G.3.3 Emergency preparedness

G.3.3.1 Emergency planning - onsite

An onsite Emergency Operating Center (EOC) is maintained at SRP to provide immediate and informed response to any site accidents. The responsibility for emergency response at the plant facilities (including L-Reactor) within the Savannah River complex is clearly defined (DPSOP 67 and 129). Adequate staffing to provide initial facility accident response in key functional areas is maintained at all times. Timely augmentation of response capabilities is available and the interface among various onsite response activities is clearly specified (DPSOP 129).

Patrol EOC personnel operate from the communications room 24 hours a day, 7 days a week, using radio and telephone equipment that links all patrol installations throughout the plant. During emergencies, specialized communications equipment is operated to contact production control rooms, local law enforcement agencies, state and Federal radiological agencies, state and local government officials and others, as required by the specific emergency.

Accident emergency classifications and action-level schemes have been established (DPSOP 129 and 175). There are provisions for prompt communications among principal response organizations and emergency personnel (DPSOP 129).

The EOC is equipped with self-contained power and service facilities and a shelter capable of housing 20 persons for 30 days under emergency conditions. The center has blast doors, air locks, and an emergency escape hatch. The EOC will withstand blast pressures to 15 psi and provides a radiation protection factor of up to 6000. It can be completely isolated from the outside environment in about 5 minutes. The EOC is also equipped with air filters for emergency use. A sand filter system is underground in a blast-resistant concrete structure. There are also two carbon filters in series. The first unit will absorb chemical warfare gases; the second unit will absorb radioactive iodine.

The EOC shift crew and meteorological operations center contains radio and telephone equipment for all necessary communications in handling response to an emergency condition (DPSOP 129 and 307). Equipment is also available for monitoring a release from the reactor areas and obtaining critical data from instrumentation in an uninhabited reactor building. The Weather Information and Display (WIND) system terminal provides facilities to accurately predict downwind hazards from chemical and radioactive releases. Maps and plotting equipment allow a visual organized presentation of the data for EOC staff personnel. Equipment is also available for monitoring radiation and chemical hazards to personnel occupying the EOC.

The EOC staff room contains a comprehensive communications network permitting the DOE-SR and Du Pont staff to monitor communications on the patrol and emergency radio networks and also to monitor telephone conversations between the Area Emergency Coordinator at the incident site and Production and Technical Management personnel in the EOC. Copies of Emergency Procedures, pictures of vital process equipment and process schematics, maps, television monitors and a number of other visual aids are available for use by EOC staff liaison personnel in keeping the EOC staff informed concerning the status of an emergency. Future information and communications improvements, either authorized or planned, include a remote detection and control (REDAC) terminal from the reactor areas and a plantwide cable television network that will provide video and audio communications between the Plant Production areas and the EOC.

During an emergency situation the organizational and emergency procedures and responsibilities are clearly defined and shared between Du Pont and the Department of Energy (DOE). Procedures for notification of emergency occurrences to state and county officials exist through current memoranda of understanding.

An Offsite Communications Center (OCC-A) is also maintained in Aiken, South Carolina. The purpose of the OCC-A is to assure a communications link with Savannah River Plant if (1) highways to SRP are impassable, (2) telephone lines are inoperative due to overloads, or (3) the Emergency Operating Center is not accessible. The OCC-A also provides an offsite location for EOC staff members or key personnel during a national or local emergency.

In the event of emergency assignment to OCC-A, Du Pont and DOE-SR management representatives would serve as liaison between the EOC staff and offsite personnel, using a direct telephone line from OCC-A to the Emergency Operating Center. A monthly check of this line is made to ensure operability. The OCC-A also contains a radio with SRP Patrol, SRP Emergency, and DOE-SR net channels. Maps, copies of emergency procedures, and other visual and briefing material are also located in the OCC-A.

As required, the OCC-A can serve as a location for use by DOE-SR Office of External Affairs personnel to brief media representatives. It could also be employed as a temporary office location for a small number of representatives from state and Federal agencies, or for local government officials.

The Dwight D. Eisenhower Army Medical Center, Ft. Gordon, Georgia, is also on call to respond to the medical requirements of the SRP (DPSOP 129). All other facilities, communications, and emergency resources are maintained within the Savannah River complex (DPSOP 129 and 175).

A minimum of four emergency training exercises are conducted annually to test and evaluate the performance of EOC personnel and equipment.

G.3.3.2 Emergency planning - offsite

South Carolina and Georgia, and their respective counties of Aiken, Allendale, Barnwell, and Burke and Richmond have existing Emergency Response Plans in varying degrees of completeness. State and county officials are being assisted by DOE in fully developing their respective Emergency Response Plans. These plans are discussed further in Appendix H.

G.3.3.3 WIND system

The Weather Information and Display (WIND) system is an automated emergency response system for real-time predictions of the consequences of liquid and atmospheric releases from the Savannah River Plant (Garrett, 1981). The WIND System has been developed over the last 10 years specifically for use at 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 the emergency response codes remotely, codes which use empirical information on atmospheric diffusion and deposition gathered at Savannah River Plant (Carlson et al., 1982; Buckner et al., 1975), and stream transport and diffusion codes that have been calibrated with dye tests in the SRP streams (Garrett and Murphy, 1981).

The SRP Health Protection Department staffs all production areas 24 hours a day with technicians who are trained to run the WIND system emergency response codes in addition to the four meteorologists, a computer system manager, a field engineer and 3 technicians who comprise the basic team that operates the WIND system.

Computer codes have been developed which allow display of latest or archived meteorological data from the SRP towers or Automatic Forecasting and Observation System (AFOS); trajectory, concentration, deposition, and dose calculations for atmospheric releases; concentration calculations for releases to SRP streams; and estimates of reactor core melt based on stack monitor data. Dose calculations include inhalation doses and whole-body doses caused by gamma radiation from noble gases and iodine. Atmospheric transport and diffusion codes range in complexity from Gaussian trajectory models (Cooper and Rusche, 1968) that can be run in less than 5 minutes, to two- and three-dimensional codes that require about 1 hour of computations.

G.4 ACCIDENT AND IMPACT ASSESSMENT

As a means of assuring that L-Reactor features meet acceptable design and performance criteria, the potential consequences of a number of postulated transients have been evaluated. These postulated transients or accidents are used to help establish system design characteristics and operating limits. They are described in the following subsections. These subsections also describe the results of analyses used to estimate the possible impacts and risks associated with a group of four postulated severe accident sequences--with a low probability of occurrence--that could release significant radioactivity to the environment. The consequences to persons offsite are described in Section G.5. The potential radiological consequences for all of these postulated incidents cover a considerable range of values depending on the particular course taken by the accident and the conditions, including meteorology, prevalent during the accident.

G.4.1 Postulated transients and accidents

The postulated transients and accidents considered in the safety analysis and evaluation of L-Reactor include all incidents listed in Table G-1. Four hypothetical severe accidents are specified to cover a spectrum of credible events postulated to release significant quantities of radioactivity. |TC

All reactor-related accidents share the common characteristic of too much power for too little cooling. Accident analyses can be grouped into three broad classes.

1. Reactivity addition (equivalent to removal of neutron absorber) that increases reactor power or power in a local region of the reactor.

2. Flow reduction, caused by loss of pumping power, reduction of circulation, or loss of coolant, that reduces the cooling capacity of the reactor or individual heat-producing assemblies. The loss of moderator will be considered in this class.

3. Non-nuclear types of accidents that are not directly related to rapid changes in the reactor conditions.

For all of the accidents analyzed, the reactor would be shut down by the primary or redundant shutdown systems before:

1. Reactor tank is damaged, or the
2. Confinement system is breached.

The radiological consequences of incidents often called anticipated operational occurrences, fall within limits of normal operational releases of radioactivity. The key postulated transients in this class are all incidents listed in Table G-1 with the possible exceptions of incidents 4 through 7, 16, and 18. Many are credible but have a very low likelihood of occurrence. Incidents 1, 2, 7, 12, 13, 14, and 15 have a nontrivial occurrence record among all of the SRP reactors.

The following discussion addresses each transient in Table G-1.

G.4.1.1 Single control-rod withdrawal

The reactor is assumed to be operating at full power at operating limits. The control rod withdrawn is adjacent to the hottest assembly in the reactor. Withdrawal is at the maximum rate possible requiring two minutes to move from the full-in to the full-out position. The hypothesized withdrawn rod is near the edge of the reactor, thus causing a radial power tilt as well as an increase of local reactor power.

Occasional unwanted control rod motion is expected to occur at L-Reactor. Reactor and hottest assembly powers would increase until scram or cessation of rod motion. In a limiting case, the reactor power increases almost linearly at a rate of 0.4 percent per second, while the hottest assembly power increases at about 1.9 percent per second. Primary scram is based on the assembly coolant |TE temperature monitor in the hottest assembly. Secondary scram is based on nearby assembly coolant temperature monitors or high-level flux monitors. The accident is analyzed for every reactor charge to establish normal operating limits and is analyzed generically to establish confinement protection limits. The operating limit on assembly coolant flow and coolant temperatures is set so that a reactor scram would prevent the coolant from reaching the saturation temperature. Under this condition there would be no damage to fuel or reactor, and radioactivity would not be released to the environment.

G.4.1.2 Partial control-rod movement

The partial rod is assumed to be centered axially in the reactor at the position of maximum absorption. The rod drives in or out at the maximum speed of 0.136 and 0.118 foot per second, respectively. The partial rod is in a control cluster near the side of the reactor, thus causing radial power tilting. Axial power distribution changes as well as radial power distribution changes will occur.

Unwanted partial rod motion is expected to occur occasionally. The scram bases are the same as for single control-rod withdrawal. Reactor and local power increases are less than or equal to those caused by unwanted full rod withdrawal. Analysis considerations are the same as for full rod withdrawal (Section G.4.1.1).

G.4.1.3 Gang-rod withdrawal at full power

Groups of control rods called gangs are moved together in normal reactor operation. Gang I consists of the inner 19 clusters of control rods, Gang II the next ring of 18 control clusters, and Gang III the outer ring of 24 clusters. The reactor is assumed to be at full power at operating limits. A gang of control rods moves out continuously at maximum speed. Significant radial power perturbation cannot occur.

Unwanted continuous gang withdrawal is not expected to occur. Short-term withdrawal, because of spurious signals in the control computer, might occur. Reactor power and hottest assembly powers would increase until scram or cessation of rod motion. Reactor power would increase at a rate of 1.2 percent per second, while the hottest assembly power would increase at 1.7 percent per second. Primary scram is based on assembly coolant temperature monitors. Secondary scram is based on high-level flux monitors (can be the primary instrument). The gang-rod-withdrawal accident is analyzed for both transient protection limits and confinement protection limits. This accident is often the most restrictive in setting reactor operating limits.

G.4.1.4 Gang-rod withdrawal at low power

It is assumed that inadvertent gang withdrawal occurs three decades (a factor of 1000) below full power (at 1 to 3 megawatt thermal). The inventory of xenon-135, which has a very large neutron cross section, is at or near maximum. At full power, xenon-135 absorbs about 3 percent of all neutrons produced in the reactor, but when the reactor is shut down, the inventory of xenon-135 increases by decay of iodine-135 to several times its full power inventory. If the reactor were restarted with a large inventory of xenon-135, its burnup would add reactivity that could exceed that caused by control rod withdrawal. Temperature coefficients do not provide sufficient negative feedback to prevent a rapid power rise until the reactor power reaches levels within a decade of full power.

This gang rod withdrawal at low power is expected to have the same frequency of occurences at full power. Reactor power would increase at rates greater than for gang withdrawal at full power, becuase of the xenon-135 burnup. Primary scram is based on high-level flux monitors and period (rate of flux increase) monitors. Secondary scram is based on assembly coolant temperature monitors at about 50-percent full power. The accident is analyzed for each type of charge. For certain high-level flux charges, restrictions are placed on the time for xenon to decay before the reactor can be restarted.

G.4.1.5 Control-rod melting

If, based on control-rod heat flux, control-rod melting is possible for the reactor charge design being analyzed, it is postulated that:

1. Control-rod heating and melting occur adiabatically
2. The neutron-absorbing material disappears from the reactor as soon as melting occurs
3. Partially inserted rods are severed at the midplane by melting, allowing the lower part to drop
4. The control rods melt in clusters on the outside of the reactor

Control rods can melt because of possible reduced cooling, provided they have sufficiently high power density. Reactor power increases are similar to those for single-rod withdrawal accidents, although the rate of reactivity addition is different. Effluent temperature monitors for assemblies in the affected control cluster would cause a scram. For a septifoil with no forced flow (unseated), control-rod melting has been analyzed to begin with film-boiling burnout at a heat flux of 102 watts per square centimeter. Typical maximum heat flux values for current charges are 57 to 74 watts per square centimeter. The control-rod melting accident is not considered in establishing limits if the maximum heat flux in the charge is less than the 102 watts per square centimeter value calculated to be required for control-rod melting.

G.4.1.6 Loss of target

The analysis of a loss-of-target accident is an effort to conceive of all possible means by which reactivity could be added to the core. The postulated loss of target is the consequence of an extremely improbable loss of cooling to only a single target and should be considered in that context. It is assumed that an abrupt reduction in coolant flow occurs in a high-power target. No known mechanism exists to cause such a reduction in flow. The target assembly is assumed to melt, whether the reactor is shut down immediately or not. The target material disappears from the reactor as soon as the target melts, which increases reactivity. If a high-power target assembly should melt, reactivity could be added at a rate greater than that for control rod or gang rod withdrawal. The flow monitor for the afflicted target assembly would be the first scram instrument to respond. The secondary scram circuit would be the assembly coolant temperature monitor. The course of the accident would be trivial if the safety-rod scram system performed as designed. If the safety-rod scram should fail, core damage would be prevented by the automatic backup systems.

Loss-of-target accidents are not considered in deriving transient protection limits because the postulated large, and abrupt, loss of flow is considered so improbable. More realistic reductions in flow to individual assemblies are considered for operating limits. The loss-of-target accident is considered for conservatism in establishing confinement protection limits.

G.4.1.7 Loss of fuel

The analysis of a loss-of-fuel accident, like the loss-of-target accident, is an effort to identify all conceivable reactivity addition transients independent of currently accepted credibility arguments. Again, the assumption is made that a sudden, abrupt loss of flow (for which no initiating mechanism has been identified) occurs to a fuel assembly. The fuel assembly melts, and some of the molten debris from the uranium-aluminum alloy fuel may be entrained and reach the moderator where it would then be exposed to a higher neutron flux than in its usual pre-melt condition. The exposure would cause a temporary increase in reactivity, until the debris is swept from the reactor core. Simultaneously, steam voids formed in the moderator around the fuel particles would decrease reactivity.

The primary scram instrument is the assembly flow monitor. Secondary scram instruments are the high-level flux monitors and assembly temperature monitors. Even with the conservative assumptions used for this accident, the calculations to assess this accident indicate that the primary and backup shutdown systems provide adequate protection.

G.4.1.8 Reloading error

A reloading error is the basis for one of four hypothetical events postulated to cover the spectrum of accidents that could have a significant impact on the environment.

The reactor is shut down and the charge-discharge operation is in progress in a mixed-lattice charge. It is assumed that an error is made when irradiated targets are discharged from adjacent positions without charging fresh targets to these positions, or in spite of mechanical interlocks, fuel assemblies are charged to target positions. It is further assumed that charge design constraints have failed, and the reactor becomes supercritical.

The result of the postulated accident could be fuel melting and radioactivity release to the reactor building. The scram system is ineffective because the safety and control rods are already in the reactor. The Supplementary Safety System is much less effective than at full power because of the reduced moderator and coolant circulation rate required during charge-discharge operations. The neutron flux changes caused by reloading errors are highly localized and therefore the two fission counters external to the reactor core might not detect the error.

To help avoid this potentially serious accident, each reactor charge is analyzed to search for the worst possible reloading error. The charge is designed such that it does not become critical during this error. An improved monitoring system, consisting of six internal fission chambers, has been designed, tested, and installed in L-Reactor.

The course of a postulated power excursion caused by reloading errors has been calculated. The highly localized damage involves less than 3 percent of the core.

Reloading accidents are not considered in establishing normal operating limits because full-power parameters are not involved. Nor are they weighed against the conventional criterion for confinement protection because the reactor coolant system is open to the atmosphere during reloading. However, calculations have shown that the confinement system integrity is not seriously challenged by this accident.

G.4.1.9 Loss of D_2O coolant pump power

Loss of all offsite a.c. power is a credible event. The onsite a.c. power generation is insufficient for simultaneous full-power operation of all SRP reactors. Thus, it is assumed that loss of a.c. power could occur for any reactor, and further that a.c. power to the six D_2O coolant pumps is lost simultaneously. The d.c. motors to the pumps continue to supply power and would maintain flow at 29 percent of full flow. Flywheels between the pumps and motors slow the flow decay transient.

As the flow decreases, fuel and target assembly effluent temperatures increase. The increased water temperature produces a negative reactivity, which causes reactor power to drop slightly in the first 2 seconds. The first scram instruments to respond would be the two plenum pressure monitors. These are backed up almost immediately by the assembly coolant flow monitor.

The accident analysis is used to set both transient protection and confinement protection limits. The safety-rod scram would limit the maximum coolant temperature to a value at least 7°C lower than the saturation temperature. If the safety rods are ineffective, the ABS-S/C would limit coolant temperatures to about 5°C less than the saturation temperature. If both a.c. and d.c. power were lost (for which no mechanism has been identified), flow would continue to decay until either some pumping power is restored or emergency coolant is introduced. This accident is considered separately as the loss of D_2O circulation accident.

G.4.1.10 Loss of H_2O pump power

The loss of H_2O pump power is a credible event. It is assumed that electrical power is lost to the pumps supplying H_2O directly to the reactor heat exchangers. The H_2O flow decreases to 25 percent of normal sustained by gravity flow. *Gravity flow is assured by the difference in elevation of the* cooling water basin and the heat exchangers at -20 ft.

As a result of the decrease in H_2O flow, the temperature of the D_2O
TE| leaving the heat exchangers and entering the reactor would increase. This would increase the D_2O outlet and fuel assembly effluent temperatures. Reactor power would decrease because of the negative temperature coefficient of the reactor charges. The fuel coolant temperatures change more slowly than for a loss of D_2O pump power. The scram instruments to respond first would be the 12 heat exchanger flow monitors, followed by the two H_2O header flow monitors and the fuel assembly effluent temperature monitors.

The accident is not used to establish reactor operating limits because the transients are slow compared to other flow reduction accidents. The temperature of the hottest assembly would be 19°C below the saturation temperature (coolant boiling) at shutdown caused by the ABS-S/C. The accident is used in establishing confinement protection limits.

Alternative forms of this accident have been postulated. Clearly, a partial loss of pumping power would yield a less severe flow transient than the total loss of power considered here. Plugging or breaking a line to an individual heat exchanger would be still less of a perturbation. A break in an H_2O supply header could, if the break were large enough, cause a more severe flow transient than loss of pumping power. The response of D_2O temperature to such an improbable abrupt and total loss of H_2O cooling has been calculated. It was found that effluent temperatures hardly change before a safety-rod scram (triggered by H_2O flow monitors) shuts down the reactor. If H_2O cooling is not restored, then the assembly and reactor effluent temperatures would eventually increase because of fission product decay heat. Manual actuation of the ECS would then be required, but the accident would be less severe than the loss-of-circulation accident. No melting would occur.

G.4.1.11 Loss of both D_2O and H_2O pumps

The possibility of simultaneous loss of a.c. power to both D_2O and H_2O pumps has been considered as an extreme extension of either of the accidents considered singly. However, analysis shows that the accident of D_2O flow reduction increases coolant temperatures so much more quickly than H_2O flow reduction that the two accidents are essentially independent. In the event of loss of both D_2O and H_2O pumps, coolant temperatures increase at the same rate as for loss of D_2O pumps only. Thus, the combined case is not considered in establishing operating or confinement protection limits.

G.4.1.12 Pump shaft break

This accident is conceivable but not likely to occur because of over 600 pump years of operation without failure and because of frequent inspections. It is assumed that a pump drive shaft breaks between the D_2O pump impeller and the flywheel when the reactor is at full power. The impeller is left free to rotate, which allows a reversal of flow through the pump.

If a shaft breaks, the fluid momentum drops to zero (and reverses) much more quickly than if a D_2O pump should lose power (because the energy stored in the flywheel would reduce the rate of decrease in flow). The resulting flow changes in the reactor are asymmetric--the fuel and target assemblies in the affected coolant sector have a greater flow reduction than other assemblies in the reactor. Some assembly flows reduce to 75 percent of normal in 2 seconds, while average assembly flows reduce to slightly greater than 80 percent normal. The primary scram instruments are the plenum pressure monitors, followed by the assembly flow monitors.

Analysis of the pump shaft break accident is used in deriving operating and confinement protection limits. The safety-rod scram would prevent the assemblies from melting. The maximum assembly temperature can exceed the boiling point if the safety rods fail to drop. Reactor limits are set such that if this happens, the steam generated does not produce a force great enough to lift the plenum. The steam generation lasts for too short a time to cause any assemblies to melt and release radioactivity.

Analysis of the case in which the broken pump shaft freezes and prevents the impeller from turning has been compared with the analysis in which the impeller is left free to rotate. The latter case produces the more restrictive limits.

G.4.1.13 Rotovalve closure

Although spurious rotovalve closure is possible and has occurred, the combination of closures specified for the postulated transient has never occurred and there is no known mechanism for an occurrence. Rotovalves are installed in the six external loops of the D_2O circulation system between each of the 12 heat exchangers and the reactor plenum. During normal, full-power operation, the rotovalves are fully open. During maintenance work, the rotovalves are fully closed to prevent loss of D_2O. It is assumed that the two rotovalves in each of two external loops close simultaneously when the reactor is at full power.

The flow reductions have been calculated for simultaneous closure of the four rotovalves. Flow in the minimum flow assembly after 2 seconds is about 97 percent of normal, compared with 75 percent of normal flow at 2 seconds for a pump shaft break accident. The difference between the maximum expected assembly effluent temperature and the saturation temperature is large compared with the difference for a pump shaft break accident.

The primary scram instruments are the plenum pressure monitors and the assembly flow monitors. Postulated rotovalve closure incidents are not used to set transient protection limits, because this incident always yields higher limits than the pump shaft break incident. Postulated rotovalve closure incidents are used to set confinement protection limits.

G.4.1.14 Flow reduction in a single assembly

It is assumed that a gradual reduction in flow occurs to a single coolant channel within an assembly. This could be caused by the swelling that accompanies a cladding failure in a uranium fuel or target assembly. Fuel failures resulting in flow reduction have occurred at SRP.

As assembly flow gradually decreases, the assembly channel effluent temperature gradually increases. The assembly coolant flow monitor would be the first to scram the reactor. The scram setpoint for the monitor is required to be at the point that would prevent coolant boiling in the hot channel. The assembly effluent temperature monitors are also set to prevent boiling in the hottest channel. This flow reduction incident is used to determine transient protection limits for the reactor. One other case is considered in establishing confinement protection limits. This is the abrupt and complete flow reduction that is postulated to lead to the loss of assembly accidents already discussed. No specific initiating mechanisms have been identified for this abrupt flow reduction.

G.4.1.15 Loss of control-rod cooling

Flow reduction or blockage in the header supplying cooling to the control rods, or in the individual septifoil housing the control rods is unlikely because there are strainers in the headers and because heat exchangers with much smaller flow passages are upstream. Instead of a flow blockage, it is assumed that the septifoil housing is unseated in the reactor, thus reducing the flow to zero.

For this transient, calculations show that for current charges control-rod temperatures increase, but damage or melting does not occur because the calculated control-rod heat flux is not in excess of a high specific limit. If the heat flux exceeds the limit, then a control-rod melt accident is considered.

A reduction in header flow would cause a reactor shutdown within 2 seconds and prevent any damage. If a control rod should melt because of very high heat flux and septifoil unseating, then the assembly effluent temperature monitors around the affected cluster would shut down the reactor. No fuel or target assembly damage would result.

G.4.1.16 Loss of blanket-gas pressure

Slow leaks of blanket gas have occured at Savannah River Plant, but not a sudden rapid drop in pressure. It is assumed that a blanket-gas leak reduces the blanket-gas pressure linearly from 0.136 to 0 MPa in 2 seconds. This loss of blanket-gas pressure would reduce the saturation temperatures in the reactor and cause evolution of the helium gas dissolved in the D_2O. Other secondary results follow. Cavitation may occur in the external cooling loops, which would reduce reactor coolant flow and increase coolant temperatures. A second consequence of losing pressure is that the dissolved gaseous helium would appear as bubbles in the D_2O, which would cause a negative reactivity effect and drive reactor power down. The lower power would produce a positive reactivity feedback and the power would rise again. A safety-rod scram would occur after 1 second. Power and temperature oscillation could occur because of evolution of helium if the scram did not occur. Oscillations in currently operating charges would be small. However, the ABS-S/C would shut down the reactor after 5 seconds, so that realistically, no oscillation would occur. |TC

The primary and secondary scram instruments are the two blanket-gas pressure monitors and the assembly effluent temperature monitors. Analysis of this accident is used to set transient protection limits. The assembly coolant temperature monitor is required to have its scram setpoint set low enough to ensure that the saturation temperature of the channel exit is not exceeded at a blanket-gas pressure of 0.129 MPa (normal operating pressure is 0.136 MPa). This ensures no reactor damage and thus no release of radioactivity. |TC

G.4.1.17 Loss-of-coolant accident

It is postulated that a leak occurs somewhere in the D_2O coolant system when the reactor is at power. There are two classes of leaks: credible small leaks and a hypothetical, very large, sudden leak.

If a leak rate greater than 15 liters per minute should occur, the moderator level in the reactor tank, the blanket-gas pressure, and the plenum pressure would all decrease. The response would be as follows:

1. Automatically shut down the reactor.
2. Isolate the leak as much as possible.
3. Activate the Emergency Cooling System, if needed, to replace the lost D_2O with H_2O.
4. Maintain circulation to cool the fuel and target assemblies. (One other result of a large leak, the release of radioactivity to the reactor building and the environment, will be discussed in following sections.)

The scram instruments that would be activated are the moderator level, blanket-gas pressure, or plenum-pressure circuits, followed by the individual assembly flow and temperature monitors.

A large effort has been expended on the analysis of credible and hypothetical leaks. An ECS supplied from four independent sources of water can be activated manually or by logic circuits connected to the reactor scram instruments. Analysis shows that no fuel melting would occur for any credible loss-of-coolant accident. Of the credible accidents, the most likely would be a break in one of the smaller pipes in the auxiliary cooling systems. An example would be a break in a pipe supplying D_2O coolant to the septifoil system. The leak rate from this system would be 14 cubic meters per minute; no fuel damage would occur after the ECS was actuated.

The analysis is also made for a hypothetical maximum leak rate--an abrupt, double-ended break of a large pipe accompanied by other circumstances that render two of the three ECS supply systems ineffective. The worst of the accidents analyzed is a break in a line that also serves as one of the lines that would supply emergency H_2O coolant. The accident is not considered credible in the SRP system of stainless steel pipe operating at relatively low pressures of approximately 100 psi. For this hypothetical large leak, the ECS would limit the accident to 1 percent core damage if the ECS were degraded by a valve failure in another ECS supply line.

Shutdown would begin about 1 second after the pipe break. Analysis of the accident indicates that fuel damage does not occur in this 1-second interval. The longer term flow transient analysis indicates that damage may occur later. The factors that enter into the analysis are reactor power, power distribution, reactor flow, flow distribution, ratio of fission product decay power to normal operating power level, the ECS supply rate, and finally the degree of fuel damage established as a function of assembly flow and power. Reactor power is limited to a value that would produce less than 1 percent core damage if this hypothetical maximum leak rate should occur and only one of the three ECS systems were operable. The releases for this accident are discussed in Section G.5.

No fuel melting is anticipated in any credible LOCA. But some radioactivity will be released to the environment in any LOCA. The main contributor to offsite dose is tritium in the moderator (formed from neutron capture by deuterons). The tritium is released mainly by evaporation. The amount released depends on the size of the leak and on the disposition of D_2O leaking from the reactor to the reactor building.

Unless the leak were stopped, the entire inventory of D_2O could be released to the reactor building. This is the basis for the large moderator spill accident which is one of the four hypothetical events postulated to cover the spectrum of accidents that could release radioactivity to the environment. Almost all of the D_2O would be contained in two closed tanks outside the reactor building. Because the only vent path for the tanks is back to the reactor building, any tritium released by evaporation would eventually be discharged through the 61-meter stack. The releases for this accident are discussed in Section G.5.

G.4.1.18 Loss of D_2O circulation

The complete loss of D_2O circulation is considered highly improbable. It is assumed that a complete loss-of-D_2O circulation occurs by loss of all pumping power or some obstruction. Loss of pumping power could occur if all electrical power were lost and the motor room were flooded so as to stop the d.c. motors. Obstruction could occur if all rotovalves were closed, or if the D_2O became frozen.

The loss of circulation would cause a reactor scram, but the ECS system would have to be activated to prevent melting fuel by the remaining decay heat. The addition of H_2O from the ECS would force a moderator-H_2O mixture out of the three pressure relief systems in the reactor tank. As a result of this accident radioactive moderator is released to the reactor building through the pressure relief ports. Even if one of the three ECS lines is inoperable, no fuel melting will occur. The reactor is shut down by numerous flow sensors. The ECS is activated manually by procedural response.

The pre-accident reactor power is adjusted to limit fuel melting to 1 percent of the reactor core. In this sense, the loss-of-circulation accident is considered in deriving reactor power limits. However, detailed analysis of this accident shows that the reactor power that would limit fuel damage to 1 percent is higher than the reactor power prescribed by other limits. Thus, no melting may occur, but radioactive moderator would be released to the reactor building and the environment. The analysis of this accident is also used to define function specifications for the operation of the ECS.

Emergency sump pumps and dams are provided to minimize the possibility of flooding of the motor room.

Another postulated mechanism for losing D_2O circulation is freezing of D_2O in the heat exchanger due to extremely low cooling-water temperature. Operating procedures specify recirculating effluent water if the river water temperature should drop to 5°C. The D_2O freezing point is 3.8°C, and on one occasion (over 30 years of operation), the temperature of the Savannah River came close to this value. But if the water temperature drops below 5°C, some of the hot-water effluent is recirculated to the water in the basin to keep the inlet temperature from falling below 5°C.

G.4.1.19 Loss of cooling during and after discharge

Irradiated fuel and target assemblies are discharged in air and transported by crane to the discharge canal. If the crane becomes disabled, emergency cooling would be required to prevent melting and release of fission products. It is assumed that the crane becomes disabled. This accident is considered credible but improbable. If all four addition paths of emergency cooling to the discharge machine should fail at the same time a discharge machine interruption occurs, melting of fuel could take place.

G.4.2 Disassembly-basin accidents

The melting of irradiated fuel or target components in the disassembly basin is considered to be highly unlikely. Assemblies are not discharged from the reactor until the calculated heat generation rate is low enough to assure adequate cooling, even if the assembly is dropped to a horizontal position. In most cases, the heat generation in fuel and target assemblies immediately after reactor shutdown is sufficiently low that no cooling-off period is required. Should some unexpected assembly damage occur, the radioactivity would be released under about 10 meters of water. The affected basin section would be isolated and the water in that section would be circulated through deionizers and sand filters. Although the disassembly area is not part of the confinement system, most of the airborne release would be filtered by the confinement system as the air from personnel areas is drawn into the lower-pressure process areas and exhausted. In addition the high partition coefficient for iodine in water would cause the majority of the iodine released from the assembly to remain in the water, and no particulates would escape to the atmosphere.

A criticality accident is also an unlikely possibility in the disassembly area; such an accident is strongly guarded against by mechanical and administrative controls. If such a criticality did occur, it would typically involve from 10^{15} to 10^{20} fissions and occur under 10 meters of water. Offsite effects would not be expected to be measurable.

G.5 RADIOLOGICAL CONSEQUENCES OF REACTOR ACCIDENTS

The range of accidents considered for L-Reactor safety has been discussed in Sections G.1 and G.4.1. This section discusses how radioactivity released by accidents may affect the public. The sources of a radioactivity release are discussed first. Then the calculation techniques and finally the results of the calculations are presented.

The spectrum of conceivable SRP reactor accidents covers the range from trivial to severe. Four specific accident cases are cited to illustrate a range of accidents (up to 3 percent damage of the core).

G.5.1 Sources of a radioactive release

G.5.1.1 Isotopes released and manner of release

The sources of radioactivity considered in this section are tritium in the heavy-water moderator and fission products in the fuel. Potential offsite doses from nonfission product isotopes (cobalt-60, plutonium-239, etc.) are considered in Section G.5.5.

G.5.1.1.1 Moderator radioactivity

This report uses a conservative value of 5,000,000 curies of tritium present in the moderator. This is 30 to 40 percent higher than actual present values in currently operating reactors. The tritium is a natural consequence of neutron capture by deuterium. This tritium could be partially released to the confinement system following ECS actuation or any LOCA.

Any tritium becoming airborne in the confinement system would be discharged from the stack, because the confinement system has no mechanism for tritium removal.

It is quite unlikely that the full moderator inventory of tritium would evaporate and diffuse into the confinement system following any accident because the moderator would flow into the two holding tanks of the liquid activity confinement system. It is estimated that no more than about 3 percent of the tritium would evaporate during the initial 2-hour period after the postulated accident.

G.5.1.1.2 Radioactivity available for release from core melting

If any fuel or target assemblies melt, fission products become available for release. Depending on the type of assemblies melting and other circumstances the radioactivity release would include noble gases (xenon, krypton), iodine, and radioactive particulates (fission products, cobalt-60, plutonium-239, etc.). The concentration of these isotopes in the core is a function of reactor power that might reach a maximum of 3000 megawatts. Table G-10 lists the total inventory of fission products. Most of these isotopes decay rapidly following shutdown; and depending on the expected accident sequence, some isotopes may not contribute significantly to potential offsite doses.

The inventory of noble gases and iodine contributing to offsite dose is shown in Table G-7 and Figure G-2. Tritium is present in lithium-containing assemblies and control rods; up to 12 megacuries of tritium may be present in plutonium-producing charges. The radioactive particulates include several different isotopes and would be captured by the HEPA filters. The amount of particulates that would penetrate the filters would not contribute significantly to the offsite dose (Cooper and Rusche, 1968; Durant et al., 1966).

Table G-7. Iodine and noble gas inventory of 3000-MW core (major contributors to 2-hour offsite dose)

Isotope	Inventory (MCi)	Isotope	Inventory (MCi)
I-131	75	Kr-87	35
I-132	115	Kr-88	75
I-133	175	Xe-133	165
I-134	180	Xe-133m	25
I-135	165	Xe-135	20
		Xe-135m	30

Three releases of radioactivity from the core are considered as credible in this EIS; they involve melting of a single fuel assembly, 3 percent damage of the core during a reloading accident, and 1 percent damage of the core during a LOCA. TC

Melting of a single assembly during discharge

The fission products in the assembly would have decayed significantly between shutdown of the reactor and the discharge operation. Fourteen hours of decay of fission products is assumed as the minimum time to satisfy other discharge constraints as discussed in Section G.4.1.19.

The reactor room emergency spray system would be used to cool a hot assembly that drops to the reactor room floor to prevent melting. If melting occurred, the spray water would keep much of the iodine and particulates from becoming airborne. No credit is taken for this, however, and 50 percent of the iodine and 100 percent of the noble gases available for release are assumed to escape the assembly and become airborne. The iodine that reaches the carbon bed is assumed to be all elemental iodine because of the high air flow and rapid transport of iodine to the carbon beds (Durant et al., 1966). These parameters are also assumed for all accidents described in the following sections.

Core melting during a reloading accident

A criticality resulting from a reloading accident is postulated to cause some melting of the core (Section G.4.1.8). Core damage would be less than 3 percent for this accident. The melting could release fission products into the moderator. For purposes of analysis, it is assumed that 50 percent of the iodine and all the noble gases become airborne. Prior to the accident, the fission products would have decayed for a minimum of 14 hours. To be conservative, no credit is taken for decay prior to the accident.

Core melting during a loss-of-coolant accident (LOCA)

The LOCA is described in detail in Section G.4.1.17. If the worst conceivable D_2O pipe break were to occur, the emergency cooling and confinement systems would control offsite doses well within the 10 CFR 100 reference values, even with failure of a single active component in the emergency cooling system. No more than 1 percent core damage is expected in the worst-case LOCA. This accident is analyzed assuming 1 percent of the core inventory of noble gases and tritium and 0.5 percent of the iodine (50 percent of that available for release from the core) becomes airborne.

G.5.1.2 Release of radioactivity

In the moderator spill accident, tritium is released to the confinement system and then discharged from the 61-meter stack. This is assumed to occur over a 2-hour period. Only a small part of the tritium would actually be released; the rest would remain in solution in the two (225,000-liter and 1.9-million-liter) holding tanks. It is conservatively assumed that about 3 percent of the tritium evaporates.

For the accidents in which assemblies are assumed to melt, the amount of fission products released is proportional to the fraction of the core that melts. Noble gases and iodine are assumed to be released into the process room. Any gases vented to the blanket-gas system would eventually be released into the confinement system. It is estimated that 1 percent of the particulates (fission products, plutonium isotopes, etc.) would be released into the building, and half of that would reach the filters (Cooper and Rusche, 1968; Durant et al., 1966). Some 99.95 percent of the iodine and 99 percent of the remaining particulates would be captured by the activity confinement system. In the event of a dropped assembly, the reactor room spray system could remove much of the airborne iodine (and particulates) and some of the tritium before they left the process room, but this was not considered in the accident analyses.

Following a postulated melting accident, all noble gases are assumed released from the stack. In comparison with other doses, the released solids are considered insignificant (Cooper and Rusche, 1968). Some of the iodine trapped on the carbon bed would be desorbed as the result of the high radiation field generated by the decay of radioactive iodine. The desorption rates, shown in Figure G-3, are used to calculate potential offsite doses as discussed in Section G.5.2.

G.5.2 Calculation of offsite dose

This section describes the techniques used to calculate offsite doses resulting from reactor accidents. The calculations and data are consistent with NRC guidelines for accident analysis (NRC, 1972; 1979). The methods discussed were used for analysis of all accidents, including the moderator spill and fuel melting accidents.

G.5.2.1 Dose calculational method and criteria

There are three parameters necessary to compute the offsite doses. First, the radioactive source term must be specified, including the release rate and isotope type. Second, the transport of the isotope to the receptor location 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 at the plant boundary are computed based on parameters of a standard man (including breathing rates) and additional parameters related to absorption of energy from a particular isotope.

Individual characteristics, time of exposure, and meteorological behavior are important variables that are generalized in computing a maximum individual dose. In an actual accident, the WIND computer system of SRP would predict the release path and indicate appropriate action to minimize exposure to people offsite (Garrett et al., 1981). Evacuation procedures, which would reduce the actual dose to an individual, are not considered in these dose calculations (Garrett and Murphy, 1981).

The doses are reported both for 2-hour exposures and for 120-hour exposures. The 120-hour exposure represents a time after which further exposure would not significantly change the overall dose.

The dose calculation uses median meteorology. This and other parts of the calculation are discussed in the following subsections.

G.5.2.2 Source terms for radioactivity releases

The maximum amount of radioactivity available for release following the postulated moderator spill or assembly melting accidents was described in Section G.5.1.

The release from the stack is assumed to propagate over a 2-hour period in one direction as a Gaussian plume, and the exposure of an individual is treated as a time-integrated calculation. 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 (Langley and Marter, 1973).

The 2-hour irradiation period begins when the radioactive material reaches the Plant boundary. Both the noble gas and iodine source terms are assumed to decay during transport.

The source terms for iodine are the amount that would penetrate (or bypass) the filters and desorb from the charcoal in the first 2 hours and the first 120 hours following the incident. The average iodine retention efficiency assumed for the carbon is that for carbon aged 19 months, 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.

G.5.2.3 Transport of release and dose calculation

The downwind concentrations of iodine, tritium, and noble gases were calculated according to an integral technique using the computer code NRC145-2 (Pendergast, 1982a,b). This code was developed at Savannah River Plant and uses a Gaussian plume model based on NRC Regulatory Guide 1.145 (1979).

The meteorological data used in the dose calculations were collected from January 1975 through December 1979 (Garrett and Hoel, 1982). 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, 6 wind speeds, and 7 stability classes. (Stability classes were based on the standard deviation of the mean wind direction).

Median meteorological conditions (50th percentile) were assumed in these calculations. The effects of other less probable meteorological assumptions are shown in Figure 4-9.

Corrections for topography and jet rise of the released plume are applied. The topography correction is prescribed by the regulatory guide (NRC, 1972) and reduces the effective stack height when the downwind terrain is higher than the ground level elevation at the point of release. The jet rise of the plume occurs because the high volume exhaust fans (continuously online) impart a momentum to the gases going up the stack and increase the effective height of the release point. The model for jet rise that is included in NRC145-2 is described in Huber (1981).

The effect of fumigation, a condition that depresses downwind plume elevation to below the release height, was not included. The long distance from the release point to the site boundary makes local fumigation insignificant. Wind shear had no effect on atmospheric mixing at a distance corresponding to the plant boundary.

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 and independently verified (Pendergast, 1982c).

The thyroid dose and the whole-body dose are each composed of an inhalation component from iodine and 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 (Pendergast, 1982a; Cooper, 1972).

G.5.2.4 Dose conversion

The transport of the radioactive release to the plant boundary is calculated using the above techniques. At the boundary the diluted radioactive material is assumed to expose a standard man. To determine the dose received, calculational methods and parameters were used that were consistent with techniques described in Pillinger and Marter (1982). For iodine and tritium, a standard man's breathing rate was used to calculate an inhalation dose. The dose conversion factor considers skin absorption as well as inhalation in the case of tritium.

G.5.3 Results of calculations

The bases and assumptions for both the radioactive source terms and the methods for computing the transport to the plant boundary were described in Sections G.5.1 and G.5.2, respectively. The doses for the four accidents considered are discussed below.

G.5.3.1 Dose from moderator spill

As discussed in Section G.5.1.1.1, this accident considers the tritium dose when moderator is displaced from the reactor (e.g., due to actuation of the ECS). The calculation assumes a release of 0.15 megacurie (about 3 percent of the assumed 5-megacurie tritium inventory in the moderator) over a 2-hour period. The calculated dose to an individual at the plant boundary is shown in Table G-8.

G.5.3.2 Dose from core melting

As discussed in Section G.5.1.1.2, three melting accidents are considered.

G.5.3.2.1 Dose from melting a single assembly during discharge

This accident, discussed in Section G.5.1.1.2, assumes 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 G-8.

Table G-8. Calculated radiation dose to a person at the SRP site boundary following four specific accidents (50-percent meteorology and 3000 Mw power)

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 (about 3% core damage)	0.39	0.51	1.5
LOCA (1% core damage)	0.13	0.17	0.50

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

G.5.3.2.2 Dose from core melting during reloading accident

As discussed in Section G.4.1.8, calculations indicate that the maximum hazard would involve less than 3 percent core damage. It is assumed that the fission product content of the core is the equilibrium concentration that would be obtained at full power. Even with this assumption, the computed dose at the plant boundary is small relative to the DOE radiation standards for normal operation (see Table G-8).

G.5.3.2.3 Dose from 1 percent core damage during a loss-of-coolant accident

As discussed in Section G.4.1.17, this accident assumes that a massive double-ended pipe break occurs. For conservatism, the break is assumed to be in one of the lines used by the ECS addition system, so that one of the ECS addition systems is incapacitated. It is further assumed that a valve does not open in one of the remaining ECS addition systems. Thus, only one of the three ECS addition systems is assumed to work as designed. The doses (Table G-8) are below the DOE radiation standards for normal operation.

G.5.4 Summary of dose calculations

In summary, the offsite doses listed in Table G-8 were calculated in accordance with accepted methods and assumptions for environmental impact statements (in contrast to the more conservative analyses employed in Safety Analysis Reports). The whole-body and thyroid doses for these postulated accidents are less than the DOE annual radiation protection standards for normal operation. TC

G.5.5 Particulates (both fission product and nonfission product isotopes)

The potential offsite dose from nonfission product isotopes (e.g., cobalt-60, polonium-210, and plutonium-238) that may be present in large quantities in the mixed-lattice charges has also been considered. Few, if any, of these isotopes will be present in sufficient concentrations to generate sufficient heat to melt the target. Hence, major releases of the product materials in mixed-lattice charges would be expected to occur only in conjunction with a major reactor accident.

For calculational purposes, it is assumed that in an accident the fractional release of the nonfission product isotopes to the building environment, transport in the reactor building, and removal by the activity confinement system will be identical to the behavior of particulate fission products discussed in Brown (1971), namely, 1 percent of the inventory in the damaged portion of the core is released to the building, 50 percent of the released fraction is deposited in the building before reaching the activity confinement units, and 99 percent of the remainder is collected by the units. For the maximum credible core damage of 3 percent, the assumed net release fraction from the damaged core portion is thus 5×10^{-5}.

To provide an estimate of the relative magnitude of the potential offsite effects of several isotopes, full-charge inventories of several possible products have been calculated. The core inventory of several typical isotopes is shown in Table G-9. The inventories are based on the production capability in a single reactor (except plutonium-238 inventory, which is based on the availability of intermediates as feed material). Lesser amounts would be present in mixed lattices involving the production of several isotopes simultaneously.

Table G-9. Potential offsite doses from nonfission product isotopes[a] (50-percent meteorology)

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Isotope	Maximum inventory (megacuries)	Amount inhaled[b] during 2-hour exposure at plant boundary (μCi)	50-year dose commitment (rem)	Critical organ
Co-60	230	2×10^{-1}	1×10^{-1}	Lung
Po-210	38	3×10^{-2}	8×10^{-1}	Lung
U-233	0.0005	3×10^{-7}	2×10^{-5}	Lung
Pu-238	0.45	3×10^{-4}	8×10^{-1}	Bone
Pu-239	0.022	2×10^{-5}	4×10^{-2}	Bone
Cm-244	0.25	2×10^{-4}	1×10^{-1}	Bone

[a]These numbers generally are based on reactor charge producing a single product (the exception is plutonium-238). If two or more products are being produced simultaneously in the same reactor, the maximum inventory of any one would be lower.

[b]Release fraction of 5×10^{-5} for all isotopes.

Values of potential doses from a postulated accident releasing 3 percent of the core inventory were computed for each isotope by multiplying the curies released by the relative concentration (χ/Q) and an appropriate dose conversion factor. The calculation was similar to the inhalation dose calculations described in Section G.5.3. The assumed breathing rate was 3.47×10^{-4} cubic meter per second.

The quantity of each isotope that might be inhaled by a receptor at the plant boundary was calculated using the method described in Section G.5.2. Fallout, deposition, and decay in transit to the plant boundary were neglected. The calculated amount of each isotope inhaled is shown in Column 3 of Table G-9, assuming all of the aerosols reaching the boundary are of respirable size. The fractional release for all isotopes is based on 3 percent damage.

There are no official guidelines related to the inhalation of isotopes in a short time (as in a reactor accident). The dose conversion factors for chronic inhalation (Pillinger and Marter, 1982) were used to compute the potential 50-year dose commitments shown in Table G-9. The most restrictive dose conversion factors were used to determine the critical organ that received the highest dose. Thus the insoluble form of cobalt-60, polonium-210, and uranium-233 was assumed with the lung as the critical organ. The soluble form of plutonium-238, plutonium-239, and curium-244 was assumed with the bone as the critical organ.

The whole-body dose from noble gases present in the fuel in the same reactor charge is not included in Table G-9. The whole-body irradiation from exposure to gamma emitters would be added to the doses received from inhalation of particulates.

G.5.6 Review of Severe LOCA Scenario

A loss-of-coolant-accident (LOCA) is defined as a leak of heavy-water coolant from the reactor's primary cooling system. No fuel melting is expected in any probable LOCA. The rate of leakage in a LOCA could range from a trickle at a flange to a major discharge if a large pipeline should experience a rupture. An emergency cooling system (ECS) is provided to add water to the reactor to cool the core in case such a leak occurs. For conservatism, the ECS design provides sufficient flow to cool the core completely for the most severe leak that can be hypothesized. No reasonable mechanism has been identified that can cause a leak of this magnitude. For smaller, more probable leaks, the ECS would supply coolant far in excess of that needed to cool the core.

The heavy water in the SRP reactors gradually builds up small amounts of radioactive tritium from neutron activation. If part of the tritium evaporates, some would mix with the reactor building atmosphere and pass to the environment via the 61-meter exhaust stack. Assuming conservatively that there is 3 percent of heavy water evaporated and that it contains a maximum tritium content, the maxium dose from exposure to tritium to a person at the Plant boundary would be 0.007 rem.

If the ECS were activated, it would flood the reactor cooling system at a rate of up to 53,000 liters per minute, causing the heavy-water primary coolant to be displaced into sumps from which the heavy water would be pumped into two holding tanks that are vented to the reactor building. The first holding tank has a capacity of 225,000 liters and would retain initially all of the displaced heavy water from the reactor. The second holding tank has a capacity of 1.9 million liters. Following an accident, the ECS flow could be reduced gradually as the leak is isolated and the residual decay power in the reactor decreases. If the leak is isolated promptly, as expected in most cases studied, the holding tanks would not be completely filled. Otherwise, the holding tanks might be filled in a few hours. In the unlikely event that the ECS flow would have to continue beyond the time the holding tanks are filled (2.1-million-liter capacity), the water from the reactor would be river water with little or no tritiated heavy water expected. This water would then bypass the holding tanks and flow to the 190-million-liter excavated basin. Some additional tritium release to the atmosphere might occur; it would, however, be very small.

Even if only one of three ECS supply lines functions properly (i.e., if the LOCA occurred in one of the lines and if valves in a second line failed to open), no melting would be expected for the more credible leak rates. For the hypothetical maximum leak rate, it has been estimated that as much as 1 percent of the core might become overheated and possibly melt in the first minutes of an accident while the decay power is high. In the event of such melting, some radioactive fission products--particulates, volatile noble gases and radioiodine--would be released from the fuel and swept along with the ECS flow. The particulates and soluble radioiodine would be carried to the holding tanks where they would be confined. Noble gases and volatile radioiodine would tend to enter the building or confinement tank and pass into the confinement filter system. More than 99 percent of the radioiodine would be absorbed on the carbon beds provided for that purpose. However, noble gases would be released to the environment. The estimated radiation exposure to the maximum individual at the plant boundary would be approximately 0.1 rem whole body and 0.5 rem to the thyroid.

As noted above, if ECS flow continues beyond the time at which the 2.1-million-liter tanks are filled, any additional discharge would bypass the holding tanks and enter directly into the 190-million-liter basin. Because possible melting and fission-product release would have occurred early in such a transient, river water entering the earthen basin after the holding tanks were filled would have passed through a well-cooled and well-flushed core. That river water would be expected to carry only a minimal quantity of fission products and other contamination into the earthen basin. No additional risk is attributed to this accident because the metallic fuels used in SRP reactors will resolidify when cooling is restored; there is an extremely low probability of delayed core damage after the ECS flow has been established and the confinement tanks have been filled.

Therefore, no radioactive material, except some tritiated moderator, would be released as a result of any expected LOCA (no melting occurs). For the more severe hypothetical and improbable case of a 1-percent core heatup and melt following a LOCA, most fission products, except noble gases and small amounts of tritium and radioiodine, which could escape from the core, would be contained within the reactor building and the holding tanks.

While there has never been a major accident to challenge the confinement system, the system was developed on the basis of a comprehensive experimental program. Routine performance tests of the confinement system are conducted regularly. Furthermore, when a source rod melt at one of the SRP reactors did challenge the major features of the system in 1969, it responded perfectly. The system is always on line (i.e., ventilation air is continuously drawn through the filters by three fans powered by two independent motors with automatic backup power supplies). Only one operating fan is required.

The confinement system ventilation air first passes through demisters that remove any water droplets, allowing the HEPA and carbon filters to operate at maximum efficiency. The effect of radioiodine overloading causing carbon to overheat has been studied extensively. Even for a maximum loading associated with a theoretical 100-percent core meltdown, the air flow from a single fan is sufficient to keep the carbon from overheating. For the postulated worst hypothetical accident of a 3-percent core melt, the margin on overheating would be much larger.

Because carbon is less effective in absorbing and retaining organic iodide compounds compared to elemental iodine, SRP has developed special impregnants for the carbon used in the confinement system. These impregnants improve the capacity of the system both to absorb and to retain organic iodide. Furthermore, the nuclear power industry is developing a considerable body of evidence that radioiodine released from fuel elements would be largely in nonvolatile forms that would stay dissolved in water or tend to remain inside the reactor vessel and the reactor building. Because of these phenomena, little volatile radioiodine was released to the reactor building during the TMI-2 accident. The Savannah River Laboratory is engaged in a research program to quantify these effects. The conclusion is that no mechanism exists by which a large portion of the iodine would be converted instantaneously to organic compounds in an accident; the effect of organic radioiodine release through the confinement system is not a significant dose factor. |TE

The potential for steam or hydrogen explosions in an accident has been analyzed; the impact of such explosions on the confinement system has been assessed. For more credible accidents, the amount of fuel damage is so small as to preclude the potential for such explosions. For the more severe hypothetical accidents, the confinement system has the capacity to accommodate the hypothetical gas or energy releases. If hydrogen were formed during an accident, it would be swept from the building by the high ventilation flow of the confinement system before explosive hydrogen concentrations could be reached. This sweepout is in contrast to a closed containment where a buildup of hydrogen gas could threaten the containment integrity in certain hypothesized accidents. The nuclear industry is considering how to deal with this threat. One option being considered, and already adopted in Sweden, is a filtered, vented containment incorporating many of the features of the SRP confinement system.

G.5.7 Improbability of fission product release

As discussed in previous sections, release of fission products to the environment would first require an initiating event to challenge the physical barriers and safety systems provided to prevent such a release, and then a breakdown or failure of these barriers and systems. Such a sequence is improbable. Although probability values are not precisely known for the rare events being considered here, estimates can be made for illustration. Several sequences using estimated or bounding probability values are discussed in this section for two of the accidents analyzed in Section G.4.1. A more complete probabilistic risk assessment study of the entire spectrum of accidents is under way.

G.5.7.1 Hypothetical D_2O pipe break

An abrupt double-ended break of a major D_2O pipe is discussed in Section G.4.1.17. It is not considered to be a credible accident because an abrupt catastrophic failure that allows unimpeded leakage from both sections of pipe is believed to be impossible with stainless steel pipe. However, the frequency of some type of large pipe failure has been previously estimated at 1×10^{-4} to 1×10^{-5} per reactor year. The log mean of this range, 3×10^{-5}, is assumed to be the upper bound of probability of the maximum possible pipe break, which is the initiating event of the sequence shown in Figure G-4. This event challenges the shutdown systems, the Emergency Cooling System (ECS), and possibly the Airborne Activity Confinement System (AACS). The shutdown systems have a very high probability of working, and are excluded as a failure mode in the sequence shown in Figure G-4. The ECS has a high probability of working, which leads to the most probable and least harmful outcome of the sequence, namely, a moderator tritium release, but no fission product release. But the ECS can experience partial or total failure; analysis of ECS failure modes lead to the probabilities shown in Figure G-4. These failure modes lead to less probable but larger releases of fission products. For total failure of ECS, the AACS is protected by the Confinement Heat Removal System (CHRS). A probability of failure of 0.5 is assumed for this illustration. The probability of outcomes that lead to larger releases of fission products is extremely small, as shown in Figure G-4.

G.5.7.2 Control rod withdrawal accident

The control rod withdrawal accidents are discussed in Section G.4.1.3. These accidents challenge the shutdown systems and possibly the AACS. The gang rod withdrawal is more challenging but less probable, and the sequence is illustrated in Figure G-5. No such event has occurred in over 115 reactor-years of operation, and this establishes an upper bound of an occurrence, with 95-percent confidence, of 3×10^{-2} per reactor year. The safety rod scram system and the automatic backup shutdown system (ABS-S/C) have a high probability of working, and success of either one leads to an outcome with negligible fission product release. Failure of both systems would lead to an undefined amount of core melting, damage to the reactor structure, and ejection of steam into the process room. Even so, there is a good, but undefined, probability that the AACS would contain most of the iodine (but release noble gases and tritium). The probability of significant or large fission product release is very small, as shown in Figure G-5.

G.5.7.3 Total risk from all postulated reactor accidents

To provide a perspective on the overall accident risk of L-Reactor operation, Figure G-6 is a preliminary total probability curve that presents the annual probability of a resident living at the SRP site boundary receiving more than a certain dose from postulated accidents. These results are based on accident analyses presented in the Safety Analysis Report (Du Pont, 1983a), 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. Six different accident initiators were considered. For all the accidents, the most probable outcome is 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). These accident sequences were as follows: EN-27

1. A loss-of-coolant accident with only one operable ECS.

2. A loss-of-coolant accident with a total failure of the ECS.

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 affects 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 G-8 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 G-7. Doses for postulated core damage of 10 and 100 percent are, respectively, 10 and 100 times 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
N-27 dose. This overall dose probability curve is shown in Figure G-6. The results are consistent with (1) the decreasing frequency of meteorological conditions that give higher doses for any accident (Figure G-7), 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 G-6 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, etc.) 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 eleven accident sequences discussed above support earlier evaluations, made for worst-case scenarios using single failure criteria, which concluded there is negligible risk to public health and safety.

G.6 CONSEQUENCE ANALYSIS FOR A 10-PERCENT CORE MELT

Any accident resulting in damage greater than the maximum calculated for the previously discussed accidents (3-percent core melt) is highly improbable. However, in order to assess the consequences of core-melting for which no reasonable mechanistic scenario can be conceived, a 10-percent melt accident (more than three times as severe as the worst accident previously considered) is postulated. Based on the discussion for the lesser consequence accidents, the probability of a 10-percent core melt would be considerably lower than 10^{-6} per reactor year.

The consequence analysis for a 10-percent core-melt accident has been carried out with the CRAC2 code (Ritchie et al., 1981). This is a revised version of the code CRAC (Calculation of Reactor Accident Consequences) which was developed for use in the Reactor Safety Study (NRC, 1975). The organization of CRAC2 is given in Figure G-8.

This section of the appendix summarizes the input data used for CRAC2 analysis. The results of this analysis are presented in Section 4.2.1.5 and summarized in Table 4-24.

Curies of fission products and actinides released to the atmosphere

The amount (curies) of each radionuclide released to the atmosphere for each accident sequence is obtained by multiplying the release fractions by the amounts that would be present in the core at the time of the hypothetical accident.

For a 10-percent core-melt accident, the release fractions are 0.1 for the noble gases, 5×10^{-6} for the particulates and 1.66×10^{-3} for the iodines. Included in the iodine release fraction is the 120-hour desorption from a 30-month service aged carbon filter bed.

The fission product inventory in any SRP reactor charge varies with the reactor charge, the irradiation history, and the operating power level. For purposes of consistency and conservatism, a 3000-megawatt operating power level and saturation inventory of the important fission gases was used. The inventory values were calculated using the Du Pont SHIELD code (Finch, Chandler, and Church, 1979) for single assemblies of both Mark 16 and Mark 31A in the highest power zone of the reactor at the end of the first subcycle. The specific power was 6 megawatts per assembly for the fuel and 2.88 megawatts per assembly for the target. Three hundred assemblies of each type were assumed to obtain a total power of 2664 megawatts. Individual assembly inventory values were then corrected by the factor (300)(3000)/(2664) to obtain full core inventory values for each assembly type. For all short-lived (half-life less than 45 days) isotopes the values thus obtained are saturation inventory values. For long lived isotopes (half-life greater than 225 days), the SHIELD code values for the fuel tubes were multiplied by 5 to obtain the approximate inventory at the end of 5 subcycles. For isotopes with half lives of between 45 and 225 days, the standard buildup decay equations were used to obtain an equilibrium inventory at the end of 5 subcycles. Since targets are not recycled, no correction is necessary for Mark 31A assemblies.

The equilibrium isotopic inventory for important radionuclides is tabulated in Table G-10. The radionuclides in this table are the same as those used in the Reactor Safety Study (NRC, 1975). The elimination of radionuclides from consideration in radiation dose calculations was based on a number of parameters, such as quantity (curies), release fractions, radioactive half-life, emitted radiation type and energy, and chemical characteristics.

Meteorological data

The CRAC2 input data file contains a full year of consecutive hourly values of windspeed, wind direction, stability class and precipitation. These were processed from measurements taken at the K-Area meteorological tower during the year 1978. Hourly precipitation data for Augusta, Georgia, was obtained from the National Weather Service. The stability category was determined by using the sigma-theta's from the K-Area meteorological data file.

Prior to sequence selection, the entire year of weather data was sorted into 29 weather categories (termed "bins"), as defined in Table G-11. Each of the 8760 potential sequences was first examined to determine if rain occurs anywhere within 50 kilometers of the accident site. If not, a similar examination was made for wind speed slowdowns. If neither of these conditions occurred, the sequence was categorized by the stability and wind speed at the start of the accident. A probability for each weather bin was estimated from the number of sequences placed in the bin. Sequences were then sampled from each of the bins (with appropriate probabilities) for use in risk calculations, assuring that low probability adverse weather conditions were adequately included (four sequences were selected from each bin in this current analysis). The proposed technique also allowed the use of wind direction statistics for specific weather conditions.

Population distributions

The population distribution around the site has been assigned to a grid consisting of 16 sectors, the first of which is centered on due north, the second on 22-1/2 degrees east of north, and so on. There are also 28 radial intervals as shown in Table G-12, which contains the predicted permanent resident population for the year 2000.

Evacuation modeling and other protective measures

In this assessment, no evacuation and special sheltering measures were assumed.

Other countermeasures

The other protective actions include (1) either complete denial of use (interdiction) or permitting use only at a later time after appropriate decontamination of crops and milk; (2) decontamination of severely contaminated land and property when it is considered to be economically feasible to lower the levels of contamination to protective action guide levels; (3) denial of use (interdiction) of severely contaminated land and property for varying periods of time until the contamination levels are reduced by radioactive decay and weathering to such a level that decontamination is economically possible as in (2) above.

These actions would reduce the radiological exposure to the people from immediate and/or subsequent use of or living in the contaminated environment. In CRAC2, these protective actions are modeled in the same way as in WASH-1400 (NRC, 1975).

Exposure pathways

The exposure pathways modeled by CRAC2 are the following. First, there is inhalation of radioactive material from the passing cloud. The inhalation dose conversion factors, which relate the curies inhaled to the subsequent radiation dose to various body organs, remain the same as those used in the Reactor Safety Study and are contained in the standard CRAC2 data file. Second, there are cloudshine and groundshine, the irradiation of body organs by gamma rays emitted by the passing cloud or by fission products deposited on the ground. The cloudshine and groundshine dose conversion factors also remain the same as in the Reactor Safety Study and are contained in the CRAC2 data file. Third, there are chronic exposure pathways, which include (1) resuspension of deposited radioactive material by the wind; (2) long-term exposure to gamma rays from deposited fission products, especially cesium, including the effects of weathering; (3) consumption of milk; (4) consumption of milk products; (5) consumption of contaminated vegetation; and (6) consumption of crops contaminated by root uptake. The treatment of these chronic exposure pathways remains precisely the same as in the Reactor Safety Study.

Health effects

In CRAC2, the calculation of the health effects caused by radiation doses delivered to various organs is still handled in virtually the same way as was done in the RSS. The health effects model in CRAC2 is based on the BEIR (1972) report of the National Academy of Sciences.

Economic costs

CRAC2 requires various elements of economic cost. These are generally in the form of a cost per person or a cost per acre, e.g., the cost of evacuating a person or of decontaminating an acre of land. The calculation of many of these costs is described in the Reactor Safety Study, Chapter 12, Appendix VI, where they are presented in 1974 dollars. Some allowance has, therefore, to be made for inflation and the CRAC2 manual contains 1980 values. Table G-13 contains a summary of important parameters. In general, it is three of these that dominate the out-of-plant property damage--the value of residential, business, and public areas; the relocation cost; and the decontamination costs. All other costs, including those for agriculture, are relatively unimportant.

Difference between CRAC and SAR analyses

As mentioned in Section 4.2.1.5, there are several differences between the CRAC2 methodology and those that were used to calculate the doses in Section 4.2.1.4. The most important difference is that CRAC2 considers more radiation dose pathways (e.g., doses from groundshine (from radioactivity deposited on the ground), inhalation of resuspended materials, ingestion of milk, milk products,

Table G-13. Economic input data

Parameter	Value (1980 dollars)	Comment
Decontamination cost for farm areas (for DF of 20)	$499 per acre	From CRAC2 Manual
Decontamination cost for residential, business and public area (for DF of 20)	$3349 per person	From CRAC2 Manual
Compensation rate per year for residential, business and public area	$6305 per person	WASH-1400, Appendix VI, para. 12.4.2.1
Value of residential, business and public areas	$31,527 per person	From CRAC2 Manual
Relocation cost	$4,344 per person	From CRAC2 Manual
Annual cost of milk consumption	$135 per person	From CRAC2 Manual
Annual cost of consumption of non-dairy products	$685 per person	From CRAC2 Manual
Evacuation cost	$165 per person	From CRAC2 Manual

and contaminated vegetation). Sensitivity studies show that these additional pathways could contribute an additional 50 percent of the total dose.

Other differences include the following:

- Meteorological data utilization.
- One-year (CRAC2) versus 5-year (SAR) meteorological data period
- Site boundary distances. In the CRAC2 analysis, the site boundary is defined as a radius of 13.7 kilometers. In Section 4.2.1.4, the actual site boundary is used.
- Iodine desorption rates. In the CRAC2 analysis, a 30-month aged iodine filter was assumed (with a 3.3-percent cumulative desorption; in Section 4.2.1.4, a 19-month aged iodine filter was assumed (with a 1.3-percent cumulative desorption).
- Population distribution. The CRAC2 analysis uses a population distribution for the year 2000; Section 4.2.1.4 uses the population distribution for 1980. Furthermore, the population distribution in the CRAC2 analysis extends to 800 kilometers rather than the 80-kilometer distribution used in Section 4.2.1.4.