APPENDIX G

ENVIRONMENTAL IMPACT OF POSTULATED PLANT ACCIDENTS

This appendix provides supplementary information for Section 4.2.1, Reactor Accidents. This appendix describes (1) the general characteristics of accidents; (2) the actual experience with SRP reactor incidents; (3) safety features of the L-Reactor and of the site that act to mitigate the consequences of accidents; (4) all postulated transients considered for the safety evaluation of L-Reactor; (5) radiological consequences of four hypothetical accidents that cover a spectrum of significant events postulated to release radioactivity above normal operating limits; and (6) input considerations for a CRAC2 analysis of a hypothetical 10-percent core-melt accident (Section 4.2.1.5).

G.1 GENERAL CHARACTERISTICS OF ACCIDENTS

The term "accident," as used in this section, refers to any postulated event that could result in a release of radioactive materials into the environment. The predominant focus is on events that can lead to releases substantially in excess of permissible limits for normal operation.

Several features combine to reduce the risk associated with accidents at nuclear plants. Safety features in the design, construction, and operation, comprising the first line of defense, are devoted to the prevention of the release of radioactive materials from their normal places of confinement within the plant. Also, a number of additional lines of defense are designed to mitigate the consequences of failures in the first line. The most important mitigative features for L-Reactor are described in Section G.3.1. Detailed descriptions of these features may be found in the Safety Analysis Report (SAR) (Du Pont, 1983).

The L-Reactor is designed to produce plutonium by the absorption of neutrons in uranium. The reactor uses heavy water (D_2O) as a moderator and as the primary coolant to remove heat generated by the nuclear fission process. L-Reactor operates at significantly lower temperatures and pressures than light-water commercial nuclear power plants designed for electric power generation. This feature in itself tends to reduce the consequences of many types of accidents. In addition, the absence of a turbine load eliminates a whole range of accidents possible with conventional nuclear power plants.

The transients considered for evaluation of L-Reactor safety are listed in Table G-1. The reactor will operate at a power limit that is determined separately for each charge and each fuel and target cycle so that for any anticipated transient, operation at or below the operating limit would prevent release of radioactivity to the environment. Major safety systems, listed in Table G-2, have been incorporated into the design and operation of the reactor to shut down the reactor and limit the release of radioactivity if necessary.

Four hypothetical accidents are evaluated that cover a spectrum of events postulated to release radioactivity. These four hypothetical accidents, which

Table G-2. Major safety systems

Reactor shutdown and safety systems	Engineered safety systems
1. Safety rods	1. Emergency cooling system (ECS)
2. Control rods	2. Water removal and storage
3. Scram instruments and alarms	3. Activity confinement system
4. Supplementary safety systems (SSS)	4. Confinement heat removal system
5. Automatic backup shutdown--safety computer (ABS-S/C)	5. Reactor room spray system
6. Automatic backup shutdown--gang temperature monitor (ABS-GTM)	6. Discharge assembly cooling

have never occurred at SRP, include (1) a total moderator spill, (2) a discharge mishap in which an irradiated assembly is dropped and melts; (3) a misloading accident during charge-discharge operations resulting in melting less than 3 percent of the reactor core; and (4) a loss-of-coolant accident resulting in the melting of 1 percent of the reactor core. No credible accident sequences FG-3 have been identified that will cause a reactor accident resulting in core damage greater than 3 percent.

The probabilities reported in this document are based on more than 115 reactor-years of operating experience at Savannah River Plant, conservative engineering judgment, and failure modes and effects analyses (Church, 1983). The probabilistic and risk assessment discussion contained in this document has been based in part on the methodology presented in the Reactor Safety Study (NRC, 1975). In addition, a probabilistic risk assessment (PRA) of the SRP reactors is being performed.

No accidents occurred during the previous operation of the L-Reactor that resulted in the release of radioactivity to the public above DOE standards for normal operations. Safety-system improvements made to other SRP reactors, as a result of years of operating experience, have reduced the probability of an accident. These improvements have also been made on the L-Reactor.

G.2 ACCIDENT EXPERIENCE AND OBSERVED IMPACTS

This section describes the actual experience with SRP reactors. No significant reactor accidents have occurred at the SRP in its 30 years of operation. The following sections describe reactivity addition, flow reduction, and other events that might have led to substantial release of radioactive material if the safety systems or automatic backup systems had failed to function properly.

G.2.1 Reactivity addition

G.2.1.1 Single control-rod withdrawal

An average of two to three unwanted control-rod movements (Jones, 1972) has occurred per reactor year since 1954, and half of the movements were withdrawals that resulted in the addition of reactivity. The incidents were caused by either personnel errors or control-rod drive system malfunctions. However, these events never caused damage to the fuel or release of radioactivity into the environment, because an unwanted rod motion was usually stopped and rod position corrected immediately after the unwanted movement was recognized. Only about 1 percent of these events persisted long enough to actuate the control-rod reversal system. Safety rod scram action has never been required for inadvertent control-rod action.

TC

G.2.1.2 Partial control-rod insertion

Approximately half of the two to three control-rod movements that have occurred per reactor year since 1954 were applicable to partial control rods. Fewer than half of these unwanted partial control-rod movements were insertions that resulted in an addition of reactivity. However, these events never caused damage to the fuel or any release of radioactivity to the environment.

G.2.1.3 Gang-rod withdrawal at full power

No unwanted continuous gang-rod withdrawal has occurred at SRP. There were cases when the control computer attempted to raise power because of an erroneous input signal (Jones, 1972). Such incidents occurred at the rate of about 0.34 per reactor year. In one such incident in 1976, a spurious signal withdrew Gang I rods 0.2 foot in 15 seconds. The withdrawal by the control computer is not continuous and is terminated when the temperature signal reaches the operating limit. In all of these incidents, no damage to the fuel and no radioactivity release has occurred.

TC

G.2.1.4 Gang-rod withdrawal at low power

No unwanted continuous gang-rod withdrawal has occurred during low-power operation at the Savannah River Plant.

G.2.1.5 Control-rod melting

No control-rod melting has ever occurred at SRP. There have been several cases where the control-rod housing was not seated which reduced the cooling of the rods (Du Pont, 1983), but no rods were damaged.

G.2.1.6 Fuel assembly melting

Fuel melting has never occurred in the SRP reactors. During irradiation of the Californium-I high-flux charge in 1969 and early 1970, several fuel assemblies experienced cladding failures that resulted in releases of activity to the moderator (Du Pont, 1983). An estimated 40,000 curies of fission products entered the moderator and were subsequently removed by the moderator purification system. The failures were caused by improper allowance for assembly rib effects in heat transfer calculations. A small amount of noble gases was released into the blanket gas and subsequently discharged to the atmosphere. The release created no undue safety hazard. Noble gas monitors were installed in each operating reactor in 1972; any releases would now be recorded.

G.2.1.7 Target-assembly melting

No target-assembly has melted at SRP. While reductions in assembly coolant flow have been observed, all such reductions have been slow enough to enable shutting down the reactor without melting the assembly.

G.2.1.8 Fuel-reloading error

No reloading errors have occurred that have caused significant approach to criticality. One misloaded assembly was detected and corrected before reactor operations began.

G.2.2 Flow reduction

G.2.2.1 Loss of D_2O coolant pumps

The abrupt and total loss of offsite (commercial) a.c. power has occurred only four times in the history of Savannah River Plant, the longest being 38 minutes in duration. There are 11 onsite generators that normally supply about half of the electrical power to the 115-kilovolt grid. The complete loss of all 11 onsite generators has never occurred. Loss of a.c. power to the D_2O pump motors has been experienced at Savannah River Plant (Du Pont, 1983). The protective systems, including the independent backup d.c. motors, prevented any potentially damaging accidents.

A project currently underway will provide automatic load shedding following a loss of offsite power. This will prevent the resultant loss of the onsite generators so that power will continue to be supplied to the 115-kilovolt grid and to associated vital equipment.

G.2.2.2 Loss of H_2O pumps

Loss of all a.c. power to the H_2O pump motors has never been experienced at Savannah River Plant (Du Pont, 1983); however, a partial loss has occurred. The protective systems, including gravity flow backup cooling, prevented potentially damaging effects.

G.2.2.3 Pump shaft break

A drive shaft break between the D_2O pump impeller and the flywheel has never occurred at SRP. D_2O pump shafts are inspected during periodic overhaul of the pumps.

G.2.2.4 Rotovalve closure

Rotovalves are installed in the six external loops of D_2O circulation system between each of the 12 heat exchangers and the reactor plenum. Spontaneous closure of rotovalves has occurred on several occasions (Du Pont, 1983). On one occasion both rotovalves in a single system closed simultaneously at full power. However, a closure involving more than one external loop has not occurred, nor has any significant loss of D_2O circulation occurred due to rotovalve closures. No core damage or release of radioactive material occurred in any of the above incidents.

G.2.2.5 Flow reduction in a single assembly

The gradual reduction in flow could occur to a single coolant channel of a fuel or target assembly caused by cladding failure. Such failures accompanied by flow reduction have occurred at Savannah River Plant: five target failures in the last 3 years of operation of three reactors. The protective system was adequate to mitigate the consequences of this accident and prevent fuel melting and the release of radioactivity.

G.2.2.6 Loss of control-rod cooling

Control-rod cooling is accomplished by D_2O upflow through the septifoil (control-rod housing) from a header supplied by lines from the heat exchangers. The header pressure (and therefore flow) decreased on rare occasions, and the header pressure scram circuit operated properly to shut down the reactor. There were two cases of septifoils being unseated for long periods of time resulting in boiling of the coolant in the septifoils (Du Pont, 1983). Even then the control rods were not damaged and no radioactivity was released.

G.2.2.7 Loss of blanket gas pressure

Slow leaks of blanket gas have occurred without damage to the reactor or release of radioactivity. No rapid drop in blanket gas pressure has ever occurred.

G.2.2.8 Loss-of-coolant accident

No loss-of-coolant accident (LOCA) has ever occurred at Savannah River Plant. Furthermore, no fuel melting is anticipated in any credible LOCA. Small leaks from seals, flanges, and valves occasionally occur. The D_2O makeup system can replace D_2O at rates up to 15 liters per minute, and it is planned to increase these capabilities to 75 liters per minute. Most of the leak rates experienced at Savannah River Plant have been less than 2 liters per minute and only two leaks have approached 75 liters per minute (Joseph et al., 1970; Nomm, 1983).

G.2.2.9 Loss of D_2O circulation

Loss of a.c. power to the D_2O pump motors has occurred in the past (Du Pont, 1983). A complete loss of D_2O circulation has never occurred due to the backup d.c. motors operated by the independent diesel generator for each system.

G.2.2.10 Loss of cooling during and after assembly discharge

The discharge machine cooling systems have always worked when required. In about 300,000 assembly discharge operations, there have been instances in which the discharge operation was interrupted and emergency cooling was required. No fission products have been released because of failure of the cooling system during assembly discharge operations. In 1969, 100,000 curies of antimony and tellurium isotopes were released to the reactor building of which 0.003 curies were released to the environment, when an antimony-beryllium source rod melted while being held in air (Olliff, 1970; Brown, 1971; AEC, 1973). This accident was the result of administrative error; appropriate procedural controls have been implemented. This was the only time that the confinement system was required to function at SRP. No irradiated assembly has been dropped at SRP during discharge operations.

G.2.3 D_2O moderator spill

A sizable spillage of D_2O moderator occurred once during the early stages of operation. In July 1954, over a 12-hour period, an estimated 45,000 liters TC

TC of D_2O moderator overflowed seal leakage collection pots in the motor room in L-Area because two valves inadvertently were left open. An estimated 38,000 liters were recovered from sumps in the building. The moderator lost contained insignificant amounts of tritium because the reactor had achieved initial criticality only 15 days before the spill. In more recent history, spills of 380 to 3800 liters have occurred at a rate of about once per year.

G.2.4 Summary

The evidence of accident frequency is a useful indicator of future probabilities. As shown in the preceding sections, there have been no significant reactor accidents at SRP.

G.3 MITIGATION OF ACCIDENT SEQUENCES

A summary of safety features of the L-Reactor and of the SRP site that act to mitigate the consequences of accidents are provided in the following subsections.

G.3.1 Design features

L-Reactor is essentially identical to the other SRP reactors currently in operation. Each unit contains features designed to prevent accidental release of fission products from the fuel and targets and to lessen the consequences should such a release occur. These accident-preventive and mitigative features are referred to as shutdown systems, engineered safety systems, support systems, and a unique reactor power limit system. To establish design and operating specifications for L-Reactor, postulated events referred to as anticipated transients and accidents are analyzed.

Ward et al. (1980) studied the effects of neutron irradiation on the stainless-steel SRP reactor vessels and concluded that the vessels have experienced no significant deleterious effects. Furthermore, no deleterious metallurgical effects are expected in the future because neutron fluence has been accumulating very slowly since operations with lithium-blanketed charges began in 1968. At the temperatures and neutron fluences experienced by SRP reactors, yield strength and tensile strength increase; ductility and impact
CU-3 strength decline with increasing neutron fluence. The temperature of the SRP reactor tank walls is too low for significant swelling to occur from voids or gas bubbles resulting from neutron irradiation. In addition, experimental evidence has demonstrated that a relaxation of preirradiation stresses also results from fast neutron fluence. The reactor tanks are not expected to be affected by fatigue damage because the stresses encountered in the low-temperature, low-pressure system are well below endurance limits, and vibration from process-water circulation has been reduced to a low level.

G.3.1.1 Limit system

L-Reactor will operate at limits which are determined by a number of accident analyses for each reactor charge. These limits define the conditions at which the reactor can operate and still allow the protective instrument system to terminate any anticipated transient without exceeding prescribed damage criteria (for example, an approach to fuel melting). Three such limits are established, and the reactor is operated at the lowest of them.

1. The first limit is defined by assuming that the safety-rod scram--the primary emergency shutdown system--works on demand. This is the "transient protection" limit.

2. The second limit is defined by assuming that the safety-rod system fails and that an automatic backup system (called the automatic backup shutdown--safety computer, or ABS-S/C) is required to terminate the transient. This second limit defines the confinement protection limit, which is based on the criterion that the airborne activity confinement system not be damaged.

3. The third limit, the emergency cooling system (ECS) limit, is established by assuming a minimal level of emergency cooling system operability.

In principle, any of the three limits could be most restrictive; however, in practice and by design, the transient protection limit is usually the most restrictive. A more complete description of the SRP Limit System is given in the SAR.

Each plutonium-producing reactor charge is moderated and cooled by D_2O and has the same spacing between fuel and target assemblies. But changes in moderator and coolant temperature coefficients during the charge exposure time and changes in the average and relative fissile content of the fuel assemblies, among others, require that an accident analysis be made for each charge. Some of the analyses can be generic in nature (such as confinement protection limits), but the more important analyses, which generally fix the operating limits for the charge, are charge-specific. A summary of the analyses required for a charge is given in Table G-3.

The range of operating variables experienced during the 30 years of reactor operation at Savannah River Plant are given in Table G-4. The large ranges shown here demonstrate the flexibility available in a charge design. L-Reactor is currently scheduled to operate with a mixed-lattice, plutonium-producing charge, as shown in Table G-5.

G.3.1.2 Reactor shutdown systems

Several redundant systems operate to rapidly shut down the reactor, if necessary. The primary reactor shutdown mechanism is safety and control rod insertion, activated by the scram instruments or manually; the secondary shutdown

Table G-3. Summary of data and analyses for each reactor charge

Data and analysis	Analysis required
Technical limits and transient-protection limits for assembly effluent temperature	Yes
Technical limits and transient-protection limits for film-boiling burnout risk	Yes
Technical limits and transient-protection limits for reactor effluent temperature	Yes
Confinement protection limits for accidents with assumed inoperative safety rods	Yes
Criticality during withdrawal of safety rods	Yes
Shutdown system worths	Yes
Primary and secondary scram circuit designation	Yes
Natural convection cooling	Yes
Mechanical and metallurgical properties during discharge	Yes
Protection against criticality during charge-discharge operations	Yes
Storage and handling of enriched uranium assemblies	Yes
Shield heat loads	Yes
Emergency cooling of irradiated fuel	Yes
Heat removal from safety and control rods	Yes
Temperature and void coefficients	Yes
Startup accident analysis	Yes
Xenon oscillations	Yes
Compliance with Technical Standards and safety analyses	Yes

system is the supplementary safety system (injection of gadolinium nitrate), activated automatically by the gang temperature monitor and the safety computers, or manually.

Safety rods

The safety rods provide a primary rapid-shutdown mechanism for the reactor and thus prevent core damage. Upon receipt of a scram signal, the safety rods drop into the reactor core in about one second. L-Reactor has 66 safety rods made of cadmium, an effective neutron absorber.

Control rods

When a shutdown (scram) signal is received, in addition to the safety-rod drop, the 61 clusters of control rods are automatically driven into the reactor. The control rod system is designed such that the reactor is subcritical

Table G-4. Range of operating variables in SRP reactor charges

Variable	Range
Thermal neutron flux (full power)	5×10^{13} to 7×10^{15}[a] $n/(cm^2)(sec)$
Reactor power (full power)	650[a] to 2915 MW (thermal)
Assembly power	Up to 21 MW (thermal)
Prompt coefficient	$+2 \times 10^{-5}$ to -15×10^{-5} k/°C[b]
Moderator coefficient	-1×10^{-5} to -35×10^{-5} k/°C
Reactivity in control rods	Up to 30% k at cycle beginning; to 0.5% k at cycle end
Reactivity in xenon after shutdown	Up to 60% k
Irradiation cycle length	4[a] to 400 days
Fuel heat flux	Up to 914 watts/cm^2
Total D_2O flow	341 to 619 m^3/min
D_2O flow per assembly	Up to 66.2 ℓ/sec
Assembly coolant velocity	Up to 22 m/sec

[a]Special high-flux charge.

[b]Overall temperature coefficient (prompt plus moderator) is always negative. k is the multiplication factor of the reactor--effectively the number of neutrons present at the end of a neutron generation for each neutron present at the start of that generation.

Table G-5. Nominal values of operating parameters for typical L-Reactor charge

Operating parameter	Plutonium producer (mixed-lattice)
Principal fuel	Enriched uranium
Principal target	Depleted uranium
D_2O flow (m^3/min)	
Per fuel	1.59
Per target	0.89
Total reactor	587
D_2O velocity (m/sec)	
Fuel	5.8
Target	7.6
H_2O flow (m^3/min)	672
Power, MW (thermal)	
Per fuel	7.4
Per target	2.5-4.8
Total reactor	2350
Fuel surface heat flux, watts/cm^2	220
Assembly effluent D_2O temperature, °C	
Fuel	113
Target	85-110

when the control rods are inserted and the safety rods are withdrawn. The control rods can be driven in singly, or by a gang drive; the rate of insertion is less rapid than that for the safety rods.

Scram instruments

The scram circuits monitor reactor operation and will cause the safety rods to fall and the control rods to drive in. The scram instruments for a particular variable (e.g., neutron flux, coolant pressure, etc.) are set to produce a scram at the operating limit imposed for safe operation. A reactor scram at the setpoint will prevent damage to the fuel and the reactor. The scram, or shutdown instruments, installed in L-Reactor are listed in Table G-6.

Table G-6. Automatic scram circuits

Variable measured	Number provided[a]
Neutron flux (High-level flux monitor)	Four
Operability of neutron flux monitors	One
Rate of change of neutron flux (period)	Two
D_2O plenum pressure	Two
Blanket gas pressure	Two
H_2O supply header flow	One for each of two H_2O headers
Individual heat exchanger H_2O flow	One for each of 12 heat exchangers
Control rod coolant supply pressure	One
Moderator level	One
D_2O pump a.c. power supply	One for each of six pump motors
Assembly coolant flow	600 in L-Reactor
Assembly average effluent temperature[b]	600 in L-Reactor
Control system power supply	One
Seismic activity	Two of three coincidence
Operability of safety computers	One

[a]A manual scram circuit is also provided.
[b]Four thermocouples in each of 600 monitor pins provide maximum and average assembly effluent temperature. Monitoring and scram signals are provided for each of the 2400 monitoring thermocouples.

Supplementary safety system

The supplementary safety system (SSS) is a fully independent system that acts as a backup shutdown system. The SSS can be actuated manually or automatically if safety rods fail to shut down the reactor. When the system is activated, gadolinium nitrate, an effective neutron absorber, is injected into the moderator. The SSS is designed such that the reactor will be subcritical even if all safety and control rods are in the fully withdrawn condition. The system has redundant tanks, piping, and valves.

Automatic backup shutdown-safety computer (ABS-S/C)

The ABS-S/C is a backup system that consists of two computers, each of which monitors an average of 300 assembly effluent temperatures and flow every 0.36 second, and which will actuate the SSS to shut down the reactor if the safety rods fail to reduce reactor power in the event of a scram. It will terminate all identified transients for which the primary shutdown mechanism, safety-rod insertion, fails.

Automatic backup shutdown-gang temperature monitor (ABS-GTM)

The ABS-GTM is a second automatic backup shutdown system that is independent of the safety-rod scram system. The sensors are dual monitor pin thermocouples in three fuel assembly positions associated with each of the three gangs of control rods. The sensors are set to actuate the SSS when monitored assembly effluent temperatures approach specified limits.

G.3.1.3 Engineered safety and support system

In addition to the systems discussed above, there are a number of other engineered reactor safety and support systems which help mitigate the consequences of an accident. Several of these systems are described below.

Emergency cooling system (ECS)

The ECS is designed to remove decay heat following a reactor shutdown by the direct addition of light water to the reactor core in case of loss of heavy-water coolant or circulation. Four sources of light water are available, at least two of which have to be online for reactor operation.

1. A diesel-driven booster pump which supplies H_2O from the 95-million-liter 186-L basin.

2. A header with a diameter of 107 centimeters pressurized by five pumps drawing H_2O from the 95-million-liter basin.

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

4. A line directly from the river water supply line, pressurized by the river water pumps.

The ECS is actuated automatically as liquid level decreases in the reactor tank or manually as abnormal conditions dictate. When the ECS is actuated, the diesel-driven booster pump starts, and valves are automatically opened or closed to couple the reactor with the primary sources of light water. Borated water from the storage header will be injected into the reactor first, to prevent a reactivity transient when the light water displaces D_2O in the reactor core.

Water removal and storage

If the heavy-water system ruptures, the heavy-water and light-water emergency cooling water would flow to sump pumps in the basement of the reactor building. The sump pumps deliver the water first to a 225,000-liter underground tank; the flow is then diverted to a 1.9-million-liter tank that sits in the 190-million-liter emergency earthen basin. Some of the water on the 0-level process room floor would drain directly to the 1.9-million-liter tank. If this tank should become full, the additional water bypasses the tank and flows into the emergency basin. The 1.9-million-liter tank is vented to the activity confinement system in the reactor building. Because the volume of the 1.9-million-liter tank represents about 10 times the reactor D_2O volume, no moderator is expected to reach the emergency basin. Hence, no tritium or fission product is expected to be carried into this basin.

Airborne activity confinement systems

L-Reactor has an airborne activity confinement system. In the event of an accident, airborne fission products may be released into the reactor room, and possibly into the heat-exchanger bay or the pump room. As shown in Figure G-1, the air from these areas is exhausted through a set of confinement filters before it is released to the stack. During normal operation, the process areas are maintained at a pressure that is lower than the pressure of the external atmosphere to ensure that all air from the process areas is exhausted through the activity confinement system.

Three large centrifugal fans exhaust the air from the process areas. Two of these fans normally are online, but only one is necessary to maintain the negative pressure. The air flow from a single fan is enough to prevent the overheating of carbon filters that might be caused by high retention of radioactivity after a severe accident. The three fan motors can be powered simultaneously by two electric sources:

1. The normal building power through at least two substations
2. The emergency building power from diesel generators.

In addition, each of the three fans has a backup motor, any two of which can be powered by dedicated diesel generators. Exhaust filters remove moisture, particulates, and halogens. The filter banks are enclosed in five separate compartments; three to five of these compartments are normally online at one time. Each compartment can be isolated for maintenance and testing; each contains the following filter banks, in the order of air-flow treatment:

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

2. Particulate filters, designed to retain more than 99 percent of all particles with diameters of 0.3 micron or larger.

3. Activated carbon beds that use an impregnated carbon to retain halogen activity if an accident were to occur. Special impregnants have been developed to improve the retention of organic iodide compounds. The effectiveness of these filters is discussed in Section G.5.1.2.

Confinement heat removal system

A confinement heat removal system (CHRS) is provided to prevent failure of the confinement system in the event of a postulated meltdown of a reactor core. Such a meltdown could occur from the nuclear decay heat if both normal cooling and emergency cooling fail. The CHRS provides limited water flooding on the 40-foot-level floor to cool any molten core material that may penetrate the reactor tank or process pipes.

The source of water for the CHRS is the disassembly basin. Only the top 1.4 meters of disassembly basin water can be drained onto the 40-foot-level floor. The remaining basin water still maintains adequate shielding and cooling for fuel elements stored in the basin. There is a system to provide makeup water to the disassembly basin from two sources.

Reactor room spray system

A system is provided in the reactor room to spray water on an irradiated assembly if one is accidentally dropped during unloading operations. This system consists of a header with twelve groups of fixed spray nozzles mounted on the reactor room wall. The spray pattern from these nozzles covers the area traversed by the discharge machine. Each spray nozzle group has its own actuation valve.

Component handling-cooling during discharge operations

During the interval between removal of irradiated fuel (or targets or other heat-producing assemblies) from the reactor and insertion in the cooling basin, the irradiated assemblies are cooled by water. Five sources of water are available to the discharge machine through four independent paths. Except at the final point of discharge to the assembly, each system has separate hoses, pipes, and actuation valves.

In normal practice, primary H_2O cooling is started automatically as soon as the assembly is withdrawn from the reactor and the water pan swings under the assembly. If primary H_2O flow stops, a secondary H_2O source is switched on automatically. Primary and secondary D_2O cooling is automatically available if the assembly is partially in the reactor, or if the assembly is over the reactor and the water pan does not move under the assembly.

The reactor room spray system is available if an assembly is dropped onto the floor of the reactor room. Assemblies are not discharged unless the maximum decay heat generation rate is less than could be dissipated by the discharge machine cooling water or by natural convective cooling in the disassembly basin if the assembly is dropped and lies in a horizontal position.

G.3.1.4 Electric power

Electric power from the SRP power grid is supplied to L-Area by two 115-kilovolt transmission lines. These lines enter L-Area from two directions. There are also three 30,000-kilovolt-ampere transformers in the area that are connected to the 115-kilovolt grid. Each transformer can carry the L-Area load.

Emergency power for the reactor building is furnished by diesel generators. Two 1000-kilowatt a.c. generators supply emergency power to the reactor building if normal power fails. Eight 103-kilowatt d.c. generators supply power to the process pump motors that maintain the cooling-water flow to the shutdown reactor if the normal a.c. power fails; six of these generators are normally operated at all times, and the remaining two are on standby. Four other diesel generators are located throughout L-Area and provide backup power for ventilation fans, street lights, and other equipment.

G.3.1.5 Process and effluent monitoring

All gaseous radioactive releases through the stack are monitored continuously by gamma spectrometry. Stack-effluent tritium is monitored by two ion chambers that operate in parallel. Moisture is removed from the air to one of the chambers to provide a differential current between the chambers. A continuous sampling technique with daily quantitative analysis is also used. All other air and water samples are monitored routinely; quantitative release records are kept. Above-normal activity levels are investigated to locate the source so the condition can be corrected.

Samples are analyzed routinely to quantify the key surveillance radionuclides from the following sources:

1. The moderator
2. The stack exhaust air
3. The effluent heat-exchanger cooling water
4. The disassembly-basin effluent purge water

G.3.2 Site features

G.3.2.1 Site location

The Savannah River Plant occupies an approximately circular area of about 800 square kilometers. The L-Reactor site is located in the south-central portion of the Savannah River Plant.

G.3.2.2 Site description

The predominant site feature that would mitigate the consequences of an accident at the L-Reactor is the distance of 9 kilometers to the nearest SRP boundary. Although South Carolina Highway 125 is only 5 kilometers from L-Reactor, there are procedures for stopping traffic and clearing all personnel off the highway within a short time of any incident at the SRP.