and 10^{-8} per reactor-year.

TC To analyze the consequences of accidents having very low probability, an evaluation independent of the SAR (Du Pont, 1983a) was performed using the computer model, CRAC2, employed by NRC to evaluate core-melt accident consequences in its Environmental Impact Statements (NUREG/CR-2901). This model considers the probability of occurrence of each of 29 meteorological conditions based on site data, population distributions as far as 800 kilometers from the site, and a number of options for mitigation of consequences that were not exercised in this evaluation. The model calculates exposures to individuals and populations from (1) direct radiation from the passing plume and material deposited on the ground, (2) inhalation, and (3) consumption of contaminated foods and milk. Finally, the model produces consequence-probability distribution curves (called complementary cumulative distribution functions, or CCDFs) for various doses, for prompt and delayed fatalities, and for economic costs (see Appendix G).

An examination of the results of these calculations must recognize that there are a number of differences between the CRAC2 methodology and the method that has been normalized to SRP conditions to arrive at the doses presented in Section 4.2.1.4. For example, mean doses determined by CRAC2 are not directly comparable to the median (or fiftieth percentile) meteorological condition employed for the doses in Section 4.2.1.4. Also, CRAC2 dose pathways include small doses from ground-deposited material, food pathways, and inhalation of resuspended radionuclides not considered in the other dose values. Other differences exist in the net effectiveness assumed for iodine retention by the charcoal filters, the duration of the releases, site boundary distances, meteorological data base, and the population data year chosen. Despite these differences in methodology and assumptions, the results are in good agreement.

Dose and health impacts

Calculations using the CRAC2 code show that, for the hypothetical 10-percent core-melt accident, there are no cases of early fatalities, no cases where the whole-body dose exceeds 25 rem, and no cases where the thyroid dose exceeds 300 rem (10 CFR 100 siting criteria). The mean value for the site boundary whole-body dose is 0.35 rem and the expected peak value (i.e., for the most improbable meteorological condition sampled) is 1.7 rem. The mean value for the site boundary thyroid dose is 1.7 rem with a peak value of 11.7 rem.

Figure 4-10 displays the calculated CCDF for latent cancer fatalities. The mean number of cancer fatalities (including thyroid cancers) is 2.4 and the peak is 20 with a conditional probability (i.e., assuming the accident has occurred) of 1.4×10^{-4} per reactor-year. (Excluding thyroid cancers, the mean number of latent cancer fatalities is 1.0 and the peak number is 15.) When the probability of a 10-percent core-melt accident (10^{-6} to 10^{-8}) is taken into account, the mean number of latent fatalities is, conservatively, 2.4×10^{-6} per reactor-year or an average of one death per 400,000 reactor-years of operation.

Figure 4-11 displays the CCDFs for total population whole-body exposure in person-rem, that is, the conditional probability that the total population exposure will equal or exceed the values given. The peak population exposure is 2.4×10^5 person-rem with a conditional probability of 1.1×10^{-4} and the mean value is 1.6×10^4 person-rem for the population within 800 kilometers of the reactor site, and 7.7×10^3 person-rem for the population within 80 kilometers of the reactor site. Again, if the probability of an accident with a 10-percent core melt (10^{-6} to 10^{-8}) is taken into account, the mean value for total exposure for the population within 80 kilometers is, conservatively, 7.7×10^{-3} person-rem per reactor-year. For perspective, this can be compared to a whole-body dose from natural background radiation of 8×10^4 person-rem per year for the population in question.

Economic and social impacts

The offsite economic impact of a reactor accident is calculated as a probability distribution for the cost of offsite mitigating actions. The factors contributing to these estimated costs include the following:

- The value of crops contaminated and condemned

- The value of milk contaminated and condemned

- Costs of decontamination of property where practical
- Indirect costs due to loss of use of property and incomes derived therefrom

The last cost would derive from the necessity for interdiction to prevent the use of property (i.e., farm crops, etc.) until it is either free of contamination or can be economically decontaminated.

The mean offsite economic risk from an accident where 10 percent of the core melts is \$73,000 and the peak cost is \$1.7 x 10^6 at a conditional probability of 2.4 x 10^{-4}. For comparison, the cost of property damage due to automobile accidents for the area of a circle with a radius of 80 kilometers is \$1.3 x 10^7 per year and the property damage due to fires for the same area is \$5.5 x 10^6 per year.

Table 4-22 summarizes all the consequences from a postulated 10-percent core-melt accident.

Table 4-22. Consequences from a postulated accident resulting in 10-percent core melt[a]

Consequence	Mean value	Peak value
Early fatalities	0	0
People with whole-body dose of 25 rem	0	0
People with thyroid dose of 300 rem	0	0
Latent-cancer fatalities (excluding thyroid)	1.0	14.9
Thyroid-cancer fatalities	1.4	5.2
Site boundary whole-body dose (rem)	0.35	1.7
Site boundary thyroid dose (rem)	1.7	11.7
Population whole-body dose (person-rem) (population to 80 kilometers)	7.7 x 10^3	2.4 x 10^5
Population whole body dose (person-rem) (population to 800 kilometers)	1.6 x 10^4	2.4 x 10^5
Population thyroid dose (person-rem) (population to 80 kilometers)	8.6 x 10^4	3.6 x 10^5
Population thyroid dose (person-rem) (population to 800 kilometers)	1.0 x 10^5	3.8 x 10^5

[a]Hypothetical 10-percent core melt as calculated with CRAC2 code.
TC| The probability of a 10-percent core melt is estimated to be less than 10^{-6}.

Table 4-23 shows average values of risk associated with population dose, early fatalities, latent fatalities, and costs for early evacuation and other protective actions, which have been calculated for a 10-percent core melt. These average values are obtained by summing the probabilities multiplied by the consequences over the entire range of the distributions. Because the probabilities are on a per-reactor-year basis, the averages shown are also on a per-reactor-year basis.

Table 4-23. Average values of environmental risks due to a 10-percent core melt, per reactor-year[a]

Offsite risk	Value
Population exposure	
Person-rem within 80 kilometers	7.7×10^{-3}
Person-rem total	1.6×10^{-2}
Early fatalities	0.0
Latent cancer fatalities	
All organs excluding thyroid	1.0×10^{-6}
Thyroid only	1.4×10^{-6}
Cost (dollars) of protective actions and decontamination	7.3×10^{4}

[a]Hypothetical 10-percent core melt as calculated by the CRAC2 code. The probability of a 10-percent core melt is estimated to be less than 10^{-6}.

4.2.1.6 Total risk from all postulated reactor accidents

To provide a perspective of the overall reactor accident risk on the Savannah River Plant and of L-Reactor operation, Figure 4-12 shows preliminary total probability curves that present the annual probability of a resident living at the SRP site boundary receiving more than a certain dose from postulated accidents (see Section G.5.7.3). These results are based on accident analyses presented in the Safety Analysis Report and a supporting document (Du Pont, 1983a; Church, 1983), including less severe accidents at the high end of the probability spectrum and an assumed hypothetical 100-percent core melt at the upper bound of the consequences spectrum (see also Section G.5.7.3). Six different accident initiators were considered. For all the accidents, the most probable outcome would be no reactor damage. For the six accidents, only 11 postulated, but highly improbable, sequences resulted in significant amounts of reactor core damage (ranging from 1 percent to 100 percent). For the postulated 100-percent core-damage accidents (sequences 2, 3, 4, and 6 below), Figure 4-12 also reflects the failure of the Confinement Heat Removal System. These accident sequences were as follows: TC TC

1. A loss-of-coolant accident with only one operable ECS.
2. A loss-of-coolant accident with a total failure of the ECS. EN-27
3. The withdrawal of a single control rod or a gang of control rods with a failure of both the safety-rod scram and the ABS-SC.
4. Loss of coolant to a single target assembly with a failure of both the safety-rod scram and the ABS-SC.
5. A loss-of-pumping accident with only one operable ECS.
6. A loss-of-pumping accident with a total failure of the ECS.

7. A reloading error during charge/discharge operations making the reactor supercritical.

8-11. Extended total loss of offsite (commercial) power together with extended loss of onsite generating capability. This sequence would affect all reactors and is postulated to result in core damage to 1, 2, 3, or 4 reactors.

The computed offsite doses for the loss-of-coolant accident with 1 percent core damage and the reloading error with 3-percent core damage are listed in Table 4-20 for median meteorology (conditions for which the more severe meteorological conditions are not exceeded 50 percent of the time). The relative doses for other meteorological frequencies are shown in Figure 4-9. Doses for postulated core damage greater than 1 percent would be proportional to the dose for 1-percent damage.

The probability of occurrence of an accident sequence was combined with the data for meteorological probability versus offsite dose for each of the above 11 sequences. Then, for a given dose rate, the occurrence probabilities were combined to obtain an overall probability per reactor-year of exceeding a given dose. This overall dose probability curve is shown in Figure 4-12. The results are consistent with (1) the decreasing frequency of meteorological conditions EN-27
that give higher doses for any accident (Figure 4-9), and (2) the extremely low probability of accidents occurring with core damage exceeding 3 percent.

The implementation of reactor safety programs has reduced the probability of occurrence of accidents to extremely low levels. Figure 4-12 indicates that the probability of exceeding the Nuclear Regulatory Commission site whole-body dose criteria for commercial power reactors (10 CFR 100) of 25 rem at the site boundary in accident situations is extremely low (less than 10^{-7} per year), even in the most severe hypothetical accidents.

The traditional approach to SRP reactor safety analysis addressed the consequences for "worst-case credible" (and even some "noncredible") accidents based on the single-failure criterion. This criterion assumes that the initial accident is compounded by the failure of the single-most-important active component designed to mitigate the accident. (An active component is one that must change its state to perform its duty; e.g., a valve must be realigned.) The initiation of the accident and the failure of the component were considered without regard to the actual probability of their occurrence. Results from the preliminary risk evaluation of the accident sequences discussed above support earlier evaluations made for worst-case scenarios using single-failure criteria, which concluded that there is negligible risk to public health and safety.

4.2.2 Non-nuclear hazards and natural phenomena

4.2.2.1 Toxic-gas release

During prior reactor operations, the effects of toxic-gas releases were analyzed, and provisions were made for shutdown, building evacuation, and remote control of coolant flow pumps and valves. The two toxic gases considered were the chlorine used to prevent biofouling of reactor heat exchangers and the hydrogen sulfide used in the heavy-water production area. Two recent changes in plant operation have essentially eliminated any hazards from these gases:

1. L-Reactor would use sodium hypochlorite rather than chlorine as the cooling-water biocide. Sodium hypochlorite presents no toxic-gas health hazard to reactor operation and would provide the same biofouling inhibition as chlorine.

2. Heavy-water production at the Savannah River Plant has stopped. The large quantities of hydrogen-sulfide gas stored in the heavy-water production area have been removed.

4.2.2.2 Fire

The presence of flammable materials in the reactor building is strictly controlled, so the probability of a large fire is low. Because of redundancies in shutdown, a fire (e.g., in an electrical cable tray) will not prevent a safe shutdown. Analyses performed (Du Pont, 1983a) for L-Reactor startup did not find any credible fire hazard that would result in a release of radioactivity. The only fire-related incident deemed credible was the possibility of extended downtime and repair costs, but no specific cause for such a fire was identified.

In addition to normal operating personnel who are instructed in basic fire fighting, a fully trained and equipped fire department is maintained at Savannah River Plant.

A large cleared area surrounding the reactor building protects against hazards from a forest fire. Smoke from a forest fire could require temporary evacuation of L-Reactor. However, normal and emergency facilities are provided to maintain safe conditions, and the reactor could also be shut down and maintained in a safe shutdown condition from the remote control station.

4.2.2.3 Earthquakes

As noted in Section 3.3.2, there are no known capable faults within 300 kilometers of the L-Reactor site, except perhaps the geophysically inferred faults in the meizoseismal area of the 1886 Charleston earthquake (Du Pont, 1980; Georgia Power Company, 1982). No reservoir-induced seismicity is associated with Par Pond, which is located about 6.5 kilometers northeast of L-Reactor.

Probabilistic and deterministic analyses, commensurate with the criteria used by the NRC in 10 CFR 100, have determined that the maximum seismic hazard at the Savannah River Plant is due to a Modified Mercalli Intensity MMI; Langley and Marter, 1973) of VII (magnitude 5.0 to 5.5) earthquake in the immediate vicinity of Savannah River Plant or a postulated MMI = X (magnitude 6.6) earthquake near Bowman, South Carolina, 95 kilometers from Savannah River Plant. In both cases, the expected site MMI = VII corresponds to a peak horizontal free field acceleration of about 0.10g (Du Pont, 1982a). A design-basis earthquake acceleration of 0.20g has been established for design and analysis of key seismic-resistant buildings, systems, and components at Savannah River Plant. This design acceleration is predicted to be exceeded only once in 5000 years (Du Pont, 1982a).

Studies performed by Rutledge (1976) and D'Appolonia (Du Pont, 1980) show that earthquake ($\leq$0.20g)-induced liquefaction is not a potential problem for L-Reactor and other SRP facilities located on the Aiken Plateau (cf., Langley and Marter, 1973, and Figure F-1).

The foundation investigations for L-Reactor were performed by the U.S. Army Corps of Engineers (COE, 1952a). At their recommendation, a soil grouting program was undertaken to improve subsurface conditions (COE, 1952b). A number of earthquake-engineering investigations have been performed to establish earthquake-design criteria and to recommend modifications to component design (e.g., Du Pont, 1968; List, 1969; Rutledge, 1976; Geotechnical Engineers, Inc., 1979; URS/JAB, 1982a,b,c).

The reactor buildings are heavy, blast-resistant, concrete structures. Several earthquake-engineering improvements have been made at P-, C-, and K-Reactors to meet the seismic criteria for a design basis earthquake of 0.20g. These improvements were also made in the L-Reactor upgrade and include the following:

- Providing additional seismic bracing on the actuator tower to reduce its dynamic response to earthquake excitation
- Strengthening the 61-meter building exhaust stack
- Improving the lateral support for the emergency cooling-system piping and the supplementary safety system (neutron poison injection system) piping
- Improving the anchors on the 12 track-mounted process heat exchangers

An earthquake monitoring system will automatically alarm at 0.002g and shut down the reactor when the earthquake excitation reaches 0.02g (one-tenth the design-basis value). In more than 28 years of reactor operation there has never been a seismic alarm.

4.2.2.4 Tornado and hurricane effects

The SRP site lies within tornado risk region B (Twisdale and Dunn, 1981) with an occurrence rate of about 2.69×10^{-4} per square kilometer per year corrected for unreported tornadoes. Based on this study and on work by Reinhold and Ellingwood (1982), the probabilities of a tornado striking a point at Savannah River Plant are calculated for the midpoint characteristics of the Fujita-tornado intensity scale (F-scale); the results are presented in Table 4-24. In addition, this table provides the probability of striking a building as large as L-Reactor at the SRP site. Risks are extremely low. TC

Hurricanes that occur along the South Carolina coast generally will not subject the Savannah River Plant to winds in the whole-gale to hurricane range

TC

Table 4-24. Annual probabilities of a tornado strike at L-Reactor for midpoints of the Fujita tornado intensity scale

Fujita intensity scale	Wind speed[a] (m/sec)	Annual probability of a tornado strike at L-Reactor[b]
F-0	16.1	7.79×10^{-4}
F-1	41.4	3.52×10^{-4}
F-2	60.4	1.65×10^{-4}
F-3	81.4	5.35×10^{-5}
F-4	104.4	1.58×10^{-5}
F-5	129.4	2.61×10^{-6}
F-6	156.2	3.01×10^{-7}

[a]Wind speeds are reported for the midpoints of the Fujita tornado intensity categories.

[b]Based on an occurrence rate of 2.69×10^{-4} tornados per square kilometer per year (Reinhold and Ellingwood, 1982, Tables 16 and 17), and an L-Reactor building width of 170 meters.

because Savannah River Plant is approximately 160 kilometers inland, and the high winds associated with hurricanes tend to diminish as the storms move over land. Winds of 33.5 meters per second were measured once by anemometers mounted at the 61-meter level of the WJBF-TV tower during the history of Savannah River Plant, as Hurricane Gracie passed north of the plant site in September 1959. At Augusta, Georgia, the fastest 1-minute wind speed for the 1950-1978 period of record was 37.1 meters per second (corrected to an anemometer height of 10 meters). The return periods for 1-minute wind speeds at Augusta are reported in Table 4-25.

Table 4-25. Return of 1-minute wind speeds at Augusta, Georgia

Return period (years)	Wind speed (m/sec)
100	37.1
1,000	46.9
10,000	56.8
100,000	66.2

TC

The L-Reactor building is a concrete structure that is blast-resistant to a pressure of about 50,000 pascals. Its weakest structural area, the disassembly area, can withstand a tornado-induced pressure drop of 20,700 pascals (Yau and Zeh, 1976), twice that created by an intensity F-5 tornado (a very low probability event; see Table 4-24).

The 61-meter-tall ventilation exhaust stack at L-Reactor is designed to withstand a 1-in-10,000 year event (see Table 4-25) with winds of 56 meters per second. However, if the stack should fall, it would not strike a portion of the reactor that would impair the ability to shut down the reactor or maintain cooling capabilities.

The resistance of the L-Reactor building to wind-driven missiles was analyzed by Yau and Zeh (1976) as part of a study to determine the tornado resistance of the reactor building. The greatest penetration of the concrete reactor building was calculated to be caused by a 30-centimeter steel pipe; less than 40 percent of the wall thickness of the disassembly area wall was calculated to be penetrated by the pipe.

Because the disassembly area is structurally the weakest part of the reactor building, the rest of the building was also deemed safe from penetration by the postulated missiles. The probability of tornado missiles passing through exterior doors, ducts, vents, or other openings that are not tornado resistant is negligibly small.

Damage to the 61-meter-tall stack, confinement system filter compartments, and other parts of the building that are not resistant to tornados would not cause, directly or indirectly, a reactor accident. A tornado strike causing damage to the filter compartments or the stack after an independently caused reactor accident would increase offsite dose effects. Such multiple-series accidents are not considered in this analysis because of the extremely low probability of a tornado striking the reactor immediately following a reactor accident.

Emergency power capabilities at L-Reactor are sufficient to maintain the reactor in a safe shutdown condition if outside power is lost during a severe weather disturbance.

4.2.2.5 Floods

As noted in Section 3.4.1, L-Reactor (floor elevation of 76.5 meters) is situated well above (1) the maximum historical flood stage of 36 meters and (2) the flood stage of 43.6 meters calculated to result from the domino failure of Savannah River dams above the SRP. Flooding of these magnitudes could cause the loss of the river pumphouses supplying cooling water, and of external electrical power. However, onsite storage of cooling water (9.5×10^4 cubic meters) is, with partial recirculation, adequate to remove heat during shutdown, and onsite emergency power generation would maintain the reactor in a safe shutdown condition.

Because of the geographic location of the site, the formation of significant amounts of ice on streams and rivers occurs rarely. A review of Augusta, Georgia, newspaper accounts dating back to approximately 1800 indicates that the formation of ice jams on the Savannah River occurred in 1827 and 1886. Neither event resulted in reported flooding (Du Pont, 1980).

The L-Area is not subject to local flooding. Pen Branch to the west and north, and Steel Creek to the east and south provide adequate drainage. Opposite L-Reactor these streams are at least 15 meters below the reactor floor elevation under normal flow conditions.

4.3 TRANSPORTATION

4.3.1 Onsite and offsite shipments

Onsite

The proposed restart of L-Reactor would increase the total number of onsite shipments by an amount typical of the individual reactor areas now operating. Rail shipments of irradiated fuel from the reactor to the separations plants could be made with existing casks and equipment using current rail crews. Truck shipments involving unirradiated reactor fuel, deionizer casks, and wastes could also be made with existing equipment using the SRP traffic and transportation (T&T) crews currently assigned to these tasks. Higher volume shipments, such as scrap metal, waste dumpsters, and D_2O drums, would require purchase of additional equipment and a modest increase in T&T crews. Also, the operation of L-Area would require about the same number of nonradioactive shipments by T&T and vendor trucks as the other individual reactor areas. No significant impact on SRP transportation systems is expected from the operation of L-Area.

Shipments on the SRP rail system would include the following:

1. Empty casks to transport reactor fuel elements.

2. Intact irradiated fuel in 70-ton casks (CD casks) on flatbed railcars to 200-F or 200-H areas.

3. Any irradiated fuel with cladding defects in a special containment device ("harp") within a 55-ton failed fuel element cask to a 200-Area.

4. Occasional containers of helium or Polybor or other nonradioactive materials.

Onsite truck shipments for L-Area would include the following:

1. Unirradiated fuel in steel shipping boxes and other reactor lattice components from the 300-M area.

2. Irradiated lithium-aluminum control rods and blanket assemblies in a 45-ton cask on a flatbed trailer from the L-Area disassembly basin to 200-H area.

3. Irradiated scrap metal in a 15-ton cask or replacement cask from the L-Area disassembly basin to the SRP burial ground (about 80 shipments annually).