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REACTOR OPERATION SAFETY INFORMATION DOCUMENT (U)

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1.0 REACTORS FACILITY DESCRIPTION

1.1 SRS Process Description

P, L, and K Reactors are part of an integrated SRS complex for the production of defense nuclear materials, including a fuel and target fabrication plant, five reactors (three currently operating), two chemical separations plants, a heavy-water production plant (on standby except for rework), and waste-storage facilities. This complex includes fabrication of fuel and target materials into elements and assemblies for loading into the reactors; irradiation in the reactors; separation of transuranic elements, tritium, and residual uranium from waste byproducts; heavy-water recovery and purification; and waste processing and storage. The Defense Waste Processing Facility (DWPF), now under construction, will immobilize high-level wastes currently stored in underground tanks.

The SRS fabrication plant manufactures fuel and target elements to be irradiated in the production reactors. Currently, its major products are extruded enriched uranium, aluminum-clad fuel; aluminum-clad depleted-uranium metal targets; and lithium-aluminum control rods and targets.

Each reactor building houses one production reactor and its supporting operational and safety systems. The reactor buildings incorporate heavy concrete shielding to protect personnel from radiation and a confinement system to minimize atmospheric radioactivity releases. The reactors use heavy water (D_2O) as a neutron moderator and as a recirculating primary coolant to remove the heat generated by the nuclear fission process. The recirculating D_2O coolant is, in turn, cooled in heat exchangers by water pumped from the Savannah River and Par Pond, a 10.7-square-kilometer impoundment. Figure 1-1 shows the reactor process system. The reactors produce plutonium by the absorption of neutrons in the uranium-238 isotope and tritium by neutron absorption in lithium -6. Rechargeable fuel and target assemblies all are clad with aluminum. These fuel and target assemblies are discharged from the reactors after a specified exposure period and stored in a water-filled disassembly basin to permit decay of short-lived radiation products.

The chemical separations plants dissolve the irradiated fuel and transuranic-bearing target materials in nitric acid. A solvent extraction process then yields (1) solutions of plutonium, uranium, or neptunium and (2) a high-heat liquid waste, containing the nonvolatile fission products. After the product solutions are decontaminated sufficiently from the fission products, further processing is performed and plutonium is converted from solution to solid form for shipment.

The tritium facilities are a complex of buildings in which tritium is separated from irradiated lithium-aluminum targets, further purified, and packaged.

In the processing system, the lithium-aluminum targets are loaded into furnaces, and the gases are extracted under heat and vacuum. The hydrogen isotopes are first separated from the helium isotopes, and then tritium is separated from protium (^1H) and deuterium (^2H). Solid wastes are packaged and shipped to the burial ground.

Heavy water for use as the reactor moderator was separated from river water at the heavy-water facility (now in standby except for rework) by a hydrogen sulfide extraction process and then further separated by distillation.

The liquid radioactive wastes produced from the chemical processing of irradiated fuel and targets are partially concentrated and stored in large underground tanks. The DWPF will immobilize the wastes from these tanks in borosilicate glass disposal forms. These solidified wastes will be stored onsite until their final disposal. Low-level radioactive solid wastes produced at Savannah River Site are disposed of in a centrally located burial ground.

1.2 Reactor Systems and Structures Descriptions

1.2.1 Reactor Sites

Figure 1-2 shows the location of the P, L, and K production reactors. Each of the three production reactor sites is roughly rectangular in shape, measuring approximately 1700 by 2400 feet. The production reactor in P Area is approximately 4.5 miles from the site's eastern boundary near the Barnwell County Industrial Park. The production reactors in L and K Areas are approximately 5.5 miles from the southern and western site boundaries, respectively. Site-specific maps of P, L, and K Areas are shown in Figures 1-3 through 1-5.

1.2.2 Reactor Structures

Figures 1-3 through 1-5 show the location of the major structures in each reactor area, which include the following:

- 105 Building - Houses the reactor and associated support systems; a fuel and target receiving, assembly, testing, and storage area; a pool for the storage and disassembly of irradiated fuel and target elements; and facilities for the purification of heavy-water moderator/coolant.
- 186 Basin - Receives and stores heat-exchanger cooling water pumped from the Savannah River. This basin has a 95-million-liter capacity.
- Retention Basin - Contains a 1.9-million-liter tank and collects cooling water discharged in the event of an accident.
- Office and shop buildings.
- Other Support Facilities - These facilities include two transformer yards, sanitary treatment facility, water treatment plant, radiological health protection, and security areas.

1.2.3 Reactor Systems

1.2.3.1 Reactor Vessel and Reactor Lattice

Each reactor vessel is a cylinder about 4.5 meters high and 5 meters in diameter made of 1/2-inch Type 304 stainless steel plate. Coolant enters through six nozzles at the top of the reactor into a plenum, flows down coolant channels in the fuel and target assemblies, and discharges into the bulk moderator. It leaves through six nozzles at the bottom of the reactor vessel (Figure 1-6). A gas plenum and top radiation shield are located under the inlet water plenum. Under the reactor vessel, a radiation shield containing 600 monitor pins provides flow and temperature monitoring for each fuel and target position. The vessel is surrounded by a 50-centimeter-thick water-filled thermal shield and a 1.5-meter-thick concrete biological shield.

Studies of the effects of neutron irradiation on the stainless-steel SRS reactor vessels concluded that the vessels have experienced no life-limiting 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. The operating temperature of the SRS reactor tank walls is too low for significant swelling to occur from voids or gas bubbles which might result from neutron irradiation. 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 is at a low level.

The reactor contains positions for 600 fuel and target assemblies; other principal positions in the reactor lattice are used for control rod housings, spargers, and gas port pressure-relief tubes. Interspersed among the principal lattice positions are 162 secondary positions which can be occupied by safety and/or instrument rods. In addition to the downflow coolant for the fuel and targets, upflow coolant is provided for the control rod assemblies and for mixing the bulk moderator.

Neutron flux in the reactor is controlled by neutron-absorbing rods in 61 positions; each position contains seven individually motor-driven control rods. These control rods can be moved in gangs (groups) for simultaneous positioning, or individually in sequence. Two half-length rods in each position control the vertical flux distribution; full-length rods control overall power and the radial flux distribution.

Process monitoring and reactor control is accomplished from a central control room. The reactor can be controlled manually by an operator or automatically by an online computer.

1.2.3.2 Primary Coolant System

Heavy water (D_2O) serves as both a neutron moderator and primary coolant to remove heat from the nuclear fission process. The heavy water is circulated through the reactor by six parallel pumping systems. In each system, about 1600 liters per second are pumped from one of six outlet nozzles at the bottom of the reactor, through two parallel heat exchangers, and into one of six inlet nozzles in the water plenum above the reactor. All components of the D_2O system, except the pump seals, are made of stainless steel. The reactors produce no electric power, which allows them to operate without the high temperatures and pressures needed in power reactors.

Each of the six circulating systems contains a centrifugal pump rated at 1600 liters per second at a total pressure head of 128 meters of water. Each circulating pump is driven by a 2500-kilowatt alternating-current (a.c.) induction motor drawing 125 amperes at full load. Pumps and motors are separated into groups of three in two pump rooms and two motor rooms. Each motor also drives a 2.7-metric-ton flywheel that stores enough energy to continue pumping heavy water for about 4 minutes if there is a loss of a.c. power. Power for the a.c. motors is supplied from either of two substations.

Backup pumping capacity for heavy-water circulation is provided by six direct-current (d.c.) motors; they are normally online when the a.c. motors are operating. If a.c. power fails, each d.c. motor will drive a pump to provide sufficient flow, enough to remove residual heat from the shutdown reactor. Each d.c. motor is connected directly to its own online diesel generator; two generators are kept in reserve.

Limits on pD (the heavy-water equivalent of pH), conductivity, and impurity levels of the heavy water are maintained to control the corrosion of aluminum and stainless steel and to reduce the decomposition of the heavy water. Sustained reactor operations at Savannah River Site have demonstrated that the corrosion rate of aluminum components and the associated problems of high radioactivity and turbidity in the process systems can be reduced substantially by controlling pD. To minimize aluminum corrosion, nitric acid is added to the heavy water through a pump suction line to maintain a heavy-water pD of about 5.2. Because some of the acid is neutralized as the process water flows through the purification deionizers (causing the pD to increase), periodic injections of nitric acid are necessary.

1.2.3.3 Secondary Coolant System

Each of the six heavy-water pumping systems contains two parallel, single-pass heat exchangers to transfer heat from the heavy water (primary coolant) to secondary cooling water drawn from the Savannah River and Par Pond and then discharged back to the Savannah River or Par Pond. Water is taken from the Savannah River at two pumphouses and one pumphouse on Par Pond, then delivered to each reactor area cooling-water reservoir (186-Basin) with flows at approximately 11 cubic meters per second. An alternate tie-line provides an emergency supply of cooling water from the river to the reservoir if the primary line from the river fails. Without a supply of water from the river, the reservoir can cool the reactor in the shutdown mode for 100 days by recirculation.

A pumphouse adjacent to the reservoir delivers water to the reactor building. If pumphouse power is lost, the options available to deliver water to the reactor building include (1) gravity flow from the reservoir through the pumphouse, (2) gravity flow from the reservoir to the emergency pump in the reactor building via a bypass line, (3) forced flow from the river pumphouses using a pipeline that bypasses the reservoir and delivers cooling water directly to the reactor building, (4) recirculation of reservoir water with the emergency pump, and (5) recirculation of disassembly-basin water with the emergency pump.

The effluent cooling water flows from the reactor building to the effluent sump. As much as 0.70 cubic meters per second can be recirculated. Normally, the water overflows a weir in this pump and flows to the Savannah River or Par Pond.

1.2.3.4 Core Reloading

New fuel is received and stored in the reactor assembly area. Racks and hangers maintain adequate spacing for criticality control; an additional safety margin for assemblies containing fuel is provided by storage in racks constructed of material that contains boron, a neutron absorber. Moderating materials are strictly controlled in the assembly area to prevent criticality. Procedural controls limit the type and amount of material in process at any time.

The equipment for core reloading includes an inlet conveyor, a charge machine, a discharge machine, and a deposit-and-exit conveyor. The charge and discharge machines are similar, and each can perform most of the functions of the other; however, only the discharge machine can provide heavy-water or light-water cooling to an irradiated assembly. Both machines travel on tracks on two parallel ledges that are part of the reactor-room wall; power for their operation is provided through cables along the ledges.

Reloading operations are conducted from a control room adjacent to the reactor control room. The charge and discharge machines can be operated manually or automatically via a computer control system. Graphic displays on the control console track the location and operation of the machines.

1.2.3.5 Fuel Discharge and Storage

Fuel and target assemblies are discharged from the reactor by the discharge machine. Four sources of water are available on the discharge machine to cool an assembly during the discharge operation—primary D_2O, primary H_2O, secondary D_2O, and secondary H_2O. The primary and secondary sources supply water through different paths to the assembly. Cooling starts automatically when an irradiated assembly is completely withdrawn from the reactor; it can also be maintained if an assembly sticks during withdrawal.

For each type of assembly, an upper limit is specified for heat-generation rate at the time of discharge; discharge of an assembly does not start until the heat-generation rate of the assembly has decayed to this upper limit.

The deposit-and-exit conveyor, located in a water-filled canal connecting the reactor room and the disassembly basin, receives an assembly from the discharge machine and carries it under the reactor room wall to a water-filled disassembly basin for temporary storage.

Irradiated assemblies are stored in the disassembly basin to allow radionuclides and heat to decay to a level low enough for shipment to the separations facilities. The assemblies are cooled by natural convection; hangers allow this cooling while maintaining adequate spacing for criticality control. The basin water also provides shielding of radiation from the assemblies. Procedural controls and instrumentation prevent shipment of insufficiently cooled assemblies.

1.2.3.6 Blanket-Gas System

The blanket-gas system, which uses helium (an inert gas), is the initial barrier to the release of radioactive gases from the reactor. This system has three primary functions: (1) to dilute deuterium and oxygen evolved from the moderator (due to radiolysis) to a nonflammable concentration, (2) to recombine the deuterium and oxygen constituents of the gases evolved to heavy water, and (3) to maintain the pressure in the moderator (pressurize the plenum of the reactor to about 34,000 pascals gauge (5 psig) and thus increase the heavy-water saturation temperature). Helium is used as the blanket gas because it neither reacts with moderator decomposition products nor absorbs neutrons to produce radioactive gases.

During operation, gases evolve from the reactor and enter the gas plenum. From the plenum, the gases are routed to catalytic recombiners and spray separators where the deuterium and oxygen are recombined and most of the entrained heavy water is removed from the helium and returned to the reactor. The helium is then returned to the gas plenum.

1.2.3.7 Airborne Activity-Confinement System

During reactor operation, the process areas are maintained at a pressure lower than the pressure of the external atmosphere to ensure that all air from the process areas is exhausted through the airborne activity-confinement system. As shown in Figure 1-7, the air from these areas is exhausted through a set of confinement filters before it is released to the 61-meter stack.

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. Fan motors can be powered by two electric sources:

(1) Normal building power, from at least two substations

(2) Emergency building power, from diesel generators

In addition, each has a backup motor; the backup motors for any two of the fans can be powered simultaneously by automatically starting diesel generators.

Exhaust filters remove moisture, particulates, and halogens. The filter banks are enclosed in five separate compartments, three to five of which are online during operation. Each compartment can be isolated for maintenance and/or testing; each contains filter banks, in the following 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 significant blinding of the particulate filters.

(2) Particulate filters—designed to retain more than 99 percent of all particulates with diameters of 0.3 micron or larger.

(3) Activated carbon beds—impregnated carbon designed to retain halogen activity.

The airborne activity-confinement system functions as a primary part of the engineered safety features and is discussed in more detail in Section 1.2.4.3.

1.2.3.8 Liquid-Radwaste System

The chemical purity of the moderator is maintained to minimize heavy-water radiolysis and to minimize the corrosion rate of aluminum and stainless steel in the reactor; in addition, moderator impurities absorb neutrons that otherwise would be utilized in the production of nuclear materials. The neutron activation of moderator impurities and corrosion products, along with any fission products released by fuel failures, contributes to the overall activity level in the moderator.

The moderator is continuously purified by circulation of a side stream to a purification area to be deionized and filtered. Most of this side stream is returned to the reactor; a small amount is distilled to remove light water (H_2O).

The purification system circulates about 1.9 liters per second through a pre-filter, a deionizer, and an after-filter. The deionizer contains deuterized cation and anion exchange resin. The filters retain particles larger than 10 microns in diameter.

The filters and deionizers are located in a shielded cell area. Radioactive impurities are concentrated in disposable filter and deionizer units. Vessels containing spent deionizer are remotely loaded into heavily shielded casks for transport to a facility for the eventual recovery of deuterium oxide. After processing, these vessels are sent to the burial ground for disposal.

Part of the reactor side stream is diverted to the distillation area for removal of light water.

An evaporator system removes particulate matter from deuterium oxide from the distillation column reboiler purge. When the deuterium oxide distillation columns are emptied for maintenance or repair, the water is either collected in a tank to be reused or drummed to be reworked at the heavy-water reconcentration plant.

Target and spent-fuel assemblies removed from the reactor are rinsed in the discharge machine. The rinse water is collected by the discharge machine-water pan and sent to the 2270-liter rinse collection tank. Rinse water is drummed and reworked.

Some radioactivity is transferred from the irradiated assemblies to the water in the disassembly basin, even after rinsing. Periodic purging of the basin water is necessary to reduce the radiation exposure to operating personnel from the accumulation of tritium. During the purging operation, water from the basin is passed through two deionizer beds in series and monitored before it is discharged to a low-level radioactive seepage basin. This process reduces the release of any radioactivity other than tritium to the seepage basin. The spent resin from the deionizer beds is regenerated in Building 245-H, and the spent regenerant is concentrated and stored in high-level radioactive waste tanks in the separations areas.

Two sand filters maintain the clarity of the disassembly-basin water. Particulate matter in the basin water tends to agglomerate and absorb radioisotopes. When the basin water passes through the sand filters, the particulate burden is reduced. The filtration rate can vary from 32 to 95 liters per second, depending on the initial fluid clarity and the demand for treatment. When the differential pressure across the filter beds indicated the need, a filter can be isolated and backflushed. Backflushed radioactive material is transferred to the chemical separations area for concentration and storage in high-level radioactive waste tanks.

1.2.3.9 Solid Radwaste

Contamination from induced activity accounts for most low-level solid waste. Work clothing, plastic sheeting, and kraft paper also become contaminated when they are used for occupational protection. Such material comprises most of the low-level waste; irreparable valves, pipe sections, pumps, instruments, and aluminum and stainless-steel reactor components also constitute such waste. Solid waste is packaged for disposal in the SRS burial ground.

1.2.3.10 Limit System

The reactors 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 that with a minimal level of emergency cooling system operability no fuel melting will occur.

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.

Each reactor charge is moderated and cooled by D_2O and has the same spacing between fuel and target assemblies. An accident analysis is performed for each charge. Some of the analyses can be generic in nature (such as confinement protection limits), but the operating limits for the charge are charge-specific. A summary of the analyses required for an example charge is given in Table 1-1.

The range of operating variables experienced during the 30 years of reactor operation at the Savannah River Site is given in Table 1-2. The large ranges shown here demonstrate the flexibility available in a charge design. Nominal values of operating parameters for typical plutonium and tritium producing reactor charges are given in Table 1-3.

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

1.2.3.11.1 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. The reactors have 66 safety rods made of cadmium, an effective neutron absorber.

1.2.3.11.2 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 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.

1.2.3.11.3 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 prevents damage to the fuel and the reactor. The scram, or shutdown instruments, installed in each reactor are listed in Table 1-4.

1.2.3.11.4 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.

1.2.3.11.5 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.

1.2.3.11.6 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.

1.2.4 Engineered Safety Systems

1.2.4.1 Emergency Cooling System

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

Emergency Cooling of the SRS reactors is accomplished by the addition of light water to the primary reactor cooling system. This water is then recirculated by the primary heavy-water pumps.

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

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

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

(2) A header pressurized by five pumps drawing water from the 95-million-liter basin (primary).

(3) Another header pressurized by five additional pumps (primary).

(4) An electric-driven pump which supplies H_2O from the 95-million-liter 186 basin (primary).

(5) Another electric-driven pump which supplies H_2O from the 95-million-liter 186 basin (primary).

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

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 and one electric-driven pump start, 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. The ECS can also be actuated manually if required by abnormal condition control procedures.

1.2.4.2 Water Removal and Storage

If the heavy-water system leaks, 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 retention 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 airborne 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.

1.2.4.3 Airborne Activity Confinement Systems

The reactors each have 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. Following a severe accident, the reactor room spray system is activated to minimize carbon filters overheating and reduce the magnitude and duration of process room pressurization. As shown in Figure 1-7, 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 three fan motors can be powered simultaneously by either of two electric sources:

(1) The normal building power through at least two sub-stations

(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.

As shown in Figures 1-8 and 1-9, the reactors are completely surrounded by a massive concrete structure which in combination with the confinement system forms a barrier of high reliability against the possible release of radioactive material. The confinement system has the capacity to accommodate unexpected gas or energy releases.

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

1.2.4.4 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 all 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.

1.2.4.5 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 and to mitigate the consequence of a postulated core melt accident. 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.

1.2.4.6 Moderator Recovery System

The Moderator Recovery System is designed to recover the moderator from intermediate process water leaks (5 to 1,000 gpm) and return it to the reactor tank via the blanket gas space of the reactor. The recovery of moderator from intermediate leaks will minimize the impact of the leaks by averting ECS action and the resulting moderator degradation and potential contamination of the retention basin. The operation of the MRS is performed manually by the central control room operator in response to a decrease in the reactor tank moderator level.

1.2.5 Support Systems

1.2.5.1 Electrical System

To ensure continuity of service under operating, shutdown, and emergency conditions, the Savannah River Site production reactors are supplied with electrical power from multiple sources. These sources include power purchased from an offsite utility, power generated onsite, and power from diesel-driven emergency generators. The power supply system is periodically inspected, tested, maintained, and upgraded as improvements are identified and implemented.

1.2.5.1.1 Normal Supply

Each reactor area receives normal electric power from two separate sources. Two 12-MW generators are located in each of the P and K Area powerhouses. The power generated in the P and K powerhouses is supplied directly to P and K Areas. P and K Areas are also supplied by two 115-kV lines from the site distribution system. The lines are connected to transformers located within each reactor area where the voltage is reduced to 13.8 kV.

Electric power from the SRS power grid is supplied to the L Area by two independent 115-kV transmission lines. Three transformers in the L Area are connected to the 115-kV grid. One transformer can carry a reactor area load. In the event of a power failure, a supervisory control cable running along these lines enables the power dispatcher to monitor and switch equipment on the plant grid.

1.2.5.1.2 Emergency Supply

Two 1000-kW a.c. generators supply emergency power to the reactor building Eight 103-kW d.c. generators supply power to the process pump motors that maintain the heavy-water cooling flow to the reactor if the normal a.c. power fails; normally, six of the generators are operated at all times, and the remaining two are on standby. Four other diesel generators are located throughout each reactor area to provide backup power for ventilation fans, lights, and other equipment. Reactor shutdown systems, including scram circuits, safety and control rod drives, and the Supplementary Safety System, are also backed up by online batteries.

1.2.5.2 Steam

Steam is generated in P- and K-Area powerhouses for process service and ventilation heat throughout each reactor area. An interarea pipeline supplies steam from the K-Area powerhouse to L Area.

1.2.5.3 Potable Water

Potable water is supplied to each reactor area from two deepwells producing from the Tuscaloosa Formation. This is also the source for clarified service water, filtered water, and domestic and fire control water. The water is processed in a treatment plant before use.

1.2.6 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 radio-nuclides 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

The secondary cooling water discharged from the reactor heat exchangers is monitored continuously to detect any radioactivity leakage from the primary coolant.

Nonradiological samples are collected in accordance with the National Pollutant Discharge Elimination System (NPDES) permit.

Table 1-1
Summary of Required Analyses for Each Reactor Charge

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

Table 1-2
Range of Operating Variables
in SRS Reactor Charges

Variable	Range
Thermal neutron flux (full power)	1×10^{13} to 7×10^{15} n/(cm^2) (sec)
Reactor power (full power)	200 to 2915 MW (thermal)
Assembly power	Up to 21 MW (thermal)
Prompt coefficient	$+2 \times 10^{-5}$ to -15×10^{-5} k/°C[a]
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 to 450 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] 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 1-3
Nominal Values of Operating Parameters for Typical SRS Reactor Charges

Operating Parameter	Plutonium Producer (Mixed-Lattice)	Tritium Producer (Uniform-Lattice)
Principal fuel	Enriched uranium	Enriched uranium
Principal target	Depleted uranium	Lithium
D_2O flow (m^3/min)		
Per fuel	1.59	1.34
Per target	0.89	——
Total reactor	587	587
D_2O velocity (m/sec)		
Fuel	5.8	7.0
Target	7.6	——
H_2O flow (m^3/min)	672	662
Power, MW (thermal)		
Per fuel	7.4	6.0
Per target	2.5–4.8	——
Total reactor	2350	2400
Fuel surface heat flux, watts/cm^2	220	221
Assembly effluent D_2O temperature, °C		
Fuel	113	109
Target	85–110	——

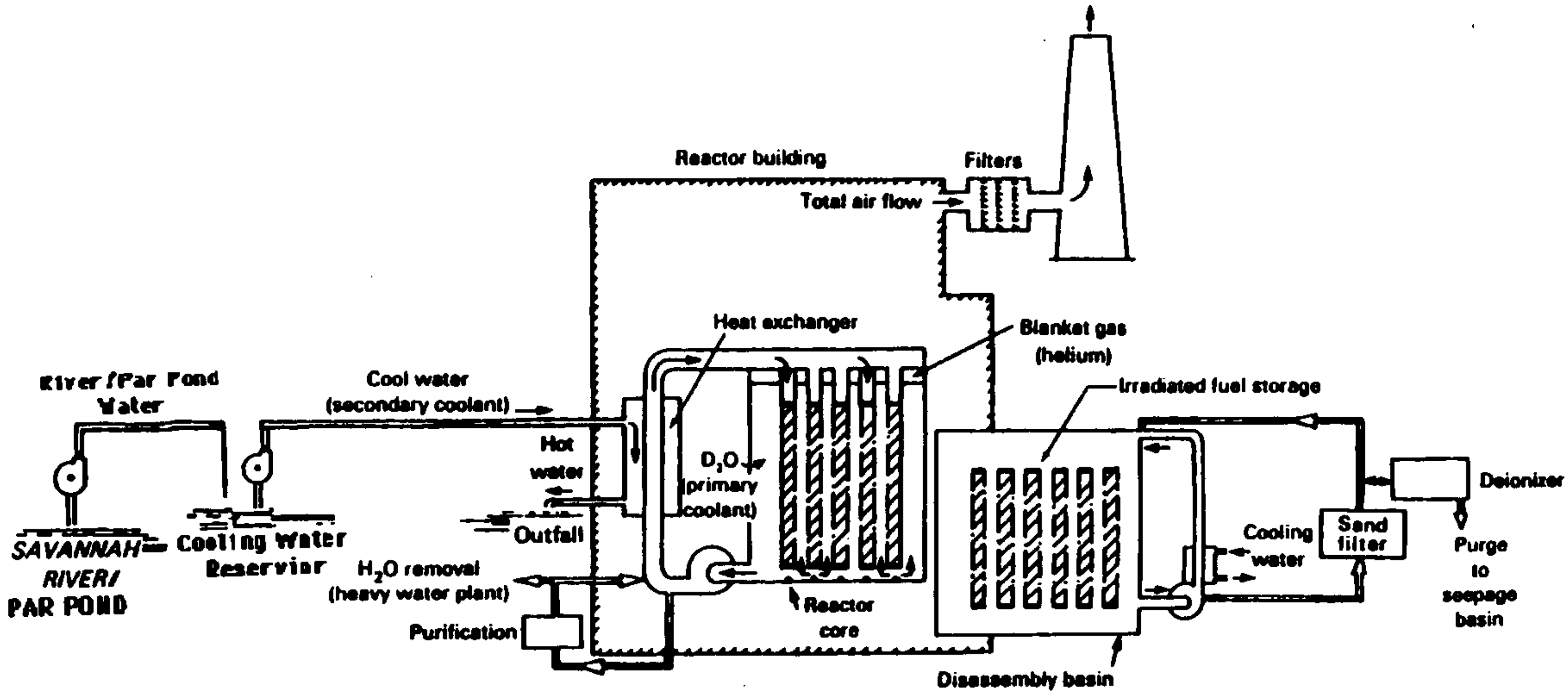
Table 1–4

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 six pump motors
Assembly coolant flow	600
Assembly average effluent temperature[b]	600
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.



TC

Figure 1–1. Reactor Process Systems

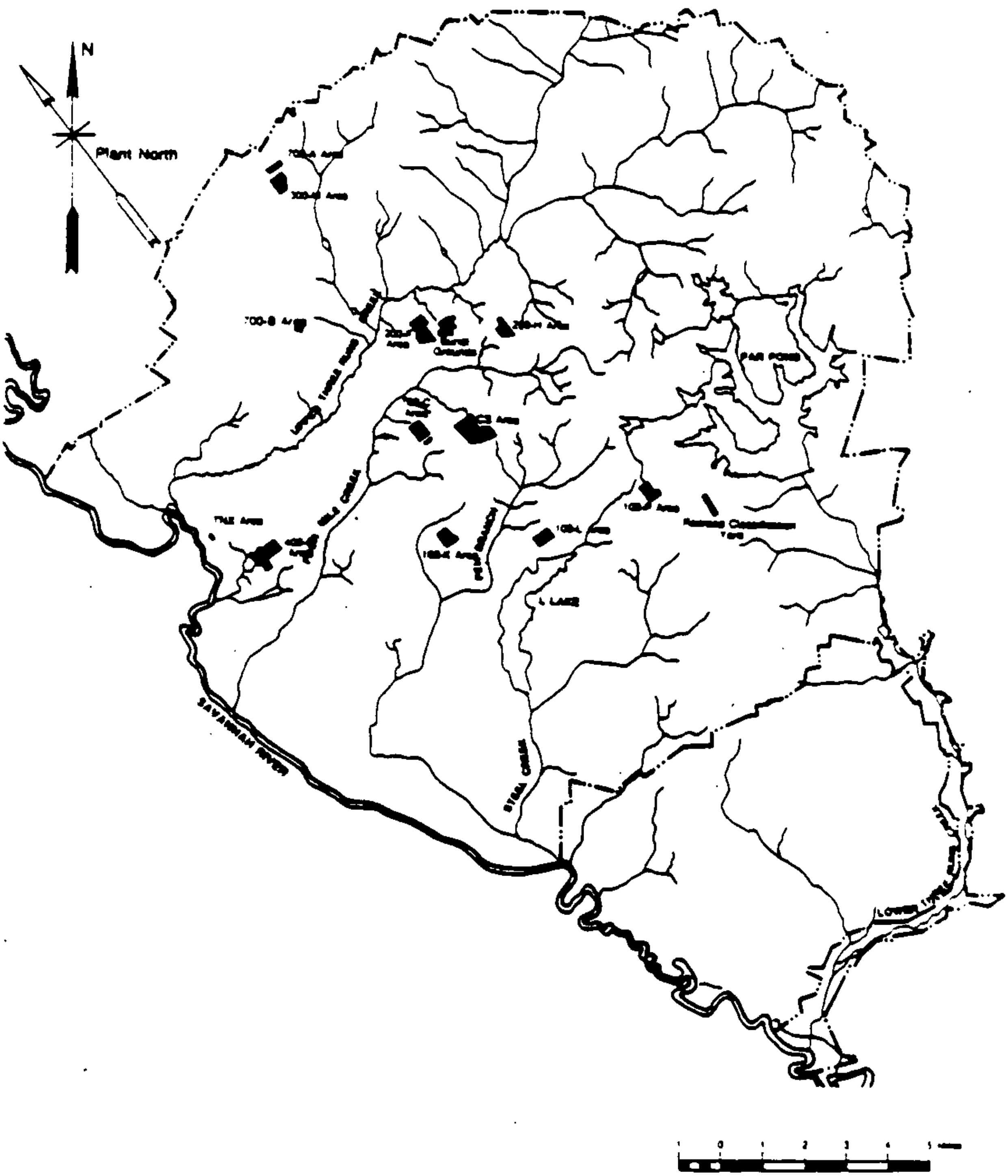


Figure 1-2. Map of SRS

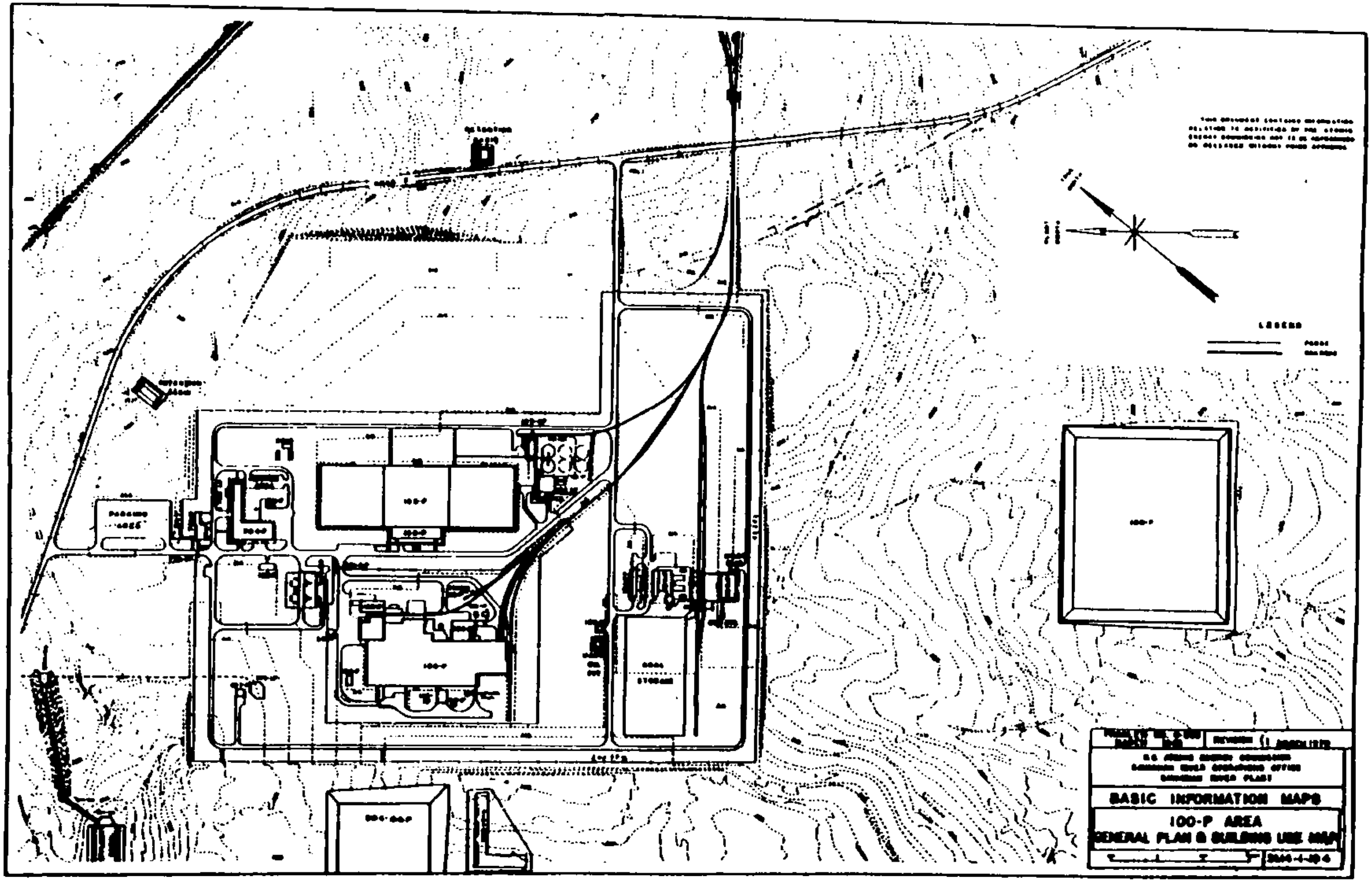


Figure 1-3. Map of P-Area

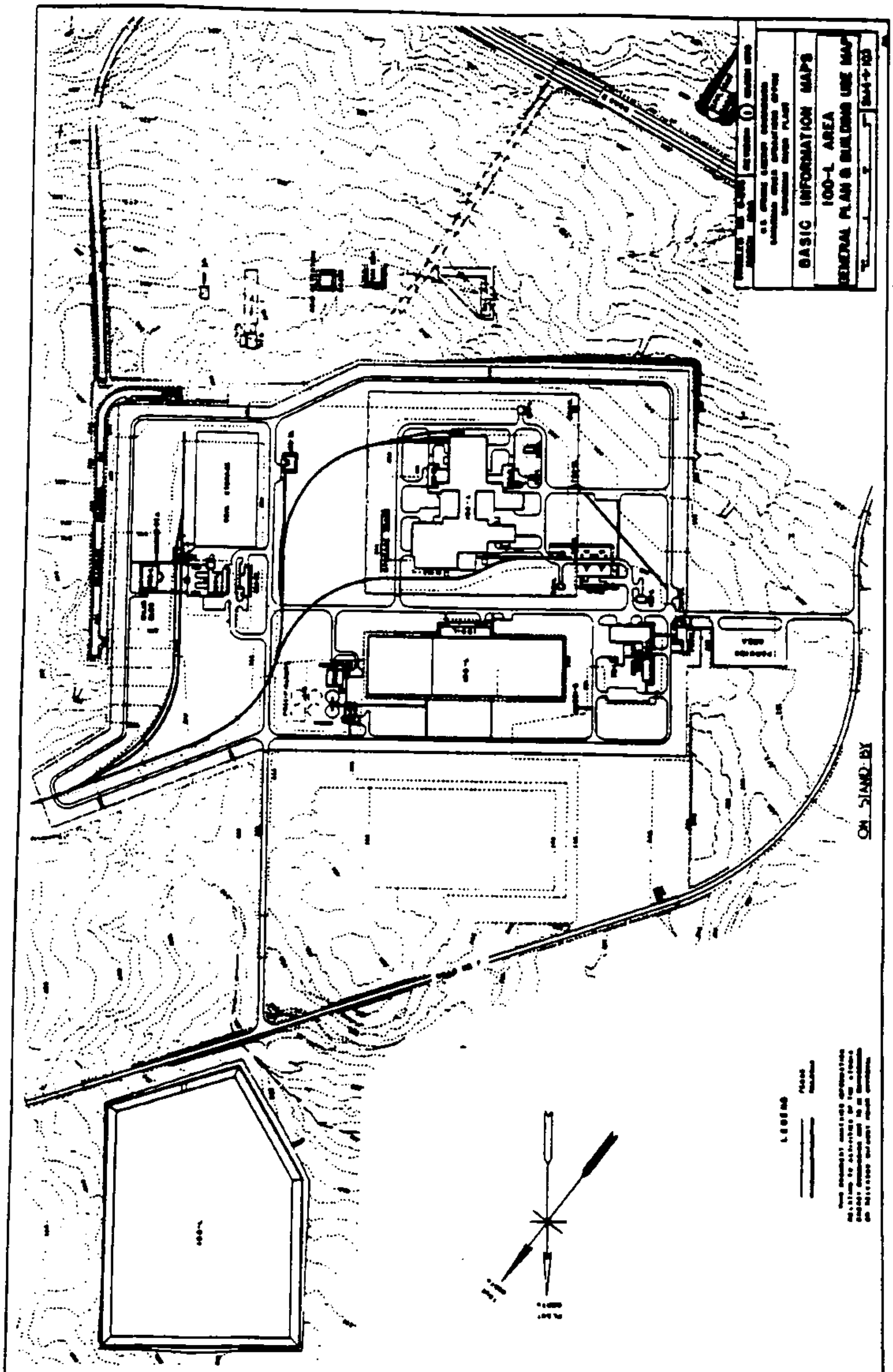


Figure 1-4. Map of L-Area

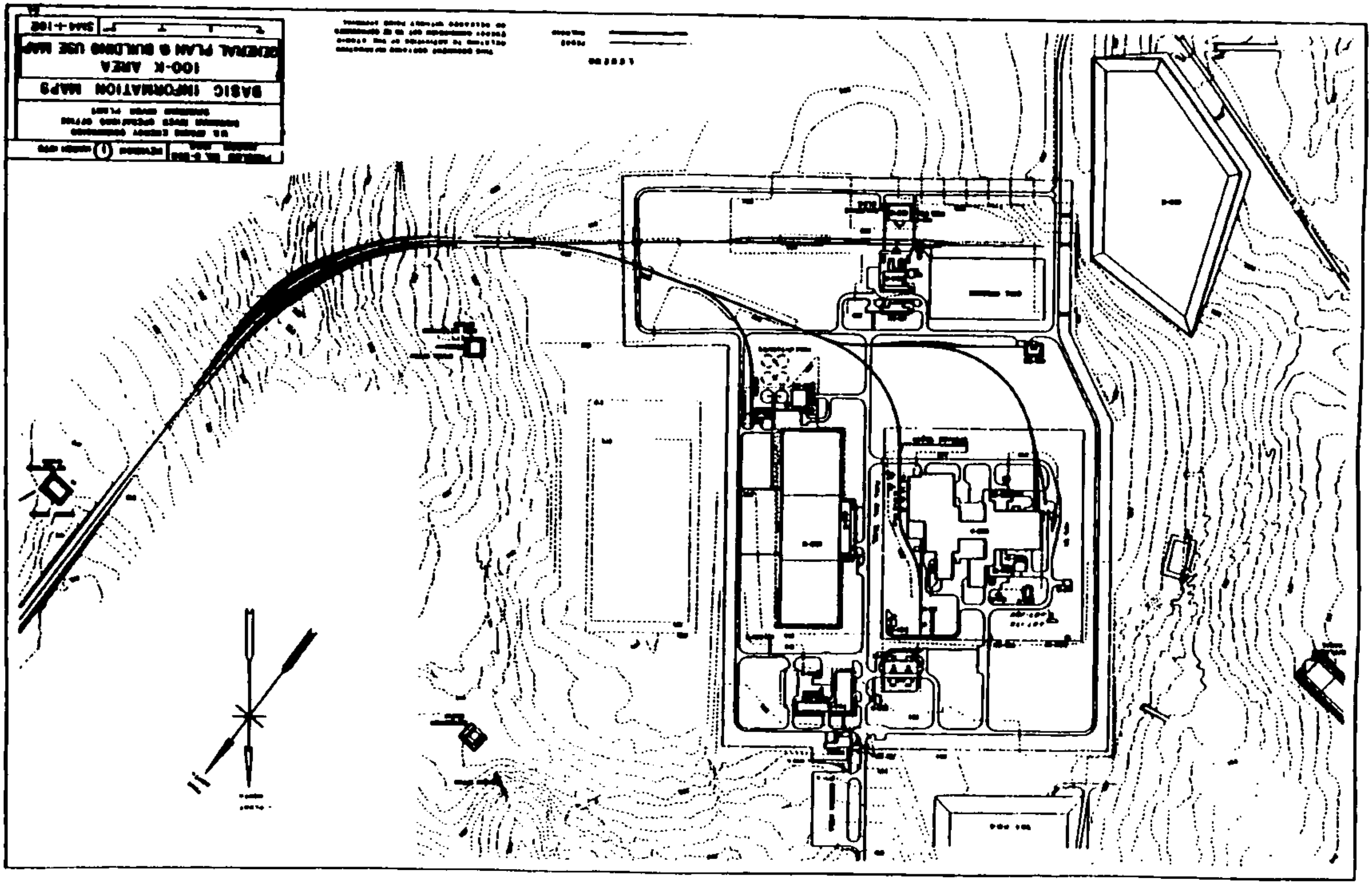


Figure 1–5. Map of K–Area

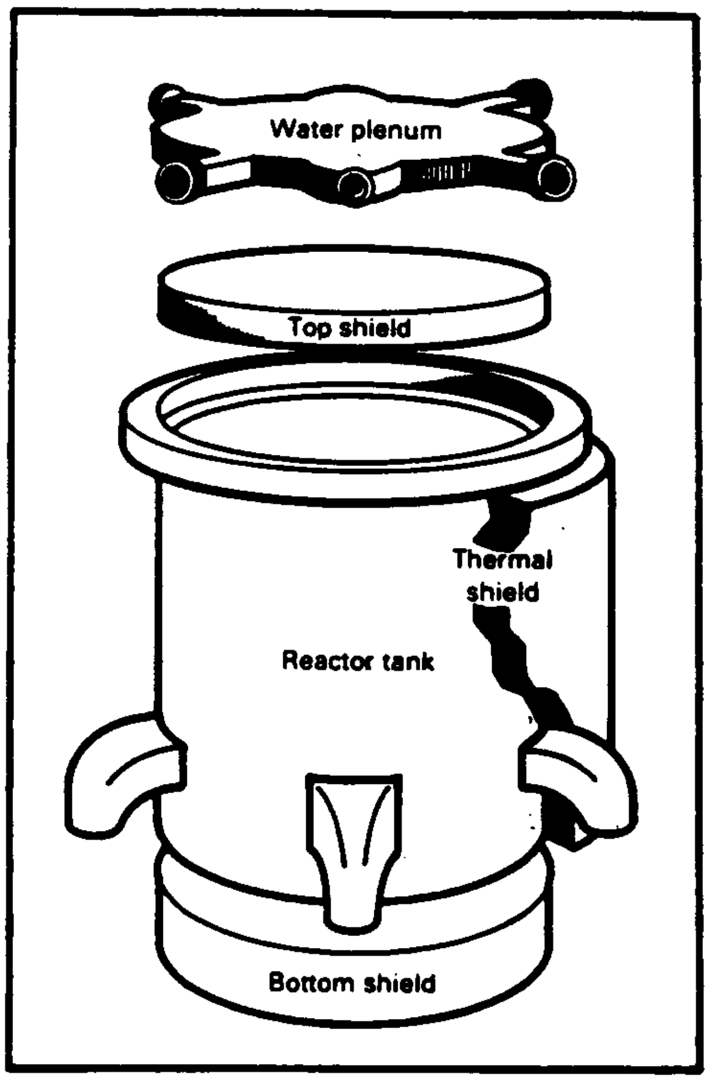


Figure 1–6. Schematic of Reactor Structure

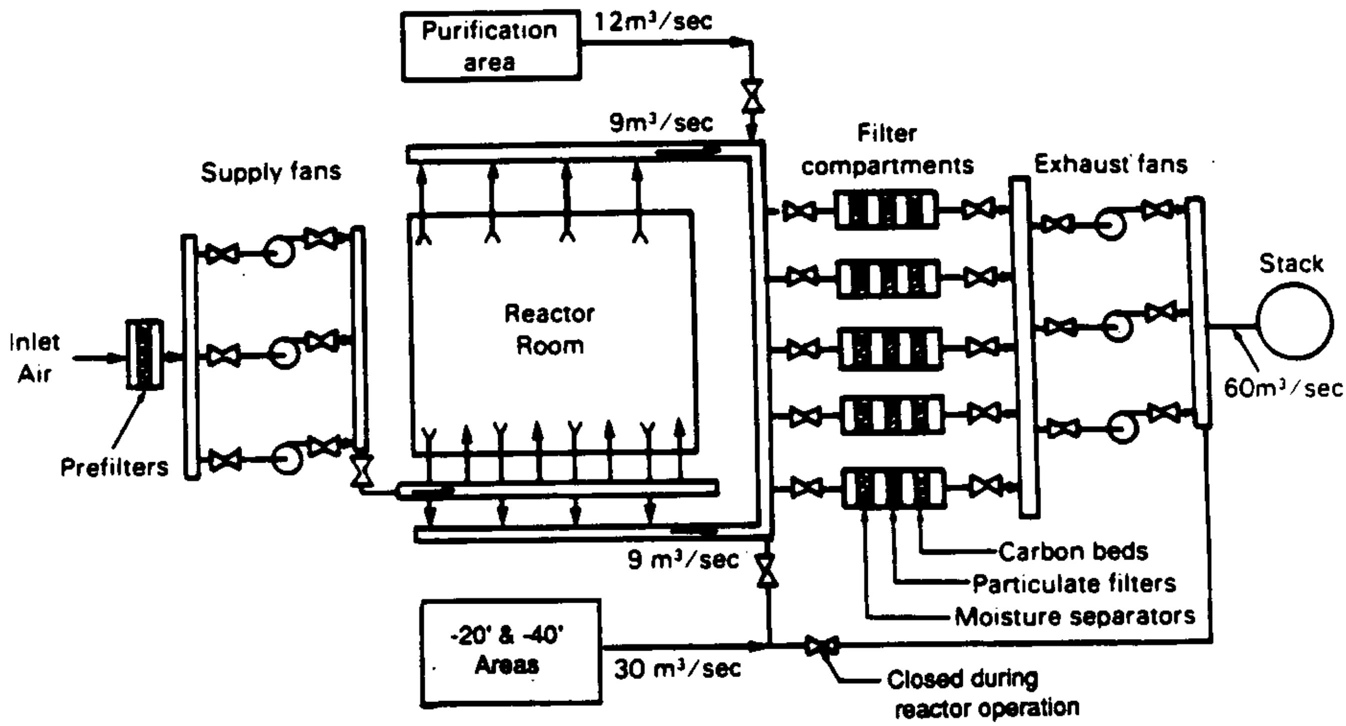


Figure 1-7. Reactor Activity Confinement System

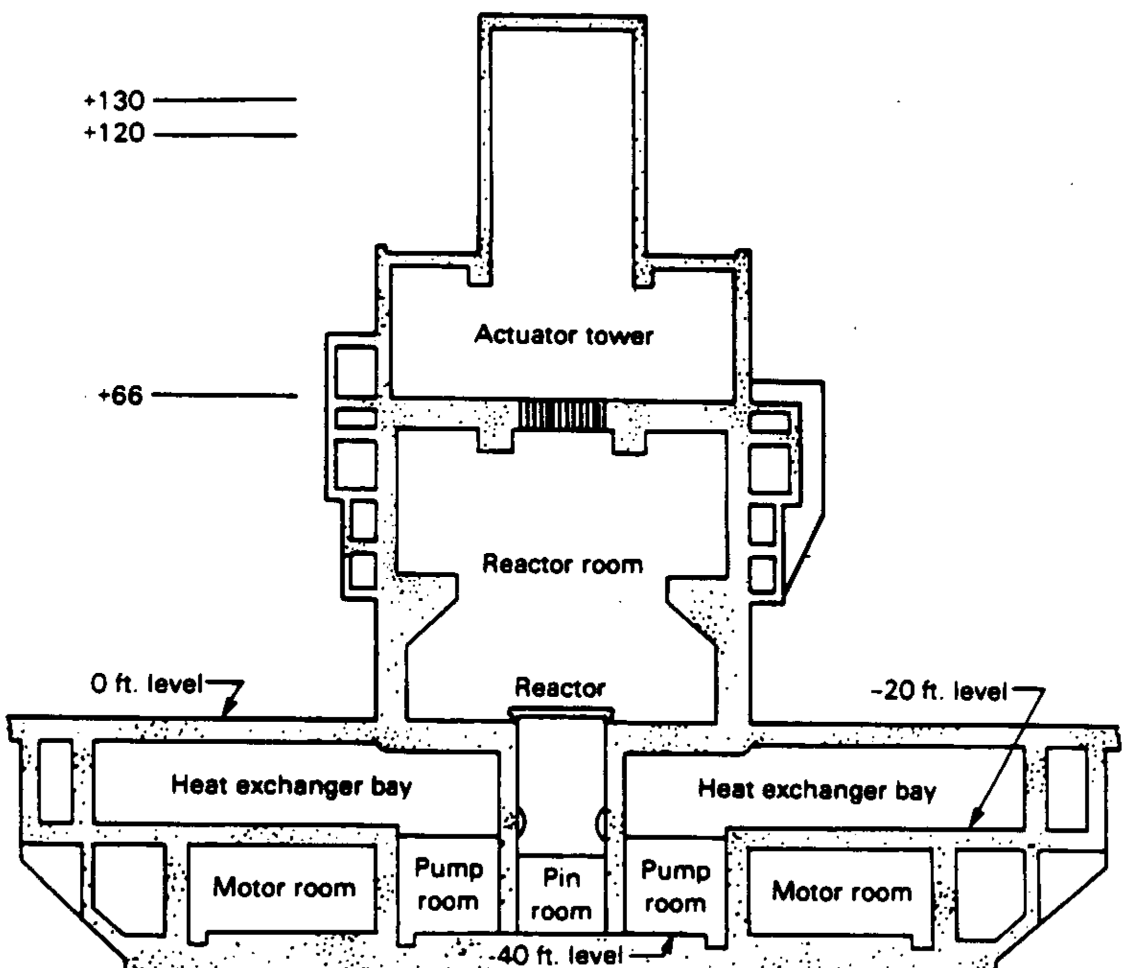


Figure 1-8. Schematic Cross Section of Reactor Process Area

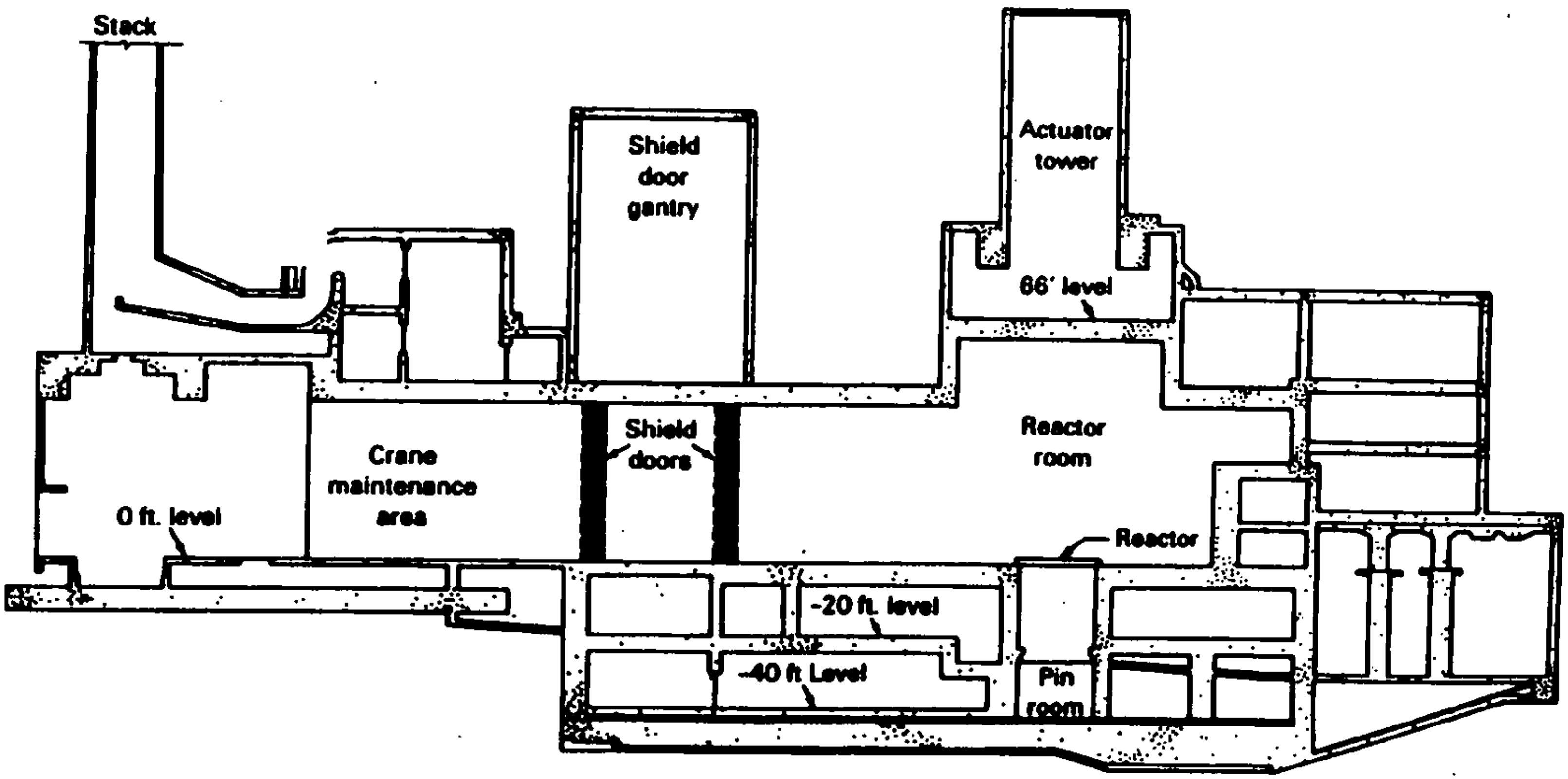


Figure 1–9. Schematic Cross Section of Reactor

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2.0 ACCIDENT ANALYSIS

2.1 Introduction

2.1.1 Background

This section of the Savannah River Site (SRS) safety information document (SID) includes safety analyses of the plant response to various postulated disturbances in process variables and to various postulated malfunctions or failures of equipment. These safety analyses have provided a basis for the selection of limiting operating conditions, limiting safety system settings, and design specifications for components/systems from the standpoint of public health and safety.

This section first describes typical transient classification and design requirements currently applied in the nuclear industry and then relates this approach to that used for the SRS reactor accident evaluations. Additional items of discussion include the flexibility of the SRS reactors and consequential impact on operating conditions, computer codes utilized, incorporation of worst-case conditions, safety system trip signals, and response characteristics. The format and content used for the accident analysis evaluations is also discussed.

The transient analyses described in the various sections of this chapter are representative of those performed for each charge design. The figures in this chapter illustrate the course of each transient, beginning at the operating limit and ending after either safety rod (SR) scram or the injection of gadolinium nitrate by the automatic backup shutdown (ABS) system. These figures are not intended to demonstrate the adequacy of the thermal hydraulic limits applied to SRS charges. The derivation of the actual limits and the demonstration of their adequacy are based on the application of the analytical techniques through the use of approved computer codes. These figures simply illustrate the use of these codes and analytical techniques. The derivation of the thermal hydraulic limits is an iterative process. The initial conditions preceding a transient are changed until the consequences of the transient fall within the acceptance criteria as discussed later in Section 2.1.3. An actual analysis must show that the acceptance criteria are met for each transient when the transient begins at the transient protection or the confinement protection limit (whichever is being determined), not the operating limit as used for the illustrative transients in various sections of this chapter.

2.1.2 Typical Transient and Accident Classifications and Design Requirements Currently Applied in the Nuclear Industry

The standard approach currently employed in the design of various licensed nuclear power plants is to use the American Nuclear Society classification of plant normal and off normal conditions. This classification uses four categories according to anticipated frequency (probability) of occurrence (Ref 2.1-1). These four categories are as follows:

- Condition I: Normal operation and operational transients
- Condition II: Incidents of moderate frequency
- Condition III: Infrequent incidents
- Condition IV: Limiting faults

Ref. 2.1-2 shows this and other categorization approaches and how they interrelate; only the above approach will be described in this section since it is the most consistent of the approaches relative to Regulatory Guide (RG) 1.70, Revision 3 (Ref. 2.1-3) requirements.

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological consequences to the public, and those extreme situations having the potential for the greatest consequence to the public should be those least likely to occur.

The transient conditions resulting from all the accidents are analyzed to such an extent that they are shown to be either:

- Inherently terminated
- Terminated by the operation of the reactor shutdown systems, which are designed to maintain the integrity of the fuel and/or process water system
- Terminated by other conditions that result in the operation of engineered safety features or other safety-related systems that are designed to maintain the integrity of the core and/or the containment/confinement and limit the potential offsite doses to the public when one or more of the protective barriers are not effective

More detailed definitions of the four accident conditions and the respective design requirements are given in the following subsections based on information from Ref. 2.1-1:

- Condition I: Normal Operation and Operational Transients - Condition I occurrences are those that are expected to occur frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter that would require either automatic or manual protective action. Since Condition I events occur frequently, they must be considered from the point of view of their effect on the consequences of fault conditions (Conditions II, III, and IV). In this regard, analysis of each fault condition is generally based on a conservative set of initial conditions corresponding to adverse conditions that can occur during Condition I operation.

- Condition II: Incidents of Moderate Frequency - These faults, at worst, result in the reactor trip with the plant being capable of returning to operation. By definition, these faults (or events) do not propagate to cause a more serious fault (i.e., Condition III or IV events). In addition, Condition II events are not expected to result in fuel failures, process water system failures, or cooling water system overpressurization.
- Condition III: Infrequent Incidents - Condition III events are faults that may occur infrequently during the life of the plant. They may result in the failure of only a small fraction of the fuel. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary, in accordance with the guidelines of 10 CFR 100. A Condition III event alone will not generate a Condition IV event or result in a consequential loss of function of the reactor coolant system or confinement/containment barriers.
- Condition IV: Limiting Faults - Condition IV events are faults that are not expected to take place but are postulated because their consequences would include potential release of significant amounts of radioactive material. They are the most drastic faults that must be designed against and they represent limiting design cases. Condition IV faults are not expected to cause a fission product release to the environment resulting in doses in excess of guideline values in 10 CFR 100. A single Condition IV event is not expected to cause a consequential loss of required functions of systems needed to cope with the fault, including those of the emergency cooling system (ECS) and the confinement/containment barriers.

Although the SRS does not categorize its transients using this method (Ref. 2.1-1), limits that meet the intent of such an approach are applied to all transients irrespective of the particular event categorization. This method is described in Section 2.1.3.

2.1.3 Spectrum of Accidents Evaluated for SRS Reactors and Acceptance Criteria/Limits for Chapter 2 Evaluations

The range of accidents reviewed in this chapter bounds the spectrum of accidents that the SRS reactors could be subjected to. These can be grouped into three general classes of transients:

- Reactivity addition accidents that increase the reactor power or the power of local regions in the reactor

- Flow reduction accidents or losses of coolant that reduce the cooling capability of the reactor or individual assemblies
- Accidents used for calculating offsite radiological doses

Table 2-1 gives an estimate of the SRS transients relative to the Reference 2.1-1 design categories.

The specific accidents considered are listed in Table 2-2. As described under the various accident descriptions, the principal reactor safety systems that are employed to limit the accident consequences are:

- Reactor Shutdown Systems and Associated Instrumentation and Control
  - Safety rods
  - Automatic backup shutdown safety computer (ABS-SC)
  - Control rods
  - Scram instruments and alarms
  - Supplementary safety system (SSS)
  - Remote monitoring and control system
- Engineered Safety Features
  - Emergency cooling system
  - Confinement heat removal (CHR) system
  - Water removal and storage system
  - Emergency spray system
  - Airborne activity confinement system (AACS)

As described in Section 1.2.3.10, limits are imposed for safe operation of the SRS reactors in the normal mode and under postulated transient conditions. The governing limits for reactor operation are called thermal-hydraulic limits. There are three operating parameters that are directly affected by these limits. The operating parameters and reasons for imposing limits on them are:

- Assembly Effluent Temperatures
  - To avoid bulk boiling of the coolant in fuel and target assemblies during steady-state operation
  - To avoid assembly and reactor damage during transients with the safety rods operable
  - To maintain the integrity of the reactor confinement system during transients with the safety rods inoperable
  - To preserve a coolable core geometry during incidents that require activation of the ECS

- Burnout Risk (BOR)
  - To prevent unacceptable levels of fission product release that could result from exceeding the critical heat flux during either transient or steady-state operation
- Reactor Effluent Temperature
  - To avoid reduced pumping capacity and cavitation damage to the reactor hydraulic system during steady-state operation
  - To avoid, during reactor transients, reductions in coolant flow greater than those assumed in the derivation of limits on assembly effluent temperature and reactor burnout risk
  - To maintain the integrity of the reactor confinement system during reactor transients with the safety rods inoperable

Five types of thermal-hydraulic limits comprise the lines of defense with regard to consequences associated with reactor transients. These are:

- Technical Limits – These limits are specified to protect the fuel and target cladding and the reactor structure for continuous operation at the limit. They are applied to the assembly heat flux and effluent temperatures plus the reactor effluent temperature to ensure that design lifetimes are achieved.
- Transient Protection Limits – These limits serve a similar purpose to the technical limits; however, they are oriented to the dynamic situation of faster acting transients. These limits are calculated assuming primary scram action and are applied to assembly effluent temperature and heat flux, plus the reactor effluent temperature with values selected to maintain reactor operation within constraints on fission product release and damage to the reactor. (Operation within the constraints is ensured by preventing the onset of flow instability in individual assemblies and cavitation in or near the reactor pumps.)
- Confinement Protection Limits – The intent of these limits is to maintain the integrity of the ASCS during specific postulated transients not terminated by safety rod action. To protect the confinement system, the criterion of preventing a major breach in the reactor tank and/or primary boundary was adopted. To achieve this, steam pressure forces are limited to values less than those required to lift the plenum, which precludes high rates of steam release to the confinement system. This limit is implemented by requiring the ABS system to activate the SSS in sufficient time to prevent attainment of a damaging pressure level. When actuated, the SSS injects a solution of gadolinium nitrate into the reactor moderator through six spargers near the center of the reactor.

Moderator circulation and diffusion distribute the nuclear poison through the reactor core.

- Emergency Cooling Limits - These limits are imposed on the assembly power effluent temperature to ensure that the ECS is capable of maintaining a coolable core geometry for loss of process water (leak) and loss of process water circulation accidents.
- Operating Limits - These limits establish the highest authorized operating power level for continuous reactor operation. They are defined as the smaller of the transient protection limit, the confinement protection limit, or the emergency cooling limit, with an allowance subtracted for normal process fluctuations.

Typical values for each limit are given in Table 2-3. A summary of these SRS thermal-hydraulic limits and how they would relate to various category accident conditions described in Section 2.1.2 is given in Table 2-4. A review of these limits indicates that at least two basic criteria are met by employing them in the SRS reactors. First, if the limits are met, a coolable inplace core geometry is ensured following any of the transients, and second, radiological releases in excess of 10 CFR 100 dose guidelines will not occur. (In fact, radiological releases, if any, will be far less than the 10 CFR 100 guidelines, as shown in Tables 2-8 and 2-12.)

2.1.4 Flexibility of SRS Reactors and Consequential Impact on Operating Conditions

Typically, there are a total of 10 to 35 reloadings of current charges each year in the three reactors to produce plutonium and tritium. This results in a wide range of assembly and reactor operating variables. The ranges of the more important operating parameters experienced to date are presented in Table 1-2 and are partially summarized in Table 2-5.

As described in Section 2.1.3, one of the limiting parameters relative to meeting the Table 2-3 limits is that sufficient critical heat flux (or burnout) margin exists. An example of how coolant temperatures and flow rates can affect the critical heat flux is given in Figure 2-1, where the critical heat flux (q''_{crit}) capability is shown as a function of local quality (x) for several typical mass fluxes (G). The local quality is defined as:

$$x = \frac{h - h_{sat}}{h_{fg}} \qquad \text{(Eq. 2.0-1)}$$

where:

h = local coolant enthalpy
h_{sat} = saturated coolant enthalpy at local pressure
h_{fg} = coolant heat of vaporization at local pressure

The above equation implicitly depicts the increase in critical heat flux with decreased coolant temperatures. This information is based on SRS experimentally verified critical heat flux correlations developed from experimental data. The Figure 2-1 method of data presentation has been found by Reference 2-5 to be the most applicable and least confusing approach to representing this type of subcooled critical heat flux information and the consequential effects of independent parameters. The method depicts the large critical heat flux margins for subcooled nucleate boiling that can be obtained as part of the SRS reactors flexibility in accommodating continually changing conditions for various core loads. Sensitivity to the independent parameters can also be found by the base derivatives of the SRS critical heat flux correlations, as given in Table 2-6 where the partial derivative with respect to coolant subcooling (T_s), coolant pressure, and coolant mass flux have been calculated. An estimate of the magnitude of these derivatives indicates that a 1 percent increase in critical heat flux capability occurs for a 0.5 to 1°C increase in subcooling, for a 1 psi increase in pressure, or for a 2 to 3 percent increase in mass flux. These are typical sensitivity values for the critical sensitivity values for the critical heat flux, as also found in other studies (Ref 2.1-5). The basic conclusion from Figure 2-1 and Table 2-6 information is that there are various methods of effectively increasing the critical heat flux if required for a particular type of core loading. This is the approach incorporated by the SRS in defining operating conditions that meet the required burnout risk (BOR), as described in Section 2.1.3.

2.1.5 Computer Codes Utilized

Summaries of the principal computer codes used in the transient analyses are given in this section. Other computer codes, particularly very specialized codes in which the modeling has been developed to simulate one given accident, are summarized in their respective accident analysis sections.

2.1.5.1 AA3 Computer Code

The AA3 computer code models neutronic and engineering aspects of the SRS reactors, including the reactor core, the primary coolant loop, the shutdown systems, and the protective instrument system. Steam generation rates and steam pressure under the top shield and plenum are determined. The neutronics are calculated using point kinetics. Normal reactor flow is symmetrical in the six external loops, so the six loops are combined as one equivalent loop. The code cannot model asymmetry; therefore, the input data must be modified to evaluate asymmetric situations.

2.1.5.2 GRASS Computer Code

The GRASS computer code is composed of 27 modules which execute a space-time coupled neutronics engineering calculation. It calculates the reactor core and process water response for all transients except loss-of-coolant accidents (LOCAs) without safety rod scram or backup shutdown. The neutronics portion of the code executes a three-dimensional, few-group, time-dependent diffusion calculation. The hydraulics portion of the calculation is done using an iterative matrix solution technique. The code is limited to nonboiling transients since it is not considered accurate when two-phase flow exists. Termination of accidents by safety rod scram or ink-injection is modeled through tabular input.

2.1.5.3 FLOOD84 Computer Code

The FLOOD84 computer code computes the equilibrium flow distribution for reactor assemblies under design basis loss-of-pumping and loss-of-coolant accidents. The computation of these flow distributions uses key parameters that are normalized to results obtained from in-reactor SRS experiments.

2.1.5.4 APOLLO Computer Code

The APOLLO computer code computes technical, transient-protection, and operating limits on assembly and reactor effluent temperatures for the SRS reactors. Nine enveloping incidents consisting of hydraulic transients, reactivity transients, and quasi-steady-states are analyzed within the framework of the transient protection limits. The reactivity transients are not modeled explicitly, but are addressed through limits based on a 5°C assembly temperature rise to scram. A point reactor kinetics formulation (using a Runge-Kutta solution method) is employed in calculating changes in assembly heat generation during hydraulic transients terminated by safety rod insertion. Flow transients are not calculated explicitly, but are based on results of GRASS (described earlier in this section) calculations and in-reactor measurements. Each assembly is assumed to undergo the same percent reduction in flow during a hydraulic transient.

2.1.5.5 TRAC Computer Code

The TRAC computer code is a best-estimate computer program for analysis of pressurized water reactor transients being used to model the overall hydraulic response of the reactor system for accidents including the large LOCA. The code supplies time-dependent pressure boundary conditions for the FLOWTRAN assembly calculations. All six external loops are modeled with 1-D components. The loops connect to 3-D vessel components at the plenum and tank bottom.

2.1.5.6 FLOWTRAN Computer Code

The FLOWTRAN computer code is a new computer code which is still under development. It has been designed primarily to provide a method for computing power limits for the SRS reactor assemblies by calculating the assembly thermal-hydraulic behavior. At present it can be applied only to transients during which the assemblies remain full of water. Such transients encompass the first few seconds of a large break LOCA, a pump shaft break, or any of a series of lesser thermal-hydraulic or reactivity-induced reactor transients. Because of the no bulk boiling criteria currently imposed on the SRS, the current computational capabilities of the code are adequate for use in evaluating power limits for SRS assemblies.

2.1.6 Conditions Evaluated

For each safety evaluation of a particular change design, the analysis accounts for variations and uncertainties in the parameters to the extent that they are pertinent to providing a conservative assessment of consequences. The typical parameters considered for accident analyses include:

- Initial conditions at the time of the transient, such as:
  - Power, coolant temperatures, pressure, core coolant flow/distribution, margin to critical heat flux, and other plant conditions taking into account control system and instrument accuracy
  - The full range of expected normal operating conditions
  - Variations in plant parameters with power and core exposure
- Spatial power distribution variations during transients which include the effects of:
  - Localized reactivity perturbation
  - Nuclear reactivity feedbacks
  - Distribution of neutron absorber introduced as the accident mitigator
- Any force or pressure transients that might affect barrier integrity
- The performance characteristics of the protection systems and engineered safety features utilized to mitigate the consequences of the incident, including electrical circuit response times, instrument errors, and dynamic characteristics of the required power and fluid systems that activate the engineered safety system
- Environmental conditions (e.g., weather, seismic, etc.)

2.1.7 Safety System Scram Signals and Response Characteristics

As described in detail in Chapter 1, the SRS reactors are equipped with an extensive array of protective instruments and shutdown systems for response to abnormal conditions. The levels of response depend on the amount of deviation from normal operating conditions. Prior to alarms or automatic corrective action, operator action through the response to observed instrumentation variations will normally correct or adequately monitor the condition. If this does not happen, then alarms (which require procedural action) provide the first indication of abnormal operation. For minor upsets, control rods are driven (rod reversal system) into the core to reduce power. For larger deviations from normal operating conditions, a scram occurs which causes the safety rods to drop (due to gravity) into the core and the control rods to be fully driven into the core. In the unlikely event that the safety rods fail to drop or that power is not effectively reduced quickly, backup protection is provided by the ABS, which injects a liquid neutron poison into the moderator space of the reactor tank (see Section 1.2.3.11).

The reactors are equipped with a set of 66 cadmium safety rods to provide rapid shutdown in the event that any of the circuits signal a scram. These safety rods are independent of the control rod system. During normal operation, the safety rods are held above the core by magnetic clutches that disengage on loss of electrical power. Upon receipt of a scram signal, the safety rods drop about halfway into the core by gravity in approximately one second (their travel the rest of the way into the core is slowed by snubbers to avoid mechanical damage). Control rods are also driven into the reactor following a scram signal but on a much slower time scale (approximately two minutes).

The scram circuits are listed in Table 2-7. The setpoints for these circuits can be adjusted to meet the changing requirements of new reactor charge designs and operating conditions. Some of the circuits initiate a scram for a fixed value of the monitored variable, while others are routinely adjusted to maintain a specified margin from the operating value of the monitored variable. Scram setpoints are routinely checked by following instructions in written procedures to verify that the setpoints are at the specified values. Bypass of scram instrumentation for testing or maintenance is rigidly controlled by operating and maintenance procedures. The status of circuits with bypass requirements for certain power level changes is monitored by the control computer when closed-loop control is used.

A brief description of several key scram circuits and associated monitors is provided below:

- <u>High-Level Flux Monitor</u> - High-level flux monitors detect the initial responses that terminate undesired increases in reactor power. Four gamma–compensated ion chambers monitor the neutron flux at any of four circumferential locations, including three axial elevations in the thermal shield. A reading of 106 percent full power, on any one of the monitors, initiates a scram.

- Assembly Average Effluent Temperature – The safety computers monitor the assembly effluent temperature from each assembly in the reactor once every 0.36 seconds and initiate a scram whenever successive passes give an assembly temperature reading above the scram setpoint (a 5°C temperature increase above the operating limit at full power) and the temperature rise is confirmed by temperature rises in surrounding assemblies.

- Period Monitor – The period monitor detects and causes a reversal (through control rod insertion) for rapid increases in neutron flux (20 second period or less). Two gamma–compensated ion chambers monitor the rate of change of neutron flux at the reactor tank wall. A scram is initiated whenever either monitor detects an increase with a period of 10 seconds or less.

- Assembly Coolant Flow – The safety computers provide protection from loss of coolant flow for reactor assemblies by monitoring pressure drops across each assembly bottom fitting every 0.15 seconds and initiating a scram whenever two successive readings are below the very low flow scram point or above the high scram point specified for the assembly.

- D_2O Plenum Pressure – The plenum pressure monitor terminates reactor operation if the fuel coolant supply pressure is significantly decreased. Two pressure switches in independent impulse lines measure pressure at the center of the plenum. A scram is initiated when the pressure in either line decreases to 90 percent of normal pressure, which corresponds to a flow reduction of about 5 percent.

2.1.8 References

2.1-1 American National Standards Institute-51.1/N18.2. Nuclear Safety Criteria for the Design of Stationary PWR Plants, 1973.

2.1-2 American National Standard Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, ANSI/ANS-51.1-1983, 1983.

2.1-3 Regulatory Guide 1.70, Rev. 3. Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition), U. S. Nuclear Regulatory Commission, November 1978.

2.1-4 Coffield, R., L. S. Tong, and W. M. Rohre. A Subcooled DNB Investigation of FREON-113 and Its Similarity to Subcooled Water DNB Data, Nuclear Engineering and Design, Vol. 11, No. 1, pp. 143-153, 1970.

2.1-5 NUREG-0800. Standard Review Plan, U. S. Nuclear Regulatory Commission, July 1981.

Table 2-1 (Page 1 Of 2)
Estimate Of SRS Transients Relative to Reference 2.1-1 Categories

Condition I: Normal operation and operational events that are expected to occur during reactor operation or shutdown.

- I-1 Normal steady-state operation
- I-2 Hydraulic startup
- I-3 Controlled nuclear startup
- I-4 Controlled nuclear shutdown
- I-5 Hydraulic shutdown
- I-6 Charge and discharge operations
- I-7 Changes in reactor power (power increases of 4 percent or less, the rod reversal setpoint). Examples of these include, but are not limited to:
    - Rod reversals
    - Control rod moves (intended and unintended)
    - Adjustments in tilt
    - Control rod faults
- I-8 Temperature fluctuations (changes that do not exceed operating temperature limits). These include, but are not limited to:
    - River water temperature change
    - Gradual change in cooling water temperature
    - Par Pond temperature changes
    - Bulk moderator temperature changes
- I-9 Operation with permissible deviations (deviations in normal operation that are permitted under the SRS technical specifications). These include, but are not limited to:
    - Online equipment testing
    - Bypassed safety circuits
    - Inoperable equipment

Condition II: Incidents of moderate frequency. These are expected to occur within a calendar year or at least once every 20 years and cause a scram.

- II-1 Loss of blanket gas pressure
- II-2 Inadvertent shutdown due to accidental activation of supplementary safety system

ESTIMATE OF SRS TRANSIENTS RELATIVE TO REFERENCE 2.1-1 CATEGORIES

II-3 Uncontrolled single rod withdrawal

II-4 Loss of all ac power to plant grid

II-5 Loss of ac motor

II-6 Loss of dc motor

II-7 Loss of cooling water pump

II-8 Inadvertent insertion of individual safety or control rod (or left in the reactor during startup)

II-9 Heat exchanger tube leaks severe enough to require reactor shutdown

II-10 Small amounts of moderator leakage

II-11 Inadvertent radioactive waste release

Condition III: Infrequent events. Events that are assumed to occur once in the life of the plant. Reactivity consequences within 10 CFR 100.

III-1 Uncontrolled gang rod withdrawal

III-2 Small process water loss-of-coolant accident (LOCA)

III-3 Small cooling water line break

III-4 Process water pump shaft break

III-5 Process water pump shaft seizure

III-6 Inadvertent operation of ECS (during reactor shutdown)

Condition IV: Limiting faults. Events that are not expected to occur, but serve as part of the plant design basis.

IV-1 Major process water LOCA

IV-2 Major cooling water line break with and without a loss-of-pumping accident

IV-3 Loss of cooling during charge and discharge operations

IV-4 Misload accident

IV-5 Large single assembly flow blockage

Table 2–2

Accidents Evaluated for SRS Reactors

Reactivity Insertion Accidents

- Decrease/increase in process water inlet temperature (Section 2.2)
- Single control rod withdrawal (Section 2.4.1)
- Partial control rod insertion or withdrawal (Section 2.4.1)
- Gang rod withdrawal (Section 2.4.2)
- Control rod melting (Section 2.3.6
- Loss of target (Section 2.3.6)
- Loss of fuel (Section 2.3.6)
- Reloading error (Section 2.4.3)

Flow Reduction Accidents

- Loss of process water pump power (Section 2.3.1)
- Loss of cooling water pump power (Section 2.2.1)
- Loss of process water and cooling water pump power (Section 2.3.2)
- Pump shaft break (Section 2.3.4)
- Rotovalve closure (Section 2.3.3)
- Flow reduction in single assembly (flow blockages) (Section 2.3.6)
- Loss of control rod cooling (Section 2.3.6)
- Loss of blanket gas pressure (Section 2.3.5)
- Loss of process water (Section 2.6)
- Loss of cooling water (Section 2.2.4)
- Loss of process water circulation (Section 2.3)
- Loss of cooling during or after discharge (Section 2.7)

Accidents Used for Calculating Offsite Radiological Doses

- Release or spill of moderator from primary coolant loop (Sections 2.2.3, 2.2.4, and 2.6)
- Fuel assembly drop during discharge operations (Section 2.7)
- Loss–of–coolant accident (Section 2.6)
- Radioactive waste release (Section 2.7)
- Reloading error during shutdown (Section 4.3)

Table 2-3

Typical SRS Reactor Thermal-hydraulic Limits[a]
(Based on Full Power Operation)

Limiting Parameter	Technical Limit	Transient Protection Limit	Confinement Protection Limit (b)	Emergency Cooling Limit (b)	Operating Limit
Assembly effluent temp					
Assembly average temp, °C	115	112	114	115[c]	110
Channel or quadrant temp, °C	115	112	—	—	109
BOR, MW[d]	0.03	—	—	—	0.02
Reactor effluent temp, °C	104	100	—	—	95

(a) These limits vary with river water or Par Pond temperatures, which range from 5 to 30°C. The limits also vary with the type of reactor charge.

(b) Confinement protection limits and emergency cooling limits are sometimes specified by assembly power. The temperature shown is consistent with typical assembly power limits.

(c) Under current operation, the ECS limits are set at values below the typical ones cited above. An ongoing safety analysis is being conducted on the ECS. Until the analysis is completed, the ECS limits will continue to be restricted.

(d) Burnout risk, based on fission product release (per reactor charge) from melting due to film boiling burnout, includes correlation for effects of ribs on cooled surfaces.

Table 2-4

Summary of SRS Reactors' Thermal Hydraulic Limits, Associated Acceptance Criteria, and Relationship to ANSI N18.2-1973 Conditions

SRS Limits	Acceptance Criteria for Operation Under Limits	Related ANSI N18.2-1973 Condition (Ref 2.1-1)
Technical limits	Protect against damage to fuel and plant components for continued steady-state operation	Condition I: normal operation and operational transients
Transient protection limits	Maintain transient design constraints on fission product release and reactor damage	Condition II: incidents of moderate frequency
		Condition III: infrequent incidents
		Condition IV: limiting faults
Confinement protection limits	Ensure effectiveness of confinement by preventing a major breach of reactor vessel and/or primary boundary	Condition IV: limiting faults
Emergency cooling limits	Ensure ECS maintains core coolable geometry for enveloping loss of coolant transients	Condition IV: limiting faults

Table 2-5

Range of Experienced Operating Variables

Variable	Range
Thermal neutron flux (full power)	5×10^{13} to 7×10^{15} n/cm^2 sec
Reactor power (full power)	200 to 2,915 MW
Assembly power	Up to 21 MW
Prompt coefficient	$+2 \times 10^{-5}$ to -15×10^{-5} $\Delta k/°C$[(a)]
Moderator coefficient	-1×10^{-5} to -35×10^{-5} $\Delta k/°C$
Reactivity in control rods	Up to 30% Δk at cycle beginning to 1% Δk at cycle end
Reactivity in xenon after shutdown	Up to 60% Δk
Irradiation cycle length	4 to 400 days
Fuel heat flux	Up to 2,900,000 Btu/hr-ft^2
Total D_2O flow	90,000 to 163,000 gpm
D_2O flow per assembly	Up to 1,050 gpm
Assembly coolant velocity	Up to 72 ft/sec
U-235 loading	Up to 1,650 kg per charge; up to 4.2 kg per assembly

(a) Overall temperature coefficient (prompt plus moderator) is always negative.

Table 2–6

Base Derivatives for Effect of Independent Parameters on Critical Heat Flux

$$\frac{\delta q''_{CRIT}}{\delta T_S} = +1.5 \times 10^4 \frac{\text{Btu/hr-ft}^2}{°\text{C}}$$

$$\frac{\delta q''_{CRIT}}{\delta P} = +1.1 \times 10^4 \frac{\text{Btu/hr-ft}^2}{\text{psia}}$$

$$\frac{\delta q''_{CRIT}}{\delta G} = +0.36 \frac{\text{Btu/hr-ft}^2}{\text{lb/hr-ft}^2}$$

Table 2–7 (Page 1 or 2)

Savannah River Reactor Automatic Scram Circuits

Variable Measured	Number Provided[a]	Typical Setpoint	Action[b]
Neutron flux (high level flux monitor)	4	104% 106%	Rod reversal scram
Assembly average effluent temperature	597 in P, K, and L reactors (each)	Increase 5°C	Scram
Rate of change of neutron flux (period)	2	20–second period 10–second period	Rod reversal scram
Assembly coolant flow	597 in P, K, and L reactors (each)	One very low flow One high ΔP	Scram Scram
Plenum pressure	2	Decrease 10%	Scram
Cooling water supply header flow	1 for each of 2 headers	Decrease 10%	Scram
Individual heat exchanger flow	1 for each of 12 heat exchangers	Decrease 10%	Scram
Control rod coolant supply pressure	1	Drop to 81%	Scram
Pump power supply	1 for each of 6 pump motors	Power lost to one motor	Scram
Blanket gas pressure	2	4 psig	Scram
Moderator level	1	Drop 12 inches	Scram
Control system power supply	1	Loss	Scram
Seismic activity	2 of 3 coincidence	2.6 mm pendulum gap	Scram

Table 2-7 (Page 2 of 2)

Variable Measured	Number Provided[a]	Typical Setpoint	Action[b]
Operability of safety computers	1	1 bypassed or inoperable	Warning
		2 bypassed or inoperable, or 1 bypassed and 1 inoperable	Scram
Operability of neutron flux monitors	1	Bypass 2 monitors Bypass 3 monitors	Warning Scram[c]

(a) A manual scram circuit is also provided.

(b) Rod reversals are initiated by the control computers when the assembly effluent temperature (either assembly average or channel effluent) increases 1°C above the operating limit. The rod reversals shown in this table are the only ones initiated by scram circuits.

(c) A program is planned to convert this to a rod reversal circuit.

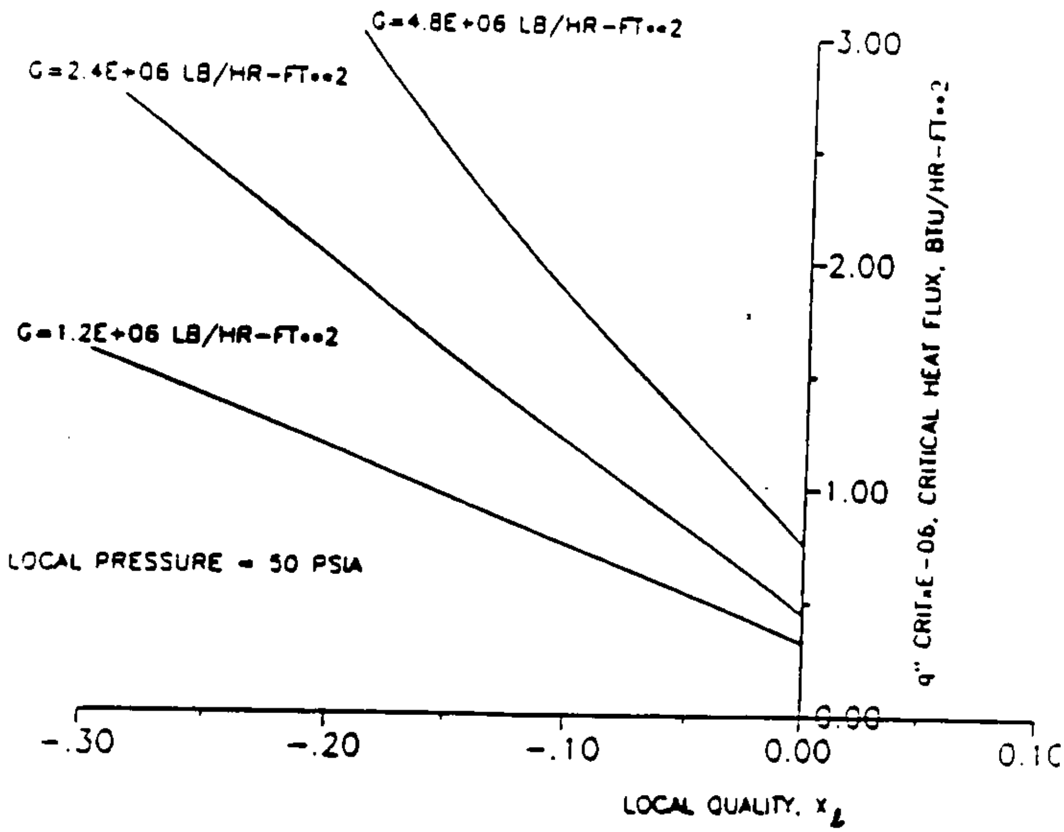

Figure 2–1

Effect of Coolant Mass Flux and Local Coolant Subcooling on Critical Heat Flux Capability (at 50 psia Local Pressure)

2.2. Increase/Decrease in Heat Removal by the Cooling Water System

Various events can be postulated that result in an increase or decrease in heat removal from the process water by the cooling water. For the SRS reactors, the cooling water is the equivalent of the secondary system on typical power producing plants. Several such transients have been identified to represent the range of possible events. These are:

- Increase/decrease of cooling water temperature
- Decrease/increase of cooling water flow
- Loss of cooling water pumping power
- Loss of cooling water reservoir supply of river water/PAR Pond water
- Loss of cooling water inventory (without resultant loss of process water pumping power)
- Loss of cooling water inventory (with loss of process water pumping power)

All of these events can result in an increase in process water reactor inlet temperature. Thus, they have been evaluated relative to the Section 2.1.3. thermal-hydraulic limits, in the following discussions.

The increase/decrease of cooling water temperature and flow transients are bounded by normal environmental temperature variations and standard variations in the equipment operations. The consequences of such variations are very slow acting transients where several degree temperature changes may develop over a several hour period. These results are enveloped by other transients described in this section and thus their transient performance has not been included.

2.2.1 Loss of Cooling Water Pumping Power

2.2.1.1 Identification of Causes and Accident Description

As described in Section 1.2.3.3, water from the Savannah River or PAR Pond is supplied to the reactor areas primarily to cool D_2O in the main heat exchanges. The water is pumped into the area reservoir, which has a 25-million-gallon capacity. It is then pumped out of the reservoir to the reactor building through two large inlet headers. Effluent coolant water leaves the reactor building in two effluent headers and enters the effluent sump. The water then overflows a weir in the effluent sump and flows back to the river via the effluent streams and swamps (K Reactor) or to a cooling pond (L and P Reactors). Water [illegible] P Area normally flows by gravity back to PAR Pond. Water [illegible] L Area returns to the Savannah River via L Lake, which [illegible] to cool the reactor effluent before discharge into the river. [illegible] pumphouse adjacent to the H_2O reservoirs in each area [illegible] two large-capacity and eight small-capacity pumps that distribute water to the reactor building. They deliver the water to the reactor building through two inlet headers. The eight smaller pumps have flywheels which extend the flow coastdown after a power failure. The two large pumps do not have fly wheels.

If electrical power were lost to all the pumps supplying cooling water directly to the reactor heat exchangers, the flow would decrease as shown in Figure 2-2. The flow levels out at a value of approximately 25 percent of full flow, which would be sustained by gravity flow from the 25-million-gallon reservoir to the effluent sump. The following analysis bounds the effects and consequences of the loss of a smaller number of pumps by assuming that all H_2O pumping is lost.

2.2.1.2 Analysis of Effects and Consequences

As a result of the decrease in cooling water flow, the temperature of process water would increase at the reactor inlet. This would result in a gradual power decrease until scram occurs. In the more severe cases, the flow would decrease rapidly and scram would occur sufficiently fast so that only minor temperature increases would occur in the reactor.

The first scram instruments to respond would be all of the individual heat exchanger flow monitors, followed quickly by the two cooling water header flow monitors and later by the assembly coolant temperature monitors. Figure 2-3 shows the worst-case core coolant temperatures for scram with the safety rods and the automatic backup shutdown (ABS) system. The curve for the ABS is superimposed over the curve for the safety rods which start decreasing in temperature earlier as shown by the figure. Even for the ABS, the maximum coolant temperature increase is less than 1°C, even though the inlet temperature has increased almost 3°C. This is due to the reactivity-induced power decrease. Since the maximum coolant temperature is less than its 103°C saturation temperature by about 19°C, no bulk boiling (and potential flow instability) would occur. These predictions were calculated with the AA3 computer code.

2.2.1.3 Conclusions

As indicated in Section 2.1.1, the transient results describe[illegible] this section are provided as an example of the type of evaluat[illegible] performed. The analyses were initiated from conditions for [illegible] operating limit which would typically result in lower [illegible] steady-state operating temperatures than those allowable [illegible] either the transient protection limit (TPL) or the confinement protection limit (CPL) as indicated by Table 2-3.

The transient temperatures and resultant consequences in each charge design are required to be within the acceptable values defined by: (1) steady-state operation at temperatures equal to or less than the TPL for shutdown by the safety rod system, or (2) steady-state operation at temperatures equal to or less than the CPL for the case where the safety rod shutdown is arbitrarily neglected and the ABS is assumed.

2.2.2 Loss of Cooling Water Reservoir Supply of River Water/Par Pond Water

2.2.2.1 Identification of Causes and Accident Description

If electrical power were lost to the pumps that supply water from the river or PAR Pond to the area cooling water reservoirs, then the water level in the reservoirs would decrease. There would be no immediate effect on the reactors, but the change in reservoir level would be detected by monitors and would lead to procedural shutdown of the reactor. To conserve water to the reservoirs, these procedures also call for immediately reducing the cooling water flow out of the reservoirs to about a tenth of normal full-power flow and to begin partial recirculation of the cooling water. The latter is accomplished by pumping water from the effluent basin back to the 25-million-gallon reservoir.

A bounding accident relative to the loss of cooling water reservoir supply of river water or PAR Pond water involves a postulated total loss of the 115 kV power, which would cause a total loss of water supply to the cooling water reservoirs. The accident is assumed to begin with the loss of offsite (commercial) power. Next, it is assumed that a loss of all 11 onsite generators that normally supply 40 percent of the electrical power to the 115 kV system also occurs. The latter event has never occurred but is considered to develop in a worst-case condition. If this overall sequence of total a.c. power loss should occur, then power from the emergency power diesel generators in each reactor area enables the recirculation of the cooling water, which extends the allowable time without ac power to several days.

2.2.2.2 Analyses of Effects and Consequences

An evaluation of the reservoir temperature response to a loss of cooling water reservoir supply of river water or PAR Pond water was performed for the bounding assumption of total loss of all a.c. power with recirculation pumps powered by the diesel-generators. Analyses show that sufficient heat capacity exists in the reservoir to meet cooling requirements for several days, as many as 12 days with optimized water usage practices.

If power is not restored at least partially to the rier pumphouses or the Par Pond pumps within several days, the reactor core would be discharged to the disassembly basin beginning within one day after shutdown. Complete discharge can be accomplished within 3 days. The disassembly basin holds about 3,000,000 gallons of H_2O. Natural convection in this basin would cool the core for several weeks, even without an outside supply of cooling water.

2.2.2.3 Conclusions

Cooling water malfunctions that result in a loss of the cooling water reservoir supply of river water or Par Pond water are very slow-acting. Because of the large reservoir heat capacity, the transient takes several days to reach the high temperature levels that would require discharge of the core. The core would be moved to the disassembly basin, which would keep it cool for several weeks by natural convection even without an outside supply of cooling water. This process would ensure a coolable in-place geometry. As indicated in Section 2.1.3, achieving these two conditions ensures the health and safety of the general public.

2.2.3 Loss of Cooling Water Inventory (Without Resultant Loss of Process Water Pumping Power)

2.2.3.1 Identification of Causes and Accident Description

As a conservative approximation to a double-ended break in one of the cooling water headers, a study was performed for an abrupt and total loss of cooling water.

2.2.3.2 Analysis of Effects and Consequences

As a result of the loss of cooling water inventory (conservatively simulated by an abrupt and total loss of cooling water), the temperature of the process water would increase at the reactor inlet. This would result in a power decrease until scram occurs.

The simulation of the pipe rupture by the abrupt and total loss of cooling water is conservative because it neglects the cooling water heat capacity and heat of vaporization. It is also extremely conservative since there is a low probability of a double-ended pipe break in both headers. However, even making these assumptions resulted in a very benign temperature response, as shown in Figure 2-4, where it can be seen that the maximum coolant temperature increases less than 1°C prior to safety rod scram and less than 10°C for the ABS system scram. This small temperature increase occurs for the latter even though the inlet temperature has increased by over 40°C. This is due to the resultant power decrease from the moderator temperature coefficient reactivity feedback combined with the core heat capacity effects. Since the maximum coolant temperature is less than its 103°C saturation temperature by 10°C, no bulk boiling (and potential flow instability) would occur. Once this initial temperature peak is over, lower temperatures will be experienced since the cooling water supplied by the second header would actually be available to remove the post-shutdown core power (although it was conservatively neglected during the initial high-temperature phase of the analyses). This is more than adequate heat removal for maintaining core temperatures to less than those at normal operation. These predictions were calculated with the AA3 computer code.

2.2.3.3 Conclusions

As indicated in Section 2.1.1, the transient results described in this section are provided as an example of the type of evaluation performed. The analyses were initiated from conditions for the operating limit which would typically result in lower initial steady-state operating temperatures than those allowable for either the TPL or the CPL as indicated by Table 2-3.

The transient temperatures and resultant consequences in each charge design are required to be within the acceptable values defined by: (1) steady-state operation at temperatures equal to or less than the TPL for shutdown by the safety rod system, or (2) steady-state operation at temperatures equal to or less than the CPL for the case where the safety rod shutdown is arbitrarily neglected and the ABS is assumed.

2.2.4 Loss of Cooling Water Inventory (With Resultant Loss of Process Water Pumping Power)

2.2.4.1 Identification of Causes and Accident Description

A piping rupture from the cooling water system could result in flooding of the process water pumprooms inside the reactor. This could cause a total loss of process water pumping, which requires activation of the emergency cooling system (ECS). The evaluations for this accident condition are presented in this section. This accident is termed the loss-of-pumping accident (LOPA).

2.2.4.2 Analysis of Effects and Consequences

Flooding the D_2O pump motor rooms would be possible if there were a massive H_2O leak inside the reactor building. Calculations predict that a large leak rate of approximately 135,500 gpm can result from a double–ended break of one of the two inlet headers that supply H_2O to the heat exchangers. No credible mechanism for this postulated break has been identified. This water would flow to the -40 foot level, where the pump motor rooms are located. Flooding of the motors would begin at a water depth of 18 inches for the a.c. motors and 35 inches for the d.c. motors. If the leak persisted at the maximum rate, the a.c. motors would begin flooding in 1.1 minutes and the d.c. motors would begin flooding in 2.7 minutes. These times apply to L and K Reactors: the times for P Reactor are nearly double because of a larger building volume in P Area.

The time necessary to flood d.c. motors would be about 0.4 minute longer if sump pumps are operable. Also, the increased flow decay time caused by flywheels would add more time before process water circulation is actually lost. Furthermore, any action to stop or reduce the cooling water pumping rate would retard or terminate the flooding. It is thus assumed that the ECS must be available within the 2.7 minute time interval prior to d.c. motor flooding in K and L Reactors. (The times for P Reactor are nearly doubled because of the larger volume.)

A major H_2O leak would be detected first by the individual heat exchanger flow monitors, which would trigger a reactor shutdown about 1 second after the leak starts.

The ECS is designed to preclude melting and maintain a coolable inplace geometry for this accident. Based on emergency procedures, the ECS would be actuated by operator action. H_2O coolant would then be forced through the reactor and out into the reactor building. The release from the reactor would be through the vacuum breakers and a H_2O–filled U tube (D_2O–filled for P Reactor). This drives most of the D_2O coolant out ahead of the ECS flow and rinses the rest out over a longer period of time. The first 560,000 gallons of this flow will be contained in the closed 60,000 and 500,000 gallon tanks. Any additional water is directed into the earthen basin outside the reactor building. The majority of the radioactivity in the D_2O moderator (principally tritium) will be contained in the closed tanks. Some will be released up the stack, however, by evaporation of water within the reactor building and in the 60,000– and 500,000–gallon tanks (which are vented back to the building). There will also be some evaporation of the moderator from the earthen basin.

The FLOOD84 computer code is used for defining safe operating power limits that will ensure that the ECS can limit fuel damage to the Section 2.1.3 requirements. The experimental database for this code includes hydraulic data (for standard SRS assemblies and their components) taken at the low flows that are expected to occur during design basis LOPA conditions. Experiments have also established relative distributions of channel flow within assemblies at low flows and acceptable power levels for operating assemblies with such flows.

2.2.4.3 Conclusions

Evaluations have been performed for the LOPA transient initiated from loss of cooling water inventory accidents. By meeting the emergency cooling limit described in Section 2.1.3 a core coolable geometry is maintained. Radiological release studies indicate that releases from the tritium-contaminated D_2O are maintained within the 10 CFR 100 dose guidelines. These LOPA consequences are identical to those for the LOCA evaluations described in Section 2.6 (dose results tabulated in Table 2-12 and 2-13).

2.3 Decrease in Process Water System Flow Rate

For the SRS reactors, the process water system is the equivalent of the reactor coolant system on typical power-producing nuclear plants. A number of transients that could result in a decrease in the process water system flow rate are postulated. Several of these transients have been evaluated to ensure that the limiting conditions have been considered. These are:

- Loss of process water a.c. pumping power
- Combined loss of process water and cooling water a.c. pumping power
- Closure of rotovalves in process water loops
- Process water pump shaft break
- Loss of blanket gas pressure
- Localized flow blockages

All of these events can result in some core assembly heatup prior to shutdown by scram. Thus, these have been evaluated relative to the Section 2.1.3 accident analysis criteria in the following sections.

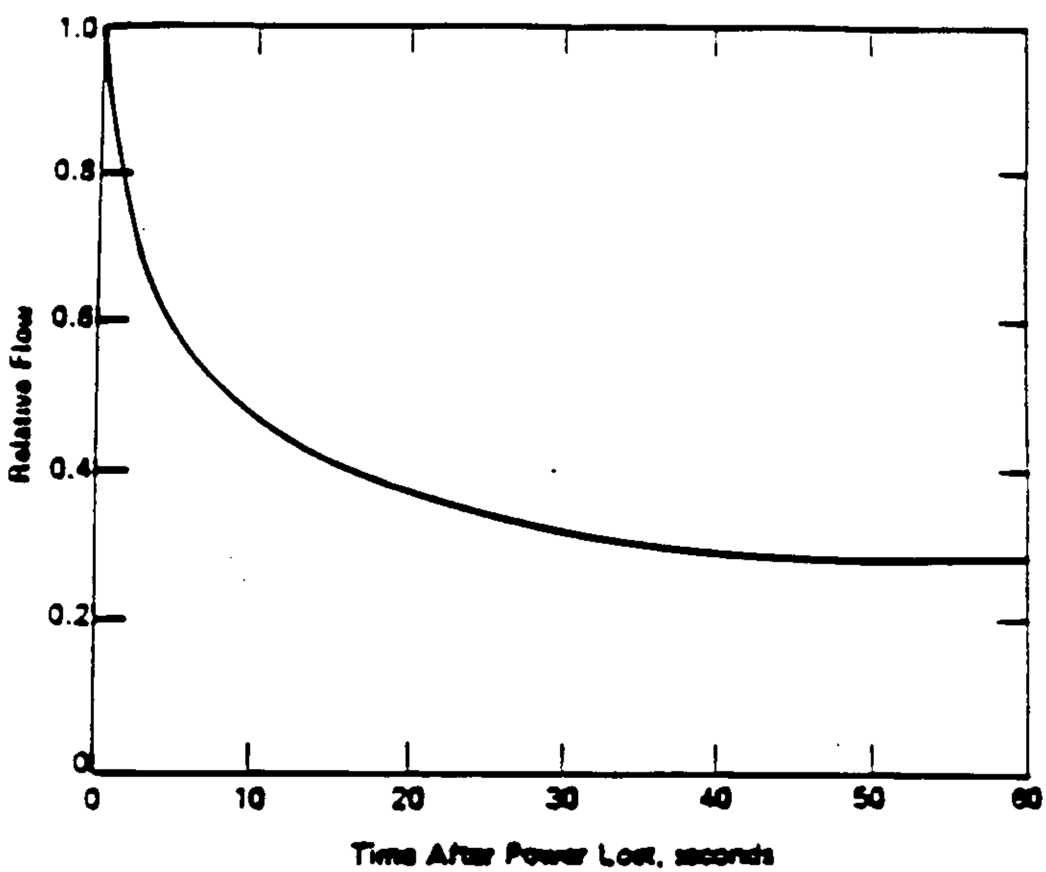

Figure 2–2

Flow Transients for Loss of AC Power to Cooling Water Pump Motors

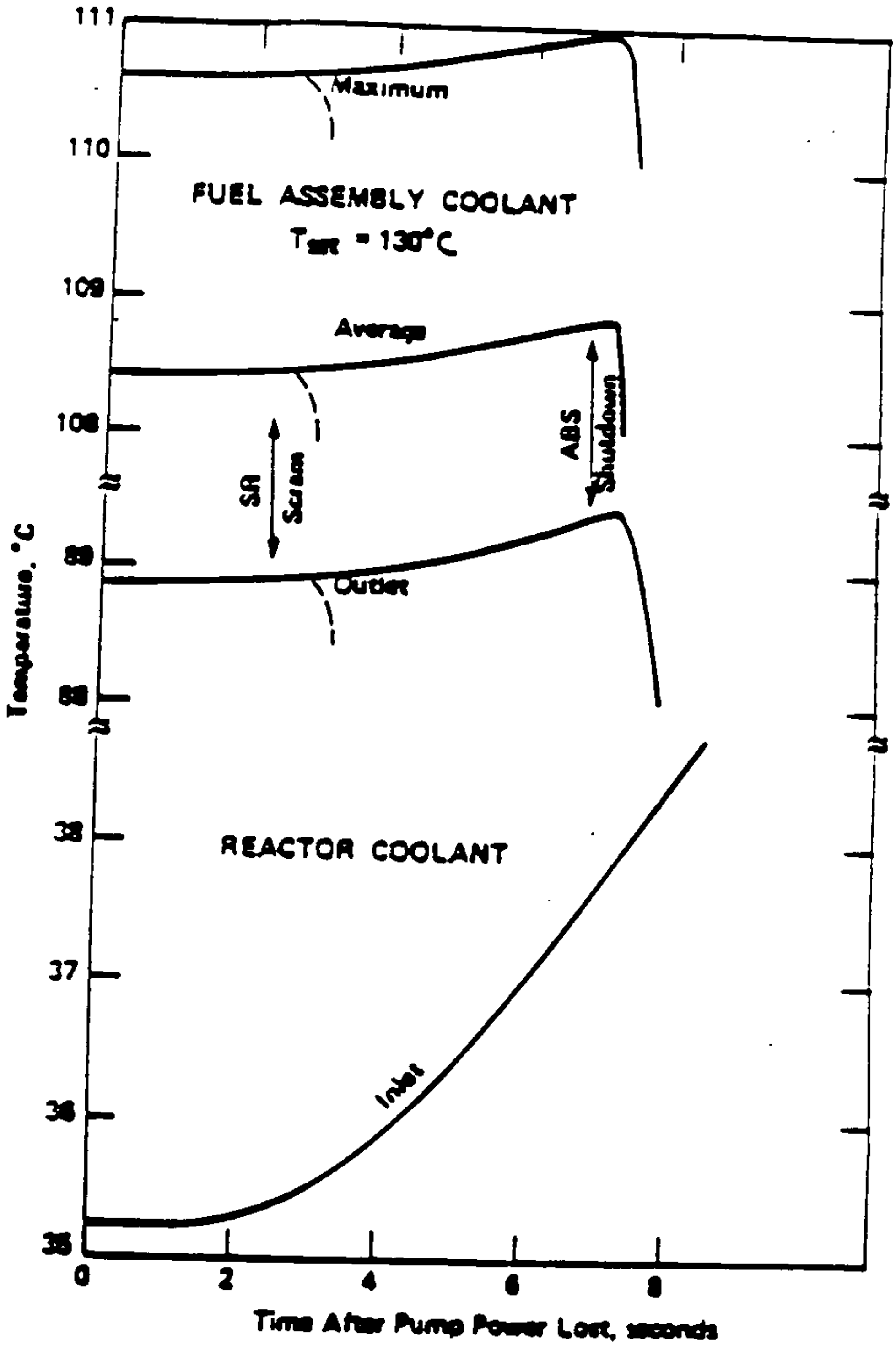

Figure 2-3

Temperature Transients for Loss of Power to Cooling Water Pumps

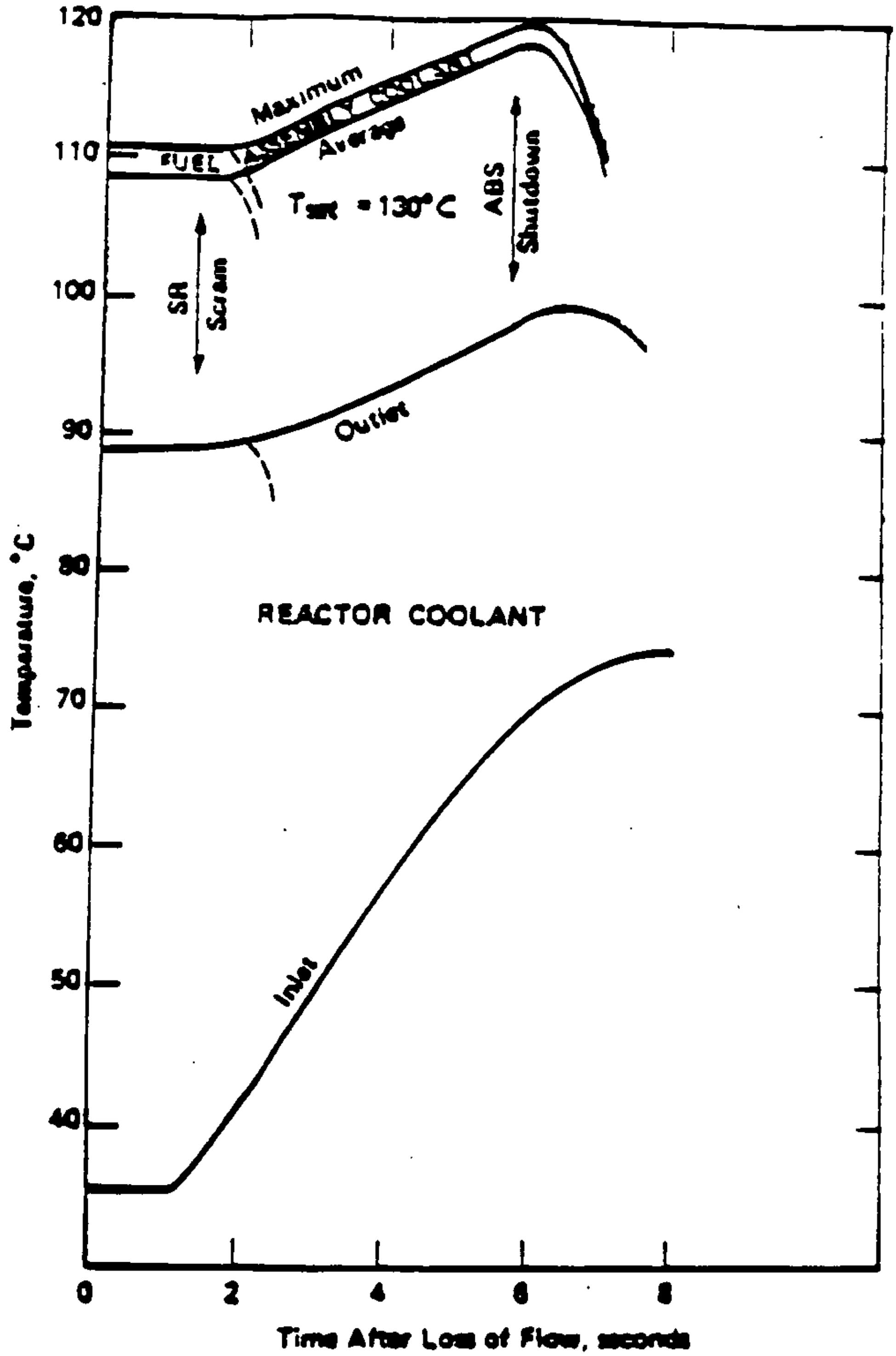

Figure 2–4

Initial Temperature Transients for Hypothetical Abrupt Loss of Cooling Water Flow

2.3.1 Loss of Process Water a.c. Pumping Power

2.3.1.1 Identification of Causes and Accident Description

Process water (D_2O) is circulated through the reactor by six parallel pumping systems. In each system, about 25,000 gpm is pumped (at a total head of 425 feet of water) from one of the six outlet nozzles at the bottom of the reactor, through two parallel heat exchangers, and into one of the six inlet nozzles in the reactor plenum. D_2O from the plenum enters the fuel and target assemblies through slots in the plenum tubes and slots or holes in the assembly sleeve-housings. Each assembly is suspended from the top of the water plenum, and the flow is controlled by flow orifices at the top of the assembly and/or by hydraulic resistance of the assembly itself. D_2O passes through the assembly and leaves through metering holes in the bottom end fitting. It then enters the moderator in the reactor tank, circulates through the tank, and then flows through the outlet nozzles back to the pump, completing the closed loop. Mixing in the moderator space is promoted by 1,800 gpm of D_2O which flows through the spargers into the moderator space. In addition, about 7,000 gpm of D_2O from the heat exchangers is diverted to supply upflow headers for cooling the control rods. Most of this D_2O flows through perforations in the control rod housings (called septifoils) to cool and agitate the moderator.

Each of the six main circulating pumps is driven by a 3,400-hp squirrel-cage induction motor that draws 125 amperes at full load (1,785 rpm) from a 13.8-kV, 3-phase, 60-hertz line. Pumps and motors are separated into groups of three in each of two pumprooms and two motor rooms. The pump is driven by a shaft extending through the four-foot concrete wall separating the pumproom and the motor rooms. The a.c. motor speed is reduced to the 1,000 rpm pump speed by a gear reduction unit. Each a.c. motor also drives a three-ton flywheel which stores sufficient kinetic energy to drive the circulating pumps and continue pumping D_2O for about four minutes after a complete power failure. Power supplies for the a.c. motors are divided between two substations.

Backup pumping capacity for D_2O circulation is provided by six d.c. motors coupled through gear reduction units to the a.c. motor drive shafts. All d.c. motors are normally online whenever the a.c. motors are operating. Their operation cannot be compromised by failure to start the diesels after a loss of a.c. power since operation of the a.c. motors is prohibited unless at least five of the process water pumps have online d.c. power.

pump can be without d.c. for up to 8 hours to allow for maintenance). Each d.c. motor is equipped with a nonreversing clutch to prevent reverse rotation. In the event of a.c. power failure, each d.c. motor will drive the pump at reduced speed to deliver about 29 percent of normal full flow to the reactor. Power to drive the six d.c. motors is supplied from eight diesel engine generator sets located in two engine rooms. Each d.c. motor is directly connected to its own diesel engine generator set. Two additional diesel engine generator sets serve as spares.

If a.c. electrical power to the D_2O pump motors fails, the reactor flow decreases as shown in Figure 2-5. As can be noted, the d.c. motors maintain approximately 29 percent of full flow. Because the d.c. motors are powered by the independent and continuously online diesel generators, a complete loss of flow including that driven by the d.c. motors is considered unlikely.

2.3.1.2 Analysis of Effects and Consequences

As the process water flow decreases, the core coolant temperatures increase. The resulting negative reactivity feedback from the negative moderator temperature coefficient causes the reactor power to decrease. This power decrease and the core heat capacity attenuates the rate of temperature rise. Because the flow is maintained at a sufficiently high value (i.e., 29 percent full flow), the flow decreases proportionately in all assemblies since the friction and momentum losses are the dominating factors determining core assembly flow rate.

As indicated in Table 2-7, the first scram instruments to respond to the loss of pumping power would be the two plenum pressure monitors. These would be backed up almost immediately by the assembly coolant flow monitors, and slightly later by the process water pump power supply scram signals.

The transient reactor power and fuel coolant temperatures predicted for the loss of process water a.c. pumping power are given in Figure 2-6. Here, it can be seen that the maximum coolant temperature only increases by about 5°C for safety rod scram and by less than 8°C for the automatic backup shutdown (ABS). Since the maximum coolant temperature is less than its 123°C saturation temperature by at least 4°C, no bulk boiling (and potential flow instability) would occur. As indicated earlier the core power is decreasing while the temperatures increase because of the negative reactivity feedback. These predictions were calculated with the AA3 computer code using conservative assumptions of the type described in Section 2.1.6. The results are typical of an enveloping evaluation as calculated for the various SRS charge designs.

2.3.1.3 Conclusions

As indicated in Section 2.1.1, the transient results described in this section are provided as an example of the type of evaluation performed. The analyses were initiated from conditions for the operating limit which would typically result in lower initial steady-state operating temperatures than those allowable for either the transient protection limit (TPL) or the confinement protection limit (CPL) as indicated by Table 2-3.

The transient temperatures and resultant consequences in each charge design are required to be within the acceptable values defined by: (1) steady-state operation at temperatures equal to or less than the TPL for shutdown by the safety rod system, or (2) steady-state operation at temperatures equal to or less than the CPL for the case where the safety rod shutdown is arbitrarily neglected and the ABS is assumed.

2.3.2 Combined Loss of Process Water and Cooling Water a.c. Pumping Power

2.3.2.1 Identification of Causes and Accident Description

If electrical power were lost to the pumps supplying cooling water directly to the reactor heat exchangers, the flow would decrease as shown in Figure 2-2. The flow levels out at a value of 25 percent of full flow, which would be sustained by gravity flow from the 25-million-gallon reservoir to the effluent sump.

Backup pumping capacity for D_2O circulation is provided by six d.c. motors coupled through gear reduction units to the a.c. motor drive shafts. Each d.c. motor is equipped with a nonreversing clutch to prevent reverse rotation. In the event of a.c. power failure, each d.c. motor will drive the pump at reduced speed to deliver about 29 percent of normal full flow to the reactor, as shown in Figure 2-5.

2.3.2.2 Analysis of Effects and Consequences

An evaluation of the decrease in process water heat removal from a combined loss of both process water and cooling water a.c. power was performed. As the process water and cooling water flows decrease, the core coolant temperatures increase. The resulting negative reactivity feedback causes the reactor power to subsequently decrease. This power decrease and the core heat capacity slows the rate of temperature rise.

One of the first scram instruments to respond to the loss of pumping power would be the two plenum pressure monitors.

These would be backed up almost immediately by the assembly coolant flow monitors, and slightly later by the process water pump power supply scram signals.

Results from the evaluation for transient reactor power and fuel coolant temperatures are given in Figure 2-7. Here it can be seen that the maximum coolant temperature increases less than 6°C for safety rod scram and less than 8°C for the ABS. Since the maximum coolant temperature is less than its 123°C saturation temperature by at least 4°C, no bulk boiling (and potential flow instability) would occur. The process water system performance dominates the core temperature response as would be expected (comparing results to those in Figure 2-6). These predictions were calculated with the AA3 computer code using conservative assumptions of the type described in Section 2.1.6. The results are typical of bounding studies as calculated for the various SRS charge designs.

2.3.2.3 Conclusions

As indicated in Section 2.2.2, the transient results described in this section are provided as an example of the type of evaluation performed. The analyses were initiated from conditions for the operating limit which would typically result in lower initial steady-state operating temperatures than those allowable for either the TPL or the CPL as indicated by Table 2-3.

The transient temperatures and resultant consequences in each charge design are required to be within the acceptable values defined by: (1) steady-state operation at temperatures equal to or less than the TPL for shutdown by the safety rod system, or (2) steady-state operation at temperatures equal to or less than the CPL for the case where the safety rod shutdown is arbitrarily neglected and the ABS is assumed.

2.3.3 Closure of Rotovalves in Process Water Loops

2.3.3.1 Identification of Causes and Accident Description

Rotovalves are installed in the external loops of the D_2O circulating systems between the heat exchangers and the reactor plenum (one rotovalve for each heat exchanger). During charge/discharge operations, the rotovalves are partially closed to reduce flow into the reactor plenum. Too much flow into the plenum during charge/discharge would cause the moderator to [illegible] out onto the [illegible]nk top with resulting loss of D_2O and release [illegible]itium. Du[illegible] postulated LOCAs that require emergency [illegible]ing system [illegible]S) addition, the rotovalves could be closed in any circulation system with a broken pipe. This would prevent backflow of the emergency coolant in that system, if it can be identified, unless the leak is between the plenum and the rotovalve.

Inadvertent closure of a rotovalve would reduce D_2O flow. Spontaneous closure of rotovalves has occurred on several occasions. On one occasion, both rotovalves in a single system closed simultaneously. The cause was a common mode failure in the automatic incident action (AIA) system that automates fast response of the ECS. This potentially made the system susceptible to spontaneous, simultaneous closure of all rotovalves in both of the two external loops connected to the AIA. Such an accident is no longer possible since the AIA was rewired to prevent reoccurrence of this event.

2.3.3.2 Analysis of Effects and Consequences

An analysis has been performed for the decrease in process water flow rate due to the simultaneous closure of all rotovalves in both of the two external loops connected to the AIA. This provides a bounding case relative to milder transients involving fewer rotovalves closing and/or slower closing rates.

The first scram source to respond would be the plenum pressure monitors, which would be backed up almost immediately by signals from assembly coolant flow rate instrumentation. The decrease in assembly flows causes temperatures to increase and the resultant negative reactivity feedback causes a decrease in reactor power. This power decrease, coupled with the core heat capacity, slows the temperature rise but does not reverse it until after scram occurs.

Figure 2-8 shows the flow decrease for average assembly and the most detrimentally affected core assembly. Assemblies with their inlet located near the loops which experience the valve closure are more detrimentally affected than assemblies more directly fed by the other loops. In fact, these other loops experience a flow increase since less system resistance exists at the lower plenum flow rates. Thus, instead of the average flow decreasing to 66 percent of the full flow it only decreases to about 72 percent full flow. The resulting power and temperature transients are given in Figure 2-9. Note that the maximum coolant temperature increases by less than 5°C for the safety rod scram and by less than 11°C for the ABS. Although bulk boiling occurs in the blanket and channels for the latter temperatures, the lifting force on the plenum was 210,000 lb, compared to the limiting value of 715,200 lb for the confinement protection limit criterion. These predictions were calculated with the AA3 computer code.

2.3.3.3 Conclusions

As indicated in Section 2.1.1, the transient results described in this section are provided as an example of the type of evaluation performed. The analyses were initiated from conditions for the operating limit which would typically result in lower initial steady-state operating temperatures than those allowable for either the TPL or the CPL as indicated by Table 2-3.

The transient temperatures and resultant consequences in each charge design are required to be within the acceptable values defined by: (1) steady-state operation at temperatures equal to or less than the TPL for shutdown by the safety rod system, or (2) steady-state operation at temperatures equal to or less than the CPL for the case where the safety rod shutdown is arbitrarily neglected and the ABS is assumed.

2.3.4 Pump Shaft Break

2.3.4.1 Identification of Causes and Accident Description

The equipment that might be involved in this accident is identical to the equipment identified in Section 2.3.1.1

Each of the six main circulating pumps are driven by a shaft extending through the four-foot concrete wall separating the pumproom and the motor room. The a.c. motor speed is reduced to 1,000 rpm pump speed by a gear reduction unit. Each a.c. motor also drives a three-ton flywheel which stores sufficient kinetic energy to drive the circulating pumps and continue pumping D_2O for about four minutes after a complete power failure.

The pump shaft break accident is postulated to be caused by breaking a process water pump drive shaft somewhere between the pump impeller and the flywheel. With the flywheel no longer connected to the impeller, fluid momentum drops to zero (and reverses) much more quickly than in the case of loss of pump motor power (Section 2.3.1).

2.3.4.2 Analysis of Effects and Consequences

An analysis has been performed for the decrease in process water flow rate from a pump shaft break transient. The first scram source to respond would be the plenum pressure monitors, which would be backed up almost immediately by signals from assembly coolant flow instrumentation. The drop in assembly flow causes temperatures to increase and the resultant negative reactivity feedback causes a decrease in reactor power. This power decrease, coupled with the core heat capacity, slows the temperature rise but does not reverse it until after scram occurs.

In addition to the fluid momentum quickly decreasing (and reversing from the pump shaft break accident), the flow perturbation to the core is not symmetric. Assemblies in the section supplied by the external loop, in which the pump shaft broke, would suffer a greater reduction in flow than other assemblies. Flow transients from the average and most detrimentally affected assemblies are given in Figure 2-10.

The resulting power and temperature transients are given in Figure 2-11. Note that the maximum coolant temperature increases by less than 10°C for safety rod scram (about 1°C margin to coolant saturation temperatures) and by less than 20°C for ABS. The ABS temperature exceeds the saturation temperature by about 8°C for about two seconds of the transient. The pressure of the resultant steam is not sufficient to lift the top shield/plenum structure and fail the roll anchors (calculated force was 212,000 lb, relative to limiting value of about 715,200 lb for a Mark 16-31B charge). These predictions were calculated with the AA3 computer code.

One important factor that can influence the course of this accident (and affect whether an enveloping evaluation has been performed) is the final condition of the broken pump shaft. If the shaft seizes, preventing the impeller from turning freely, flow will drop more quickly than if the impeller were free to rotate. On the other hand, a free impeller will have less resistance to backflow, yielding lower assembly flows at the end of the transient. Detailed evaluations of both conditions have found that the free-impeller case results in more restrictive limits, and this assumption has been used for the analysis in Figure 2-11.

2.3.4.3 Conclusions

As indicated in Section 2.1.1, the transient results described in this section are provided as an example of the type of evaluation performed. The analyses were initiated from conditions for the operating limit which would typically result in lower initial steady-state operating temperatures than those allowable for either the TPL as indicated by Table 2-3.

The transient temperatures and resultant consequences in each charge design are required to be within the acceptable values defined by steady-state operation at temperatures equal to or less than the TPL for shutdown by the safety rod system. Because of the very low probability of this event, it is not combined with the postulated failure of the safety rods in setting the confinement protection limits.

2.3.5 Decrease in Process Water Flow Rate from Loss of Blanket Gas Pressure

2.3.5.1 Identification of Causes and Accident Description

The equipment that might be involved in this accident is identical to the equipment identified in Section 2.3.1.1.

During reactor operation, the moderator is subjected to intense gamma radiation that results in the radiolytic decomposition of some of the D_2O molecules into deuterium and oxygen gases. These gases evolve from the moderator and must be collected and recombined to prevent the accumulation of explosive mixtures. A gas plenum containing helium is used to collect the D_2 and O_2; it is located between the top of the top shield and the bottom of the D_2O plenum. The bottom 5 inches of the gas plenum are filled with D_2O that entered from the reactor tank through the three gas ports and the annular space between the reactor tank and the top shield. Helium is used as the collector gas because it does not react with other gases, nor does it absorb neutrons to produce radioactive gases. The mixture of helium and evolved gases is swept continuously from the gas plenum by one of two blanket gas circulating systems (one circulating system is operated at a time). The blanket gas pressure is maintained at 5 psig, which increases the D_2O saturation temperature by about 4°C and thus permits operation at somewhat higher power levels. As indicated in Table 2-7, a reduction in blanket gas pressure to 4 psig results in a scram signal to the safety rods.

2.3.5.2 Analysis of Effects and Consequences

An analysis has been performed for the loss of blanket gas pressure transient. The cases evaluated have been formulated to bound less severe variations of the accident scenario. The blanket gas pressure decrease assumed for the evaluation was a linear decrease from 5 to 0 psig in two seconds. This conservative transient pressure decrease is about a factor of two quicker than experimentally measured rates of decrease from a mockup facility of the system.

As the pressure decreases, the reduction in saturation temperatures can cause cavitation in the external cooling loops (e.g., in the pumps) and a consequential reduction in total reactor coolant flow. The reduction in coolant flow would increase coolant temperatures, and the subsequent negative feedback would decrease reactor power. Both the reduction in pressure and the reduction in flow would lower the saturation temperatures inside the assembly. The normal operating limits for the safety rod scram system are set low enough to preclude boiling in the hottest subchannel.

Figure 2-12 shows the transient variations in reactor power and temperatures. The maximum coolant temperature increases less than 1°C prior to safety rod shutdown (giving over 20°C margin to saturation temperature) and increases less than 19°C prior to ABS actuation. The temperature at which ABS actuates exceeds the saturation temperature by approximately 10°C for about four seconds of the transient. The pressure of the resultant steam is not sufficient to lift the top shield/plenum structure and cause failure of the roll anchors (calculated force was less than 715,200 lb limiting valve for a Mark 16-31B charge). These predictions were calculated with the AA3 computer code.

Although evolution of some of the helium gas that had been dissolved in the D_2O could be a consideration, experiments have shown that this occurs slowly enough relative to the scram system response that it does not significantly affect the transient performance. The helium void fraction reached a maximum in the laboratory experiments about 15 seconds after an abrupt loss of pressure. Experiments in a production reactor indicated that this time was even longer. The laboratory results were used in computing the transients (Figure 2-13). A large negative void coefficient of reactivity tends to increase the severity of the reactor response to helium bubble formation and evolution. A large negative moderator coefficient that accompanies the void coefficient tends to reduce the power variation. The calculated transient of Figure 2-13 represents the power history for abrupt loss of helium pressure for a Mark VIB charge, assuming automatic shutdown systems fail. Here it can be noted that the resultant power variation is limited to a band between 94 and 105 percent full power. The analysis has not been repeated for current SRS reactor charges since either the safety rod scram or ABS would preclude power operation after four seconds; and thus the oscillatory effect shown in Figure 2-13 would not occur. Also, current charges have moderator coefficients (for both temperature and voids) much smaller than any of the charge designs considered in deriving Figure 2-13.

2.3.5.3 Conclusions

As indicated in Section 2.1.1, the transient results described in this section are provided as an example of the type of evaluation performed. The analyses were initiated from conditions for the operating limit which would typically result in lower initial steady-state operating temperatures than those allowable for either the TPL or the CPL as indicated by Table 2-3.

The transient temperatures and resultant consequences in each charge design are required to be within the acceptable values defined by: (1) steady-state operation at temperatures equal to or less than the TPL for shutdown by the safety rod system, or (2) steady-state operation at temperatures equal to or less than the CPL for the case where the safety rod shutdown is arbitrarily neglected and the ABS is assumed.

2.3.6 Local Core Assembly Flow Blackages

2.3.6.1 Identification of Causes and Accident Description

The equipment that might be involved in this accident is identical to the equipment identified in Section 2.3.1.1.

Various methods can be postulated for causing local flow decreases to the core assemblies (e.g., blockages due to fuel swelling or foreign objects carried with the coolant, unseated septifoil, etc.). The bounding cases are given in the following discussion relative to local flow reductions for control rods and fuel/target assemblies.

Loss-of-target and loss-of-fuel accidents due to an assembly meltdown resulting from a large and abrupt loss of assembly flow are improbable. No credible initiating mechanisms have been identified. Even though this is the case, the loss-of-target accident (but not the loss-of-fuel accident) is considered in establishing confinement protection limits described in Section 2.1.3. The LOCA has been postulated in the SRS reactors accident analyses simply for conservatism since it was one of the events in the original list of cases considered for confinement protection evaluations.

2.3.6.2 Analysis of Effects and Consequences

2.3.6.2.1 Analysis of Effects and Consequences of Loss of Control Rod Cooling

Control rod coolant is supplied by the D_2O leaving the heat exchangers in the external loops. The three loops on each side of the reactor feed a header that normally supplies coolant to half of the control rod housings (called septifoils). Thus, a reduction in control rod coolant flow is possible either in one of these headers or in an individual septifoil. Reduction of flow in a single header is unlikely to cause total loss of coolant to the control rods supplied from that header because the two headers are connected to each other at each end by smaller, cross-tie lines of half the diameter of the headers. However, even with total loss of flow to a septifoil, the control rods in current charges will not melt because heat generation in them is too low. Cooling for this postulated situation would be provided by convection currents supplied through the 1.25-inch opening in the bottom of the septifoil and through rows of 0.285-inch holes found at 8-inch intervals along the sheath of the septifoil. Control rod melting would require burnout, beginning for a septifoil (with no forced flow) at a heat flux of over 200,000 pcu/hr-ft^2 by critical heat flux correlations. Pound centigrade unit is abbreviated "pcu". The maximum heat flux of the control rods (evaluated in routine limits calculations) is required to be less than this figure. Typical maximum heat flux values for control rods are 100,000 to 130,000 pcu/hr-ft^2 for operation at near the 2,500 MWt full power level.

2.3.6.2.2 Analysis of Effects and Consequences of Flow Reduction in a Single Assembly

Coolant flow to an individual assembly can be postulated to be reduced by various means and at various rates. These scenarios are considered in analyses of both transient protection and confinement protection limits.

One possibility considered is a gradual reduction in flow to a single coolant channel which might be caused by the swelling that accompanies failure of a uranium metal fuel slug in a depleted uranium target assembly. Eventually, the assembly coolant flow monitor would detect this transient and scram the reactor. If the safety rods fail, then the analyses are bounded by a case that is considered in establishing confinement protection limits. This is the case of an abrupt and complete flow reduction that leads to a single assembly melting accident. Such an accident is merely postulated as a way of melting an individual assembly. No reasonable initiating mechanisms have been identified to cause it to happen. It is used as a demonstration of the ability of the plant to confine the radiological consequences of such a bounding condition to within acceptable values.

2.3.6.2.2.1 Loss-of-Target

As indicated earlier, this accident is postulated to begin with the abrupt reduction in coolant flow to a fairly high-powered target assembly. The flow reduction is arbitrarily assumed to be severe enough and the target assembly power high enough to melt the affected assembly whether the reactor is shut down immediately or not.

In this accident, the target material is assumed to be removed from the reactor as fast as it melts. This gives rise to a reactivity addition transient that increases reactor power and distorts the distribution of power. Two alternative forms of the reactivity versus time are shown in Figure 2-14. The two forms correspond to two different melting models assumed for SRS targets. One form corresponds to the case in which melted material disappears while unmelted material holds its position. This case should apply to continuously extruded light targets (e.g., Mark 53 neptunium targets), as opposed to heavy targets comprised of stacks of slugs (e.g., Mark 31 depleted uranium targets). For the latter case, it is assumed that as melted material disappears any unmelted material above it drops down to take its place. Thus, the column of slugs seems to disappear from the top (like a candle burning) even though melting begins in the middle. This "candle" model is the lower curve shown in Figure 2-14.

The principal factors that affect the severity of a loss-of-target accident are the neutron absorption strength of the assembly (i.e., how much the reactivity increases because the target is lost), the heat capacity of the assembly, and the assembly power. The last two factors determine how quickly an assembly will melt. Also, as with single-control-rod accidents, the reactivity change is amplified by radial tilting of the flux distribution if the affected assembly is toward the outside of the reactor.

The course of this accident is not extremely severe for a target assembly with low ratio of power to heat capacity. Loss of flow would trip scram instruments, and the reactor could be shut down (even by the ABS) before melting begins. Thus, when the affected target does melt, the reactor would remain subcritical since targets are designed so that the reactivity increase caused by a postulated loss of a target will not override either reactor shutdown system.

For a target assembly with a higher ratio of power to heat capacity, the course of the accident is also straightforward if the safety rod scram system performs as designed. But, if the safety rod scram should fail, melting could begin before the ABS responds. In this case, the ensuing reactivity transient would cause increases in reactor and hottest assembly power, as illustrated in Figure 2-15.

Studies have been performed with the AA3 computer code to determine the consequences of the loss of target transient for a Mark 16-31A charge. The studies included the conservative assumption of neglecting negative reactivity from steaming. It was found, for all current charge conditions, that the automatic scram system would terminate the loss-of-target accident fast enough to prevent additional melting of other assemblies (even though some of the adjacent assemblies entered the flow instability regime), and that the primary pressures remained sufficiently low so as to not violate the confinement protection limit (described in Section 2.1.3).

Fission product release (except for noble gases and tritium) from the single melted target would be contained within the reactor and removed by the purification system (See Section 1.2.3.8). Tritium and noble gases would be contained in the reactor system and would escape containment only through blanket gas leaks or if the blanket gas system were vented. In the worst case and if the safety rod scram were to fail, the subsequent steam pressure would be relieved through the vacuum breakers and U tube. For this case the noble gases from the single melted target would be vented to the reactor room and released through the 200-foot stack.

2.3.6.2.2.2 Loss-of-Fuel

This accident is almost an exact counterpart of the loss-of-target accident described in the preceding section and is similarly considered to have an extremely low probability of occurrence. The essential difference is that one of the key assumptions invoked for targets is neither accurate nor conservative for fuel. This is the assumption that melted material disappears from the reactor as fast as it melts. Molten debris from aluminum-clad, uranium-aluminum alloy fuel could reach the moderator in the form of particles, some of which could be swept along with the flowing moderator. This would cause a temporary increase in reactivity (until the fuel debris is swept from the core), rather than the decrease that would result if molten fuel disappeared immediately. Counteracting this increase in reactivity could be steam voids in the moderator caused by heat transfer from the fuel particles to the moderator. However, if the particle temperatures are above the Leidenfrost temperature, the amount of steam generated would be limited.

Single assembly melting would have no greater radioactive release effects than other accidents discussed in this chapter. The fission product release (except for noble gases and tritium) would be contained within the reactor and removed by the purification system. Tritium and noble gases would be contained in the reactor and would escape to the containment only through blanket gas leaks or if the blanket gas system were vented. As was the case for the loss of target accidents, if steam pressures were sufficient to vent through the vacuum breakers and U tube, then the noble gases from the melted initiator would also be vented to the reactor room and released through the 200-foot stack.

2.3.6.3 Conclusions

Loss of flow to control rods does not cause a safety problem since they can operate at their full power level with a postulated zero coolant flow and not melt.

Neither loss-of-target nor loss-of-fuel accidents have ever occurred in the SRS reactors. While reductions in assembly coolant flow have been observed, all such reductions have been low enough to enable shutting down the reactor without melting the assembly. Automatic scram circuits exist for the assembly coolant flow to terminate flow reductions before damage would exceed that allowed for the transient protection limits described in Section 2.1.3.

Loss-of-target and loss-of-fuel accidents resulting from a large and abrupt loss of assembly flow have extremely low probabilities. No credible initiating mechanisms for them have been identified. Therefore, they are not considered in

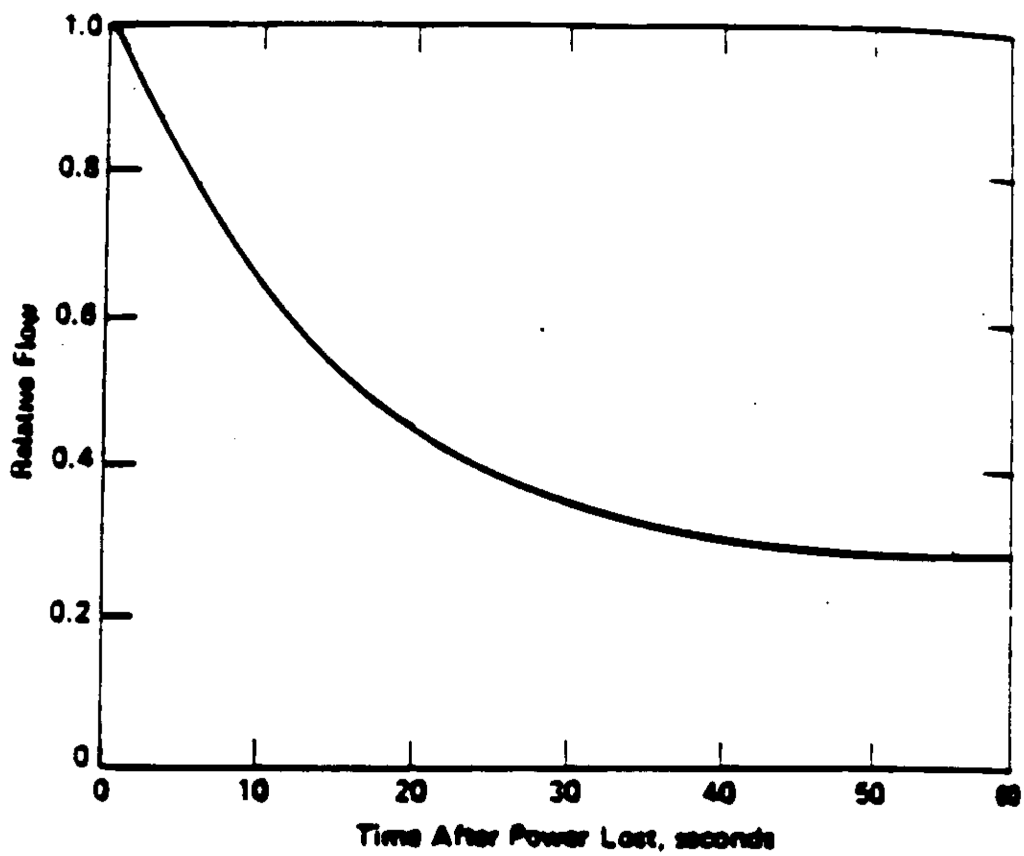

Figure 2-5

Flow Transient for Loss of AC Power to
D_2O Pump Motors

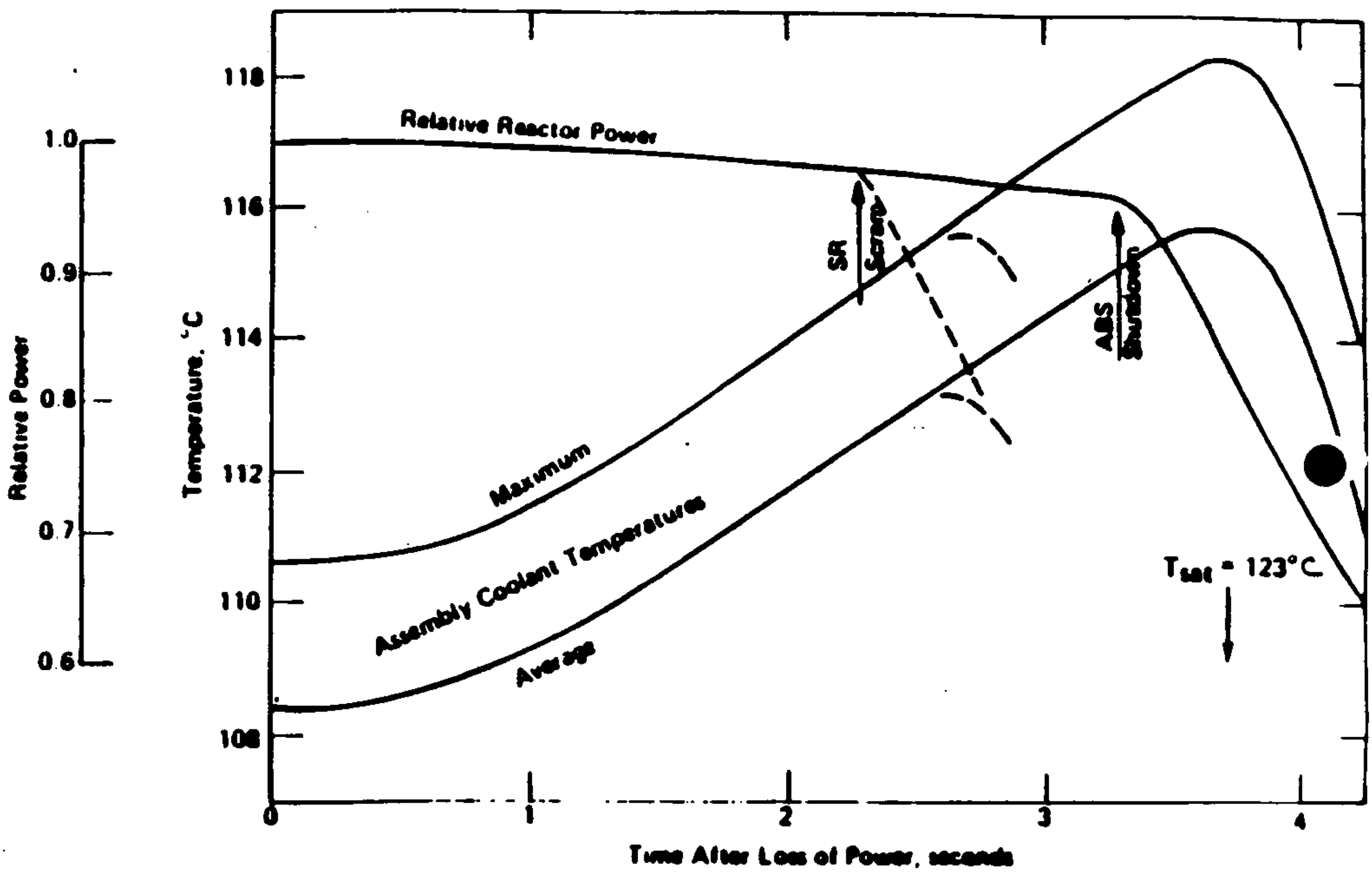

Figure 2–6

Power and Temperature Transients for Loss of AC Power to Process Water (D_2O) Pumps

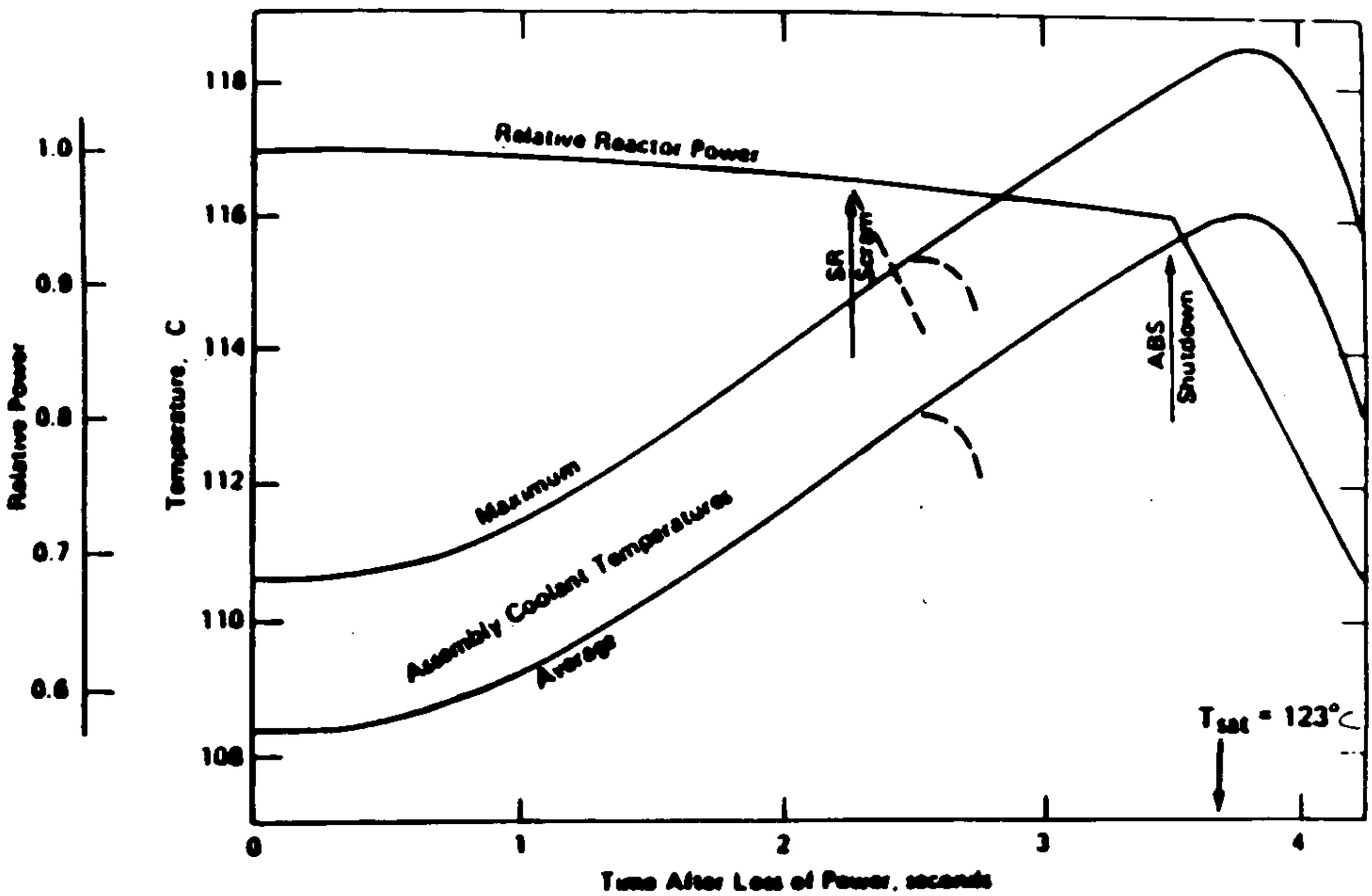

Table 2–7

Transients for Loss of AC Power to Both D_2O and H_2O Pumps

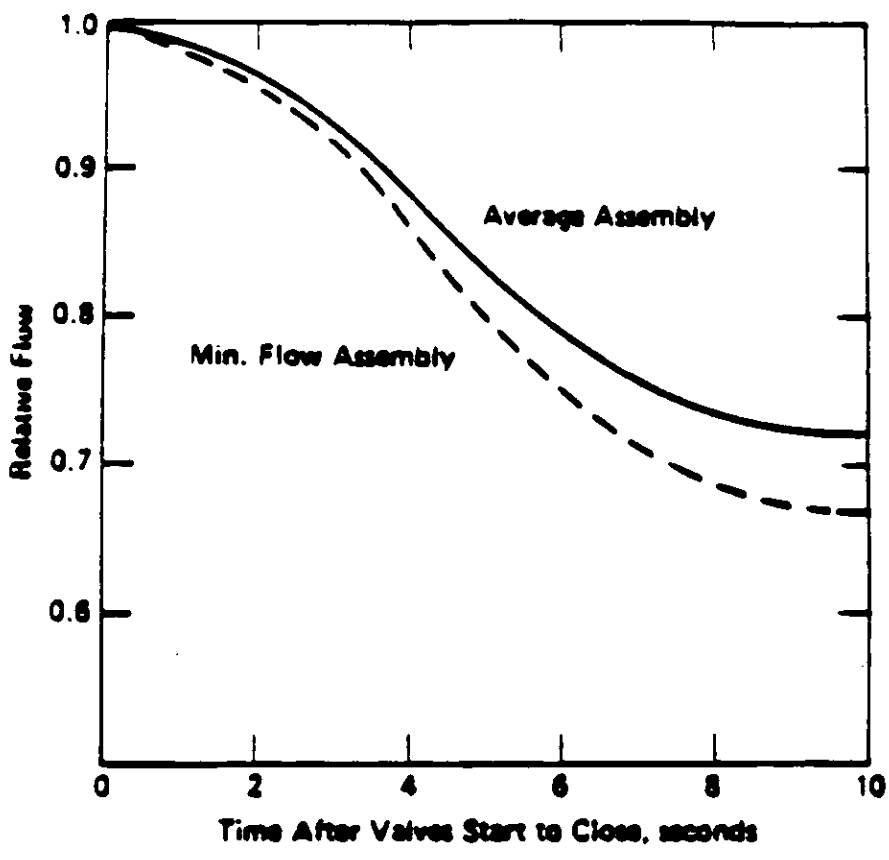

Figure 2–8

Flow Transients for Rotovalve Closure in Two External Loops

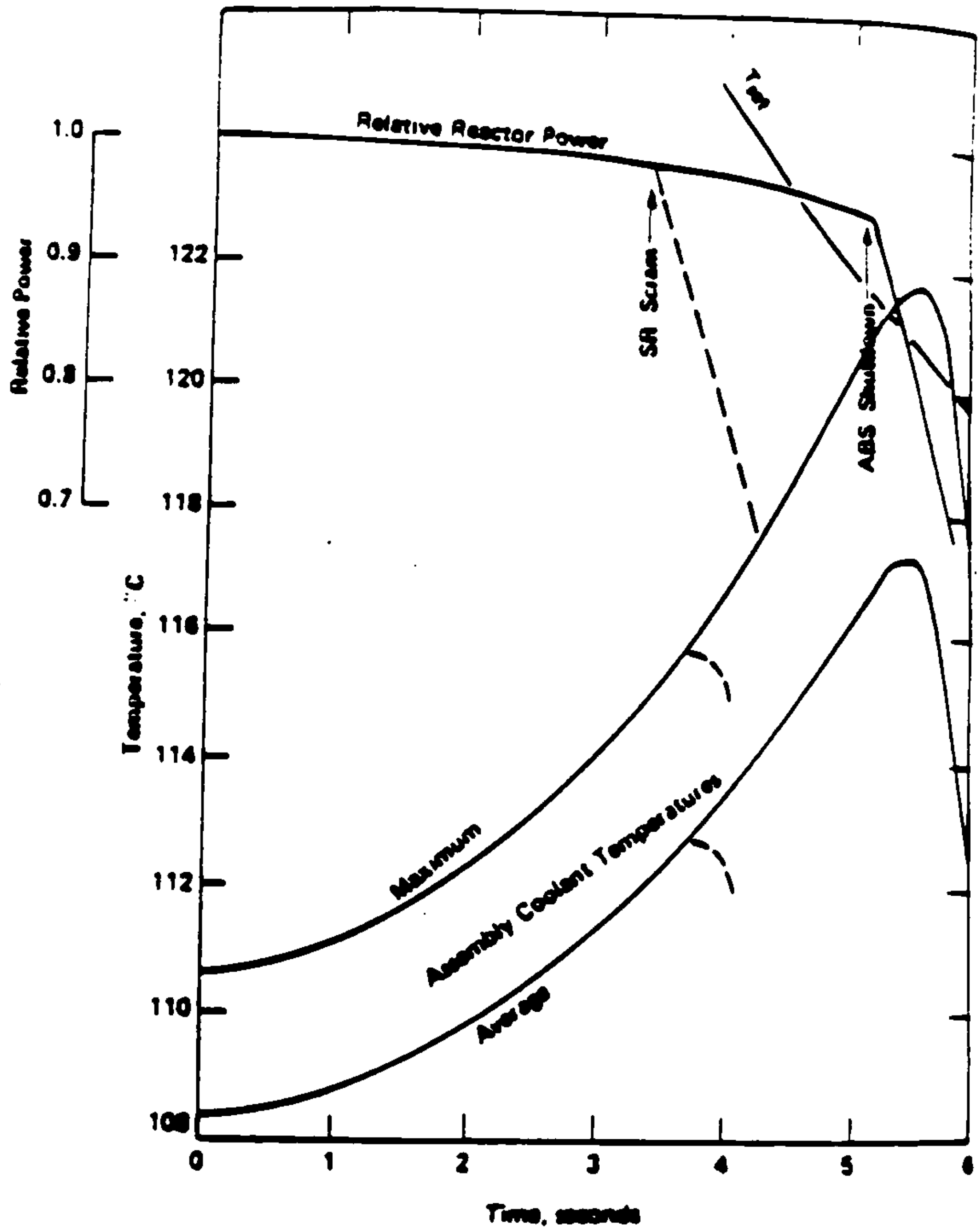

Figure 2–9

Power and Temperature Transients for Rotovalve Closure

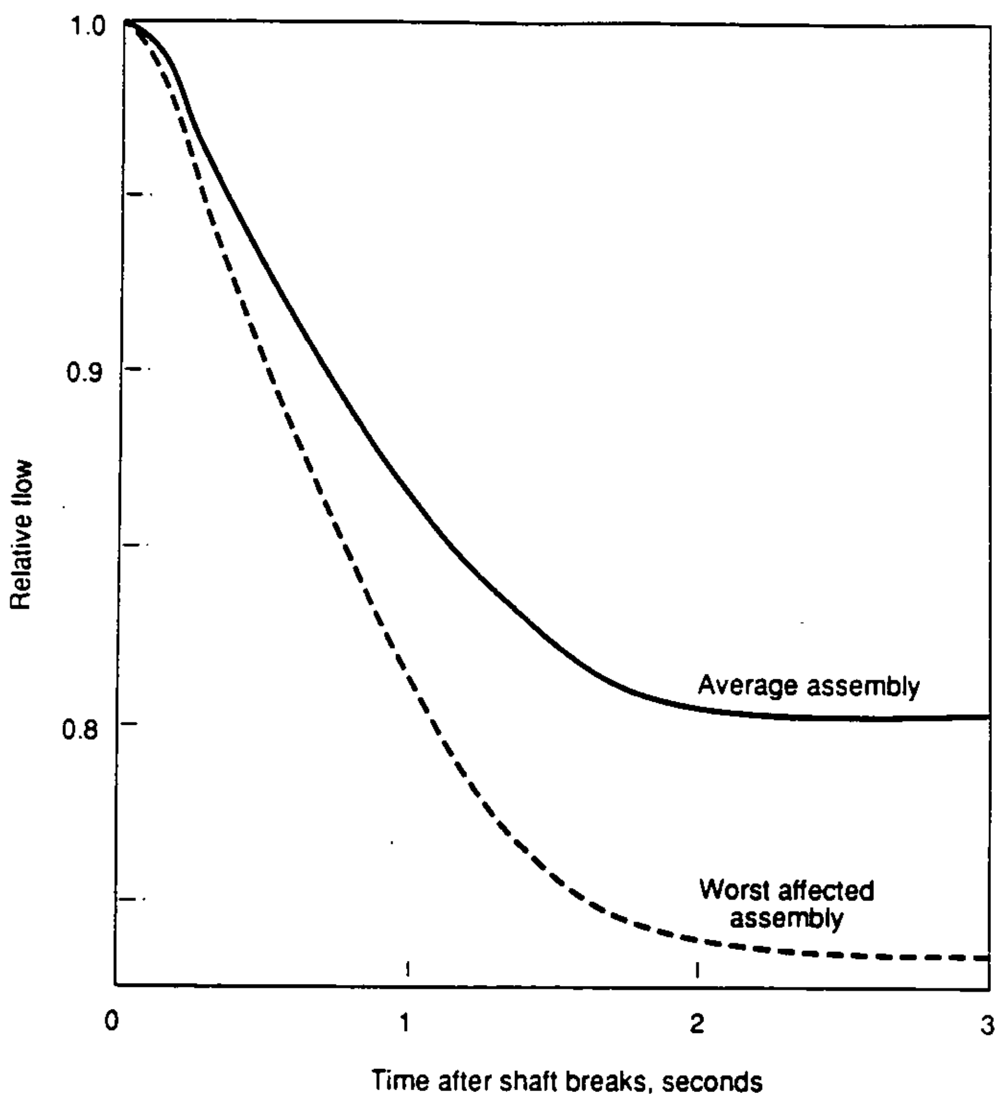

Figure 2–10

Flow Transients for Pump Shaft Break

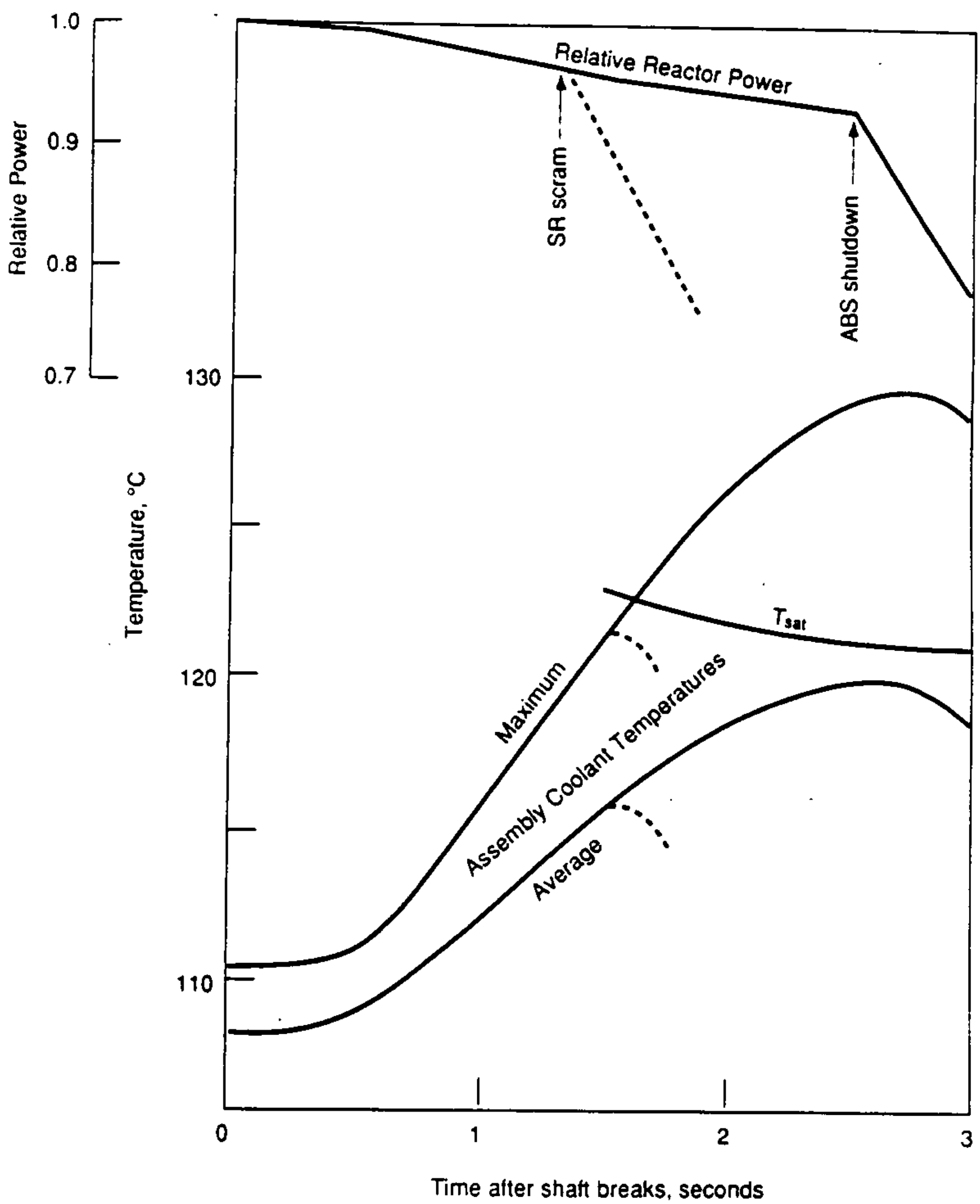

Figure 2-11

Power and Temperature Transients for Pump Shaft Break

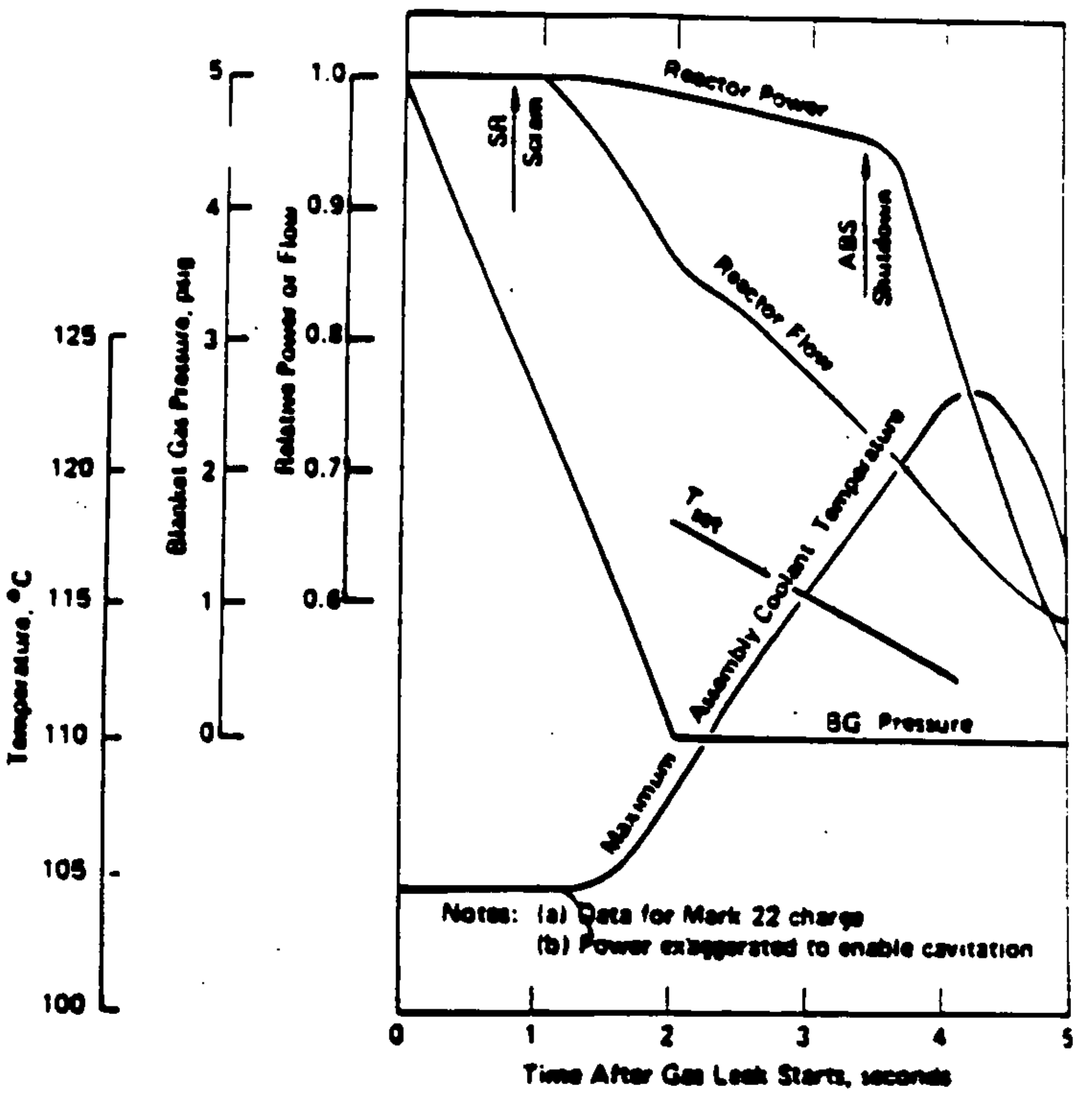

Figure 2–12

Possible Transients for Loss of Blanket Gas Pressure

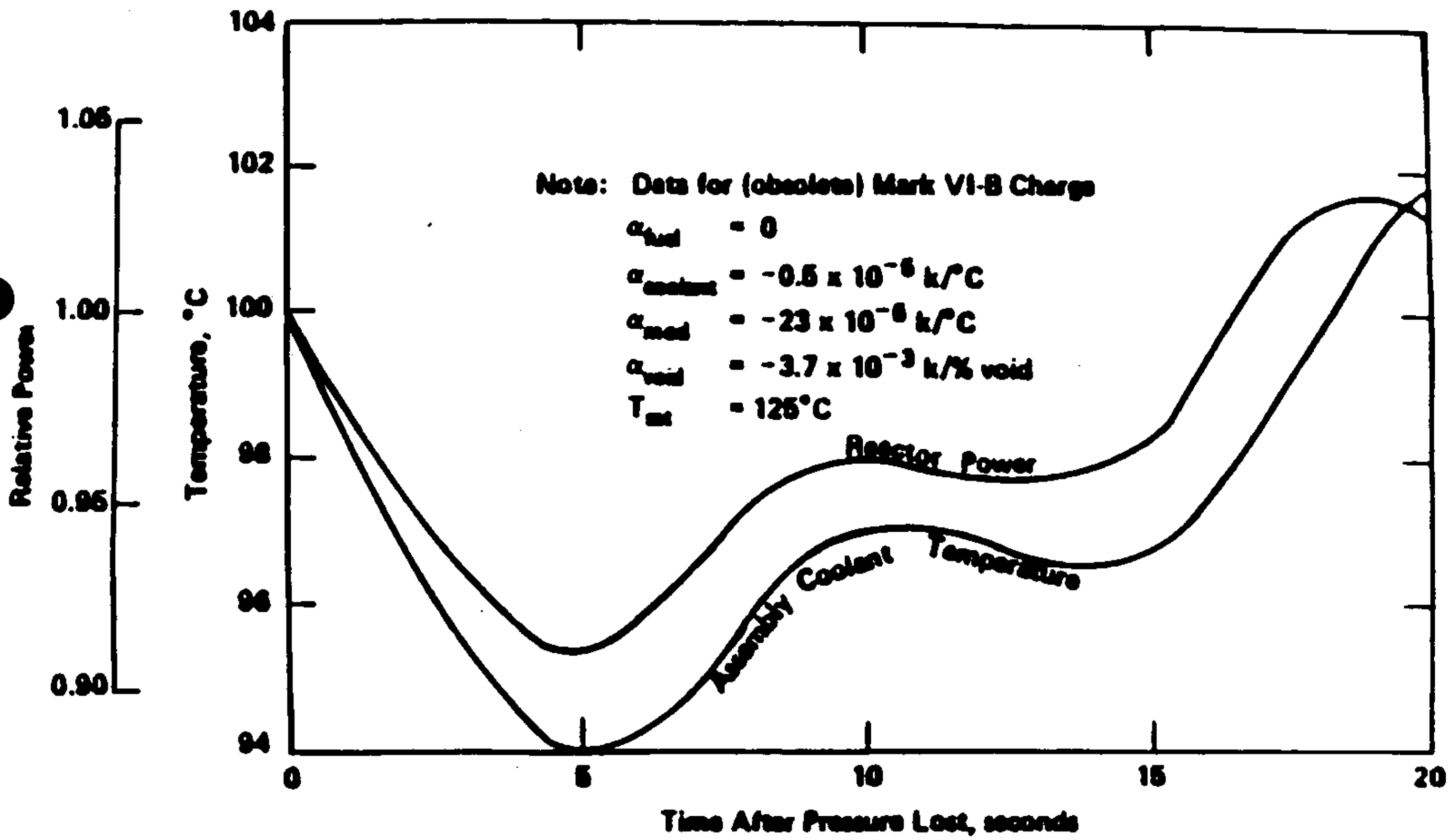

Figure 2–13

Power and Temperature Transients for Helium Bubbles Caused by Sudden Loss of Blanket Gas Pressure

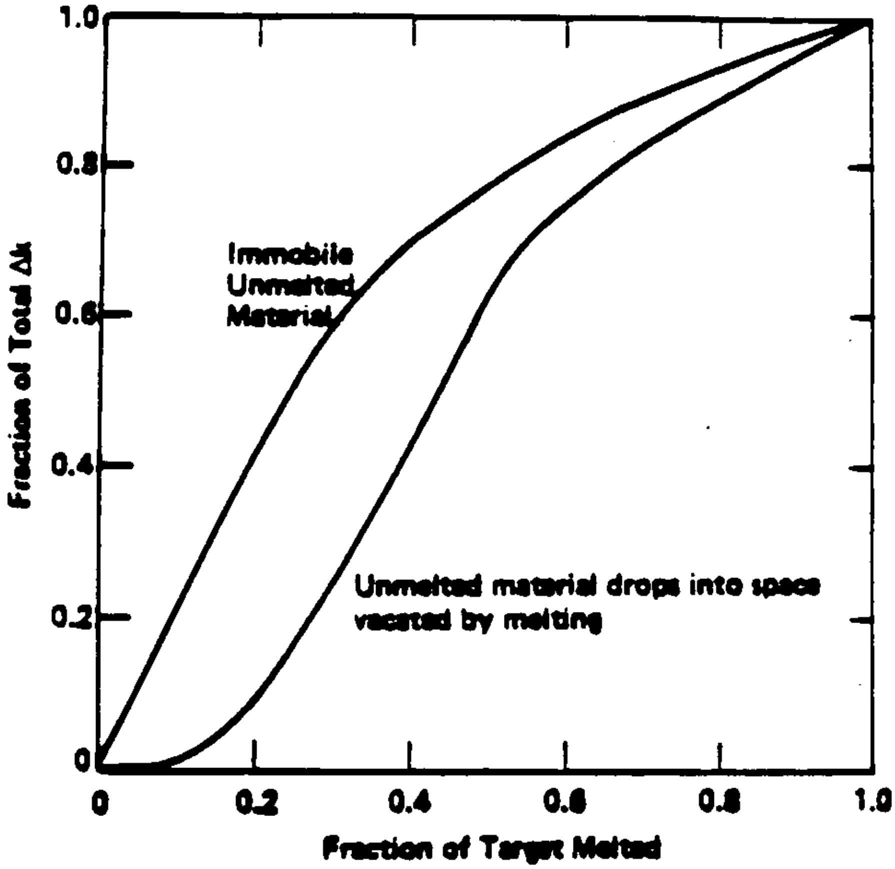

Figure 2–14

Approximate Reactivity Transient for Melting Target Assemblies

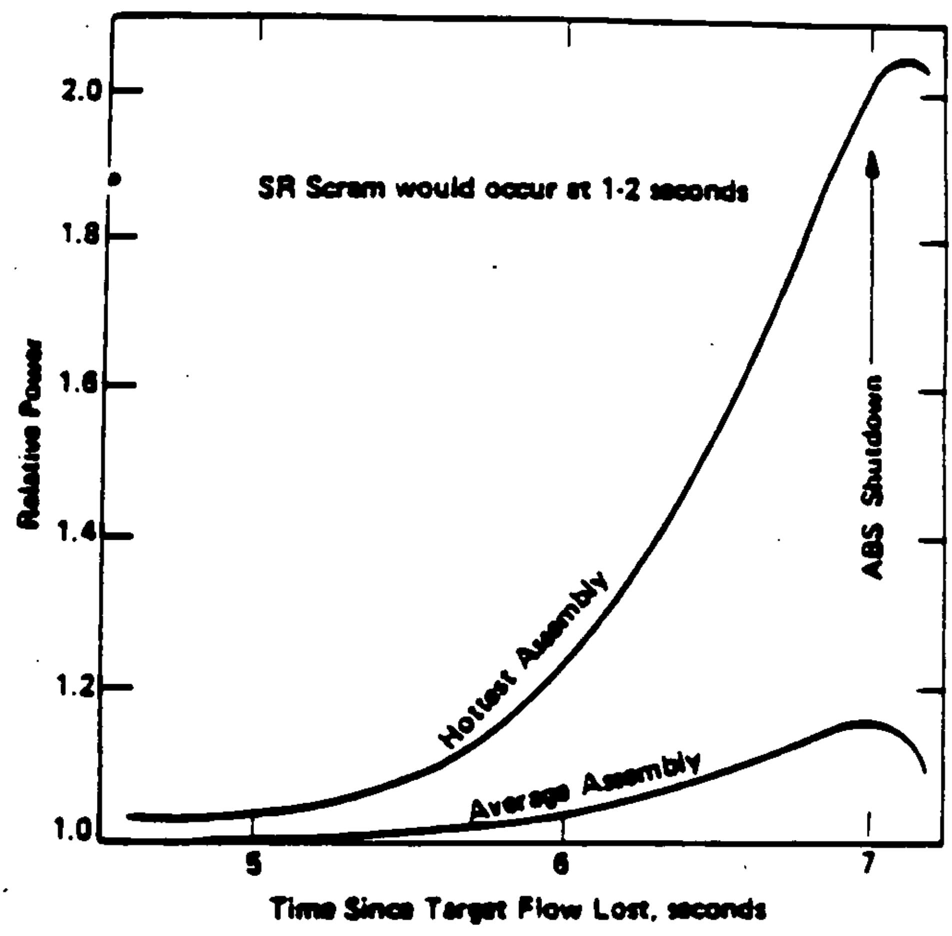

Figure 2–15

Assembly Power Transients for Loss of Target (Extreme Case)

2.4 Reactivity and Power Distribution Anomalies

A number of transients that could result in reactivity and power distribution anomalies are postulated for the Savannah River Site (SRS) reactors. Several of these transients have been identified as limiting. These are:

- Single control rod withdrawal/insertion accidents
- Gang rod withdrawal accidents
- Reloading errors

Several other transients cause reactivity insertion effects and are not addressed in this section since the source of their occurrence is not reactivity effects. This includes the meltdown of a core assembly due to a large and abrupt flow blockage (described in Section 2.3.6), accidents from a decrease in process water flow rate, the light water addition transient from operation of the moderator recovery system (described in Section 2.6.2.2), and accidents from an increase of reactor coolant inventory.

All of the accidents covered in this section can result in some core assembly heatup prior to shutdown. Thus, they have been evaluated relative to the Section 2.1.3 accident analysis criteria in the following evaluations.

2.4.1 Single Control Rod Withdrawal/Insertion Accident

2.4.1.1 Identification of Causes and Accident Description

Normal reactor operation requires that control rods be moved into and out of the reactor to accommodate the burnup of fuel and production in target material as well as other changing reactor conditions. Both speed and direction of control rod motion are controlled. However, inadvertent motion at maximum speeds can be postulated for both single rods and gangs of rods. The maximum control rod drive speed depends on the conditions of rod motion and how many rods are being driven. The maximum single rod drive speed of 0.136 ft/sec can be obtained only during insertion of rods; the maximum withdrawal speed is 0.118 ft/sec for a single rod.

Withdrawal of a single full-length control rod will add reactivity; its insertion removes reactivity except under very unusual circumstances such as uneven control rod burnup. Partial-length control rods, however, have their active portion at the bottom half of the rod. Normally these partial rods (two per cluster) are set with their active portion nearly axially centered in the core to flatten the axial flux and help achieve higher reactor power. in such cases, further motion of the partial rod, either in or out, will move the neutron absorbing portion of the rod to regions of less neutronic importance. The result is an increase in reactivity.

The rods can be driven into the reactor inadvertently either by spurious signals to the drive motors or by gravity if electrical power is lost. The drive motors can move the rods at either the normal control rod drive speed (0.022 ft/sec) or at the maximum speed of 0.136 ft/sec. A gravity–driven "drift" into the reactor has been observed on occasions when the electrical signal used to control the rod position is cut off completely. In this case, the only force tending to hold the rod in place is friction in the drive motor and other mechanical components. In some motor positions this force may be overcome by gravity, and the rod drifts a short distance until it encounters a position of stronger friction. Sometimes a rod will drift through several such steps. The average speed of this "drifting" motion has not been measured explicitly, but it is significantly less than the maximum speed of 0.136 ft/sec. Therefore, it can conservatively be assumed that the maximum drive speed bounds the gravity driven rate of an insertion.

2.4.1.2 Analyses of Effects and Consequences

2.4.1.2.1 Single Rod Withdrawal

The single rod withdrawal accident is initiated by assuming the inadvertent withdrawal of a single control rod. Removal of absorber adds reactivity, causing a rise in reactor power. The localized nature of the perturbation distorts the spatial distribution of power. In analyzing such accidents, it is assumed that the reactor is initially at full power, that the rod withdrawn is adjacent to the hottest assembly in the reactor, and that reactivity is added at a constant rate equal to the maximum rate possible. Under these circumstances, the powers of both the reactor and of the hottest assembly increase almost linearly with time. These power relationships can be expressed by:

Reactor power: $P(t)/P(O) = 1 + Rkt$

Hottest assembly: $P(t)/P(O) = 1 + (A + R)kt$

where k is the reactivity addition rate, t is the time since initiation, and R and A are factors of proportionality for the reactor and hottest assembly, respectively. For current reactor charges, R is about 120, while the peaking factor, A, for single rod accidents, may be in excess of 500. Thus, the hottest assembly power increases several times faster than the average assembly. The maximum reactivity addition rate for single rod withdrawal accidents in current charges is about 3 x 10^{-5} Δk/sec. Taking the derivative of the above relation for the increase in power of the hottest assembly and using the worst-case conditions for the parameters, the rate of power increase would be less than 2 percent of full power per second. At this slow rate of power increase, the temperatures nearly directly follow the power rise. A maximum temperature rise of less than 5°C would occur before shutdown by safety rod scram. The first scram instrument to respond to a single rod withdrawal accident would be the assembly coolant temperature monitor. As indicated in Section 2.4.2, the gang rod withdrawal has been found to cause more severe consequences if the primary scram does not occur; thus, the consequences of a single rod withdrawal with automatic backup shutdown (ABS) can be considered to be bounded by that analysis.

The speed, strength, and core location of the control rod withdrawn are the principal factors that affect the severity of this accident. Even with the maximum rod withdrawal speed of 0.118 ft/sec, about 2 minutes are required to move a rod from fully in to fully out. Maximum speeds are obtainable only under unusual conditions. For example, a special control device is capable of moving only one rod (either full or partial) at a time at maximum speed. An amplifier failure can, under certain limited circumstances, also cause this maximum drive speed. Strength refers to the neutron-absorbing capability of the rods being moved (i.e., the linear density of absorber). Withdrawal of a rod near the edge of the reactor "tiles" the normally flattened radial and azimuthal distribution of flux and amplifies the local peaking. The net effect is a large change (reactivity and local power distortion) as a result of rod withdrawal near the edge of the reactor as compared to withdrawal of the same strength rod at the center of the reactor.

Inadvertent movement of individual control rods in the SRS reactors has occurred about two to three times per reactor-year; most of the moves were insertions. None of these cases persisted long enough to actuate the scram system.

The temperature coefficients of reactivity and the lattice migration area are two reactor charge parameters that affect the consequences of a single rod withdrawal accident. The temperature coefficients express the amount of reactivity feedback from increased temperatures of metal components, coolant, and moderator. The migration area determines how localized the power distortion will be for a given rod withdrawal. Thus, heavy charges (small migration area) will exhibit less of a reactivity change but sharper (more localized) power peaking than light charges (large migration area).

2.4.1.2.2 Partial Rod Insertion

For the partial rod insertion accident, the rate of reactivity addition due to rod motion is proportional to the rod speed and to the increment of reactivity added per unit of distance moved ($\Delta k/\Delta x$). The maximum rod speed for insertion is about 15 percent greater than for withdrawal (as described earlier), but the maximum value of $\Delta k/\Delta x$ for full-length rods exceeds that for partial rods by at least 30 percent. Partial rod insertion accidents are thus currently bounded by full rod withdrawal accidents and need not be considered explicitly in limits analysis.

2.4.1.3 Conclusions

From the analyses of effects and consequences, it is concluded that the worst condition resulting from either a single rod withdrawal or a partial rod insertion transient would be a maximum coolant temperature increase of less than 5°C for the hottest core assembly terminated by a scram of the safety rods. This transient temperature and resultant consequence is within the acceptable value range as defined by steady-state operation at the temperatures equal to or less than the transient protection limit (Section 2.1.3). Evaluations for the confinement protection limits using the ABS are not presented in this section since they are bounded by the Section 2.4.2 gang rod withdrawal accident with ABS.

2.4.2 Gang Rod Withdrawal Accident

2.4.2.1 Identification of Causes and Accident Description

The equipment that might be involved in this accident is the same as that identified in Section 2.4.1.1. Groups of control rods, called "gangs," are moved together in normal operation. The maximum gang rod withdrawal speed of 0.022 ft/sec is much slower than the 0.118 ft/sec withdrawal speed for a single rod. All of the control rods are in one of three gangs. Each gang is azimuthally symmetric, with Gang 1 comprising the central portion of the reactor, Gang 2 the next region, and Gang 3 nearer the outer edge of the reactor core.

For accident analysis, a gang of control rods is assumed to move inadvertently starting at both full power and low power (the latter case simulates a startup accident). These two cases are evaluated in the subsequent discussion.

2.4.2.2 Gang Rod Withdrawal Starting at Full Power

This accident is similar to the single rod withdrawal accident (Section 2.4.1) in terms of cause, consequence, and controlling factors. Assuming a constant rate of reactivity addition yields a nearly linear power ramp characterized by:

Reactor power: $P(t)/P(O) = 1 + Rkt$

Hottest assembly: $P(t)/P(O) = 1 (A + R)kt$

where k is the reactivity addition rate, t is the time since initiation, and R and A are factors of proportionality for the reactor and hottest assembly, respectively. The R factor is the same for the gang rod withdrawal as for the single rod withdrawal accident (i.e., about 120). The peaking factor (A) is about 10 times lower for gang rod withdrawal than for single rod withdrawal (i.e., about 50). Compared to a single rod withdrawal, peaking is not as important in these accidents, but the maximum reactivity addition rate is much higher (about 10^{-4} Δk/sec in current charges compared to about 3×10^{-5} Δk/sec for a single rod). A gang rod withdrawal accident of the type postulated has not been observed in the SRS reactors. There have been incidents of a similar type, however, when the control computer attempted to raise power because of erroneous input signals. Such incidents have occurred about once every three reactor-years. A significant difference between the two types of incidents is that withdrawal of control rods by the control computer is not continuous and would be terminated when the temperature signal equals its operating limit.

The factors affecting consequences of a gang rod withdrawal accident are the same as for a single rod withdrawal accident with one exception. The gang rod withdrawal perturbation is always symmetric azimuthally, eliminating the radial and azimuthal "tilting" effect that influences single rod withdrawal accidents.

The first scram instrument to respond to a gang rod withdrawal accident is the assembly coolant temperature monitor. While temperature monitoring is relied upon in limits analysis, the neutron flux monitor could respond first in some cases. This can occur if the rod temperatures happen to be slightly above the elevation of one of three tiers of flux monitors external to the core, thus adding the perturbation in axial flux shape to the total in flux level.

Since the net effect of the rate of power rise is very close to that for the single rod withdrawal (Section 2.4.1) and the same scram signal occurs, the resultant maximum temperature increase of less than 5°C in the hottest assembly would similarly result in safety rod scram.

Relative to shutdown by the ABS, various rod withdrawal accidents in the SRS reactors were evaluated using the AA3 computer code. It was found that the most limiting case occurred for gang rod withdrawal at high (rather than low) reactor power. Results of these studies show that the consequences are acceptable relative to those confinement protection limits.

2.4.2.3 Gang Rod Withdrawal Starting at Low Power

A gang-rod withdrawal accident starting at very low power (three decades below full power) is analyzed for two purposes:

- To establish confinement protection limits
- To determine whether some minimum downtime is required before a reactor may be started up after a shutdown

The key factor involved in this analysis is the inventory of the fission product, Xe-135. This nuclide has a very large neutron absorption cross-section. At full-power equilibrium, it absorbs about 3 percent of all of the neutrons produced in the reactor. When a reactor is shut down after operating at full-power equilibrium, the inventory of xenon builds up to several times its full-power inventory (by decay of another fission product, I-135) as illustrated in Figure 2-16. If the reactor were restarted with a large inventory of xenon, burnup of the excess would add reactivity. At high flux levels, this phenomenon can add reactivity at a rate of roughly:

$$Xe(t)/Xe(0) \times 10^{-5}\ \Delta k/sec,$$

where Xe(t) and Xe(0) represent the current and equilibrium xenon inventories, respectively. At high values of Xe(t)/Xe(0), reactivity addition from xenon burnup could exceed that caused by rod withdrawal.

No downtime limit is required for current charges although one was required for earlier high flux charges. A xenon inventory limit is calculated at startup for each type charge by analyzing the accident assuming the maximum inventory present. If the ABS provides confinement protection with the full inventory present, then no limit is required. If not, the xenon inventory must be reduced until confinement protection is provided. A limit is implemented by requiring some minimum downtime following a reactor shutdown. This allows the xenon inventory to progress through its maximum value and decay to an acceptable level.

Another phenomenon peculiar to low-power accidents is that the temperature coefficients of reactivity do not provide appreciable negative feedback until reactor power reaches levels within a decade of full power. Thus, abnormally fast power transients can be developed at low powers. Still, reactor power asymptotically approaches the same power ramp as occurs during the full-power gang rod withdrawal and single rod withdrawal accidents.

The first scram instruments to respond to gang rod withdrawal at very low power would be the neutron flux instruments, the high level flux monitors, and the period monitors. These are backed up by the assembly coolant temperature monitors before reactor power gets within 50 percent of full power. Thus, core temperatures are bounded by the gang rod withdrawal accident initiated from full power (since the control rods are only pulled at lower power after full process water flow has been established).

2.4.2.4 Conclusions

As indicated in Section 2.1.1, the transient results described in this section and provided as an example of the type evaluation performed. The analyses were initiated from conditions for the operating limit which would typically result in lower initial steady-state operating temperatures than those allowable for either the transient protection limit (TPL) or the confinement protection limit (CPL) as indicated by Table 2-3.

The transient temperatures and resultant consequences in each charge design are required to be within the acceptable values defined by: (1) steady-state operation at temperatures equal or less than the TPL for shutdown by the safety rod system, or (2) steady-state operation at temperatures equal to or less than the CPL for the case where the safety rod shutdown is arbitrarily neglected and the ABS is assumed.

2.4.3 Reloading Errors

2.4.3.1 Identification of Causes and Accident Description

Reloading of the SRS reactors is done with strict inventory control. For the purpose of analysis, it has been postulated that a small region of a reactor could be made super critical (causing a power excursion) by incorrect reloading operations. If the reactivity is large enough, the power excursion would continue until termination of the nuclear chain reaction by fuel melting in a highly localized region of the reactor core. Two mechanisms for starting such an accident occur when the charge being reloaded is a mixed lattice, such as the Mark 16B-31. Normal reloading operations call for discharging a target assembly in one step, and charging a fresh target to that position in the next step. One postulated error is to discharge additional targets from adjacent positions without charging fresh targets. A second postulated error is to charge a fresh fuel assembly in place of a fresh target assembly. Such errors would add reactivity. To achieve criticality from either postulated error, charge design constraints, which limit the lattice reactivity to accommodate such errors, would also have to fail.

For current core designs, the postulated errors in reloading Mark 16B-31 charges described in the preceding paragraph have been evaluated as resulting in the largest reactivity change per incorrect reloading step. However, other types of errors could lead to about the same reactivity insertion. For example, the Mark 22 assembly includes both enriched uranium and lithium components. If some of the lithium components were omitted, or if the lithium were omitted from some of the components, then each fuel-replacement step would add more reactivity than intended. The likelihood of such errors is extremely small because individual fuel elements and slugs are tested to determine their effect on the reactivity of a neutron multiplying facility. Another example of a reloading error resulting in reactivity insertion is a charge design error, e.g., prescribing less lithium than actually needed. If the magnitude of such errors were large enough and the neutron monitoring systems did not detect the abnormality, calculations have shown that the reactor may experience a significant power increase (yet enveloped by the Mark 16B–31 boundary core).

2.4.3.2 Analysis of Effects and Consequences

Although reloading errors are possible in any charge, the mixed–lattice design (e.g., Mark 16B–31) is of particular concern because very large reactivity changes (> 1 percent k) can be effected by a single reloading step. To preclude such reloading errors, four measures have been taken:

(1) Special hardware modifications have been designed to lessen the probability of charging fuel into target positions.

(2) The reactivity of each new reactor charge is analyzed carefully, including a deliberate search for the worst possible reloading error. If the maximum calculated reactivity due to a misloading should exceed a criterion that is conservatively prescribed to prevent criticality, the charge would be redesigned.

(3) An improved system for monitoring the reactivity status of the reactor during reloading has been installed. The flux changes caused by reloading errors may be highly localized. To detect local power increases, a larger number of in–core neutron detectors is preferable to a smaller number of external detectors. The improved system uses six incore detectors.

(4) A computer system controls operation of the refueling machines. This computer is able to differentiate between fuel and target assemblies. With each reactor position being identified as either a fuel or a target position, the computer can stop the refueling machines if the wrong type of assembly is being charged. In addition, refueling operations are stopped if vacated positions are not refilled when required.

If a reactor was made critical by a reloading error, procedural injection of the supplementary safety system (SSS) gadolinium nitrate solution (Section 1.2.3.11.4) would be required. The safety rod scram system would be ineffective because all safety rods and control rods are already in the reactor during reloading operations. The reduced D_2O circulation rate and altered moderator flow pattern that exists during reloading operations would render the SSS much less effective than during normal operations. Several minutes would be required for the poison to reach the edge of the reactor where high-reactivity errors are possible.

The course of a postulated power excursion caused by reloading errors in a mixed-lattice charge has been calculated. The highly localized damage would involve less than 3 percent of the core.

Reloading accidents are not considered in establishing normal operating limits because fullpower parameters are not involved. Nor are they weighed against the conventional criterion for confinement protection because the process water system is open to the atmosphere during reloading. However, analysis has shown that the confinement system integrity would not be reduced by reloading accidents. Moreover, the charge-discharge process takes into account the effect of the most probable misloadings on reactivity.

2.4.3.3 Radiological Consequences

The results of the analyses presented in this section demonstrate that radioactivity released to the environment by a reloading error does not result in doses exceeding the reactor siting guidelines of 10 CFR 100.

The mathematical models used to calculate the activity releases during the course of the accident and the resultant doses are described in Chapter 3.

2.4.3.3.1 Release of Radioactivity

For 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, tritium, and iodine are released into the process room and below grade areas. It is estimated that 1 percent of the particulates (fission products, plutonium isotopes, etc.) would be released into the building, and would reach the HEPA filters, where 99.5 percent of the particulates (0.005 percent of total solids bypass) would be captured (see Section 2.4.3.3.2). Fifty percent of iodine in the melted assemblies is assumed to become airborne.

Tritium that might be released after melting would probably be in the form of tritiated water, primarily T_2O, because the high temperature (>600°C) and the presence of air would favor oxidation. Tritiated water is the most hazardous form biologically. Some tritium would probably be removed as moisture by the moisture separators, but no credit is taken for this.

Following a postulated melting accident, all airborne noble gases and tritium are assumed to be released from the stack. As discussed above, 99.5 percent of the airborne particulates are assumed to be captured on the HEPA filters and 99.95 percent of the airborne iodine (0.05 percent instantaneous bypass) is assumed to be trapped on the carbon beds. In comparison with other doses, the released solids are considered insignificant. 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. These desorption rates are used to calculate potential offsite doses, as discussed in Section 3.6.2. The NRC guidelines assume that all noble gases and 50 percent of the iodine are released to the containment building. Half of the airborne iodine is assumed to condense on various structural surfaces in the containment. Thus, 25 percent of the iodine inventory is assumed to be available for leakage from the containment. The postulated accident for SRS reactors assumes that 50 percent of the iodine available for release becomes airborne and reaches the carbon filters (no credit taken for spray system or condensation on structural surfaces). The remaining iodine would be held in the reactor or reactor building. Only a small fraction of this iodine would become volatile as organic iodides or other species.

2.4.3.3.2 Method of Analysis

A criticality could result from a reloading accident, as described in Section 2.4.3.1. Core damage would be less than three percent for this accident. For offsite dose determination, three percent core damage is assumed. There is no credible accident for which the computed core damage would exceed three percent.

In a release involving core damage, the release of noble gases, tritium, and iodine within the confinement system is discussed (See Section 2.4.3.3.1). Greater than 99 percent of the particulates (fission products and nonfission products) that become airborne are retained on the HEPA filters and do not contribute significantly to the dose. Volatile radionuclides would vent from the reactor and be released into the process room. The release could include tritium from targets containing lithium. Fission-product inventories would depend on neutron flux and are proportional to reactor power.

The potential offsite dose, from nonfission product isotopes (e.g., Pu-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 enough 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 calculation 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; namely, one percent of the inventory in the damaged portion of the reactor core is released to the building. For the maximum credible core damage of three percent, the assumed net release fraction is thus 1.5×10^{-6}.

2.4.3.3.3 Doses from Core Melting During Misloading Accident

Regulatory positions provide guidelines for siting a reactor in a manner that minimizes risk to the public to as low as is reasonably achievable. These guidelines specify reference values for the maximum dose an individual might receive at the outer boundaries of the plant. The reference dose values are 25 rem to the whole body and 300 rem to the thyroid. The individual is assumed to be at the plant boundary for two hours immediately following the postulated release of fission products, or at the outer boundary of the low population zone for two hours during the entire passage of the radioactive cloud. Because noble gases and tritium are not delayed before release, the dose from those isotopes beyond two hours will not increase. Noble gases and tritium contribute about 99% of these effective dose equivalent.

This analysis does not take credit for emergency plans to evacuate personnel beyond the plant boundary. The plant boundary and the low population zone for the SRS site are assumed to be identical for this analysis.

The "very unlikely" dose calculation uses specified meteorology that also has very low probability of occurrence. At the SRS, the meteorology determining the maximum two-hour, whole-body dose occurs less than 1.1 percent of the time. This and other parts of the calculation are discussed in Chapter 3.

The offsite radiation doses from core melting due to a reloading error are given in Table 2-8.

Although special procedures and engineered devices are provided to prevent loading errors, such an accident is analyzed to indicate the potential hazards if these safeguards were to fail. As discussed in Section 2.4.3.1, calculations indicate that less than three percent damage of the core would occur during this postulated accident. Prior to the accident, the fission products would have decayed for a minimum of 14 hours. As discussed in Section 2.4.3.3.2, to conservatively account for the additional fission products that would be formed during the postulated accident, i.e., to maximize the calculated dose, no credit is taken for decay prior to the accident. Even with this assumption, the computed dose at the plant boundary is small when compared to the 10 CFR 100 reference values of 25 rem whole-body dose and 300 rem thyroid dose (Table 2-8). Table 2-9 shows the dose to an individual onsite from a misloading accident at various distances from the affected reactor for typical and conservative conditions.

As noted in Section 2.4.3.3.2, there is no credible accident for which the computed core damage would exceed three percent. Doses from postulated releases higher than three percent of the fission product inventory are, for a specific meteorology, proportional to the percent damage and to the power level. And for the very unlikely conditions of meteorology and reactor power defined in Table 2-8, to exceed the NRC reference values for whole-body dose would require releasing into the reactor building more than 12.7 percent of the fission product inventory of the core. This is four times the maximum credible release. For thyroid doses to exceed the NRC reference values would require a corresponding release of 75 percent of the fission product inventory. Table 2-8 shows the typical and very unlikely doses for three percent core damage.

To provide an estimate of the relative magnitude of the potential offsite effects of several nonfission product isotopes included in particulates, full-charge inventories of several possible products have been calculated. The core inventory of several typical isotopes is shown in Table 2-10. The inventories are based on the production capability in a single reactor (except Pu-238 inventory, which is based on the availability of intermediates as feed material). Lesser amounts may be present in mixed lattices involving the production of several isotopes.

The quantity of each isotope that might be inhaled by a receptor at the plant boundary was calculated using the method described in Chapter 3. Fallout, deposition, and decay in transit to the plant boundary were neglected. The fractional release for all isotopes is based on three percent damage. Values for greater postulated damage are proportional to the assumed percent damage.

Values of potential doses from a maximum credible release of three percent of core inventories were computed for each isotope by multiplying the curies released by the relative concentration factor (X/Q) and an appropriate dose conversion factor. The calculation was similar to the inhalation dose calculations described in Chapter 3. The assumed breathing rate was 3.47×10^{-4} m^3/sec, and the relative concentration was that for the 99.5th percentile worst sector (see Chapter 3).

There are no official guidelines related to the inhalation of isotopes in a short time (as in a reactor accident). The most restrictive dose convers...

factors were used to determine the critical organ that received the highest dose and the effective dose equivalent to the entire body. The dose conversion factors were taken from ICRP-30. The insoluble form of U-233 was assumed, with the lung as the critical organ. The soluble forms of Pu-238, Pu-239, and Cm-244 were assumed, with the bone as the critical organ.

The effective dose equivalent from noble gases in the same reactor charge is not included in Table 2-10. The whole-body irradiation from exposure to gamma emitters would be added to the doses received from inhalation of particulates.

2.4.3.4 Conclusions

Reactivity and power distribution anomalies from reloading error type accidents could occur during shutdown conditions. However, analysis has shown that the intent of the 10 CFR 100 dose guidelines is met.

2.4.4 References

2.4-1 NCR Regulatory Guide 1.4. *Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors*, June 1974.

Table 2–8

Calculated Radiation Dose to a Person at the SRS Site Boundary Following a Reloading Error

Accident	Operating and Meterological Conditions (a)	Calculated Dose, rem: Effective Dose Equivalent (2-hr)	Calculated Dose, rem: Thyroid (2-hr)
Reference values for reactor siting in 10 CFR 100		25	300
Misloading Criticality (3% core damage)	Typical	0.51	0.39
	Very unlikely	5.9	4.01

(a) Typical conditions are 2,500 MW reactor power, average (50 percent) meteorology, and 19-month service age carbon filters. Very unlikely conditions are maximum anticipated reactor power of 3,000 MW, very unfavorable meteorology as specified in RG 1.145 (95 percent site, 99.5 percent worst sector), and 19-month aged carbon filter. Values shown are maximum for any of the P, L, and K Reactors. The core inventory of tritium is included in the whole-body calculations.

Table 2-9

Maximum Dose (REM) for Individuals Onsite at Various Distances[(a)]

Distance From Reactor, Miles	Misloading Criticality (3% core Damage) Very Unlikely	Misloading Criticality (3% core Damage) Typical
0.5	43.9	22.0
1.0	37.7	13.5
2.0	27.8	7.2
4.0	15.8	3.0
6.0	7.5	1.0
Plant Boundary	5.9	0.5

(a) This table represents the maximum effective dose equivalent at locations spanning the site directly under the plume. Very unlikely meteorological conditions are meteorological conditions not exceeded 99.5 percent of the time. Typical conditions can occur 50% of the time.

(b) Doses are the maximum of all 16 directions around each reactor for a two hour exposure.

Table 2-1[illegible]

Potential Offsite Doses from Nonfission Product Isotopes for 3 Percent Core Damage

Isotope[(a)]	Maximum Inventory, megacuries	Critical Organ Dose rem		Critical Organ	Effective Dose Equivalent rem	
		Very Unlikely	Typical		Very Unlikely	Typical
U-233	0.0005	0.001	<0.001	Lung	<0.001	<0.001
Pu-238	0.15	3.19	0.27	Bone	0.18	0.015
Pu239	0.022	0.54	0.045	Bone	0.029	0.005
Cm-244	0.25	3.16	0.26	Bone	0.18	0.015

(a) These numbers generally are based on a reactor charge producing a single product (the exception is Pu-238; see Section 2.4.3.3.2). If two or more products are being produced simultaneously in the same reactor, the maximum inventory of any one would be lower.

Release fraction of 5×10^{-5} times the fraction of core damage (0.03) for all isotopes. Dose reported is the maximum for P, L, and K Reactors at the SRS boundary. Very unlikely meteorological conditions are not exceeded 99.5 % of the time, typical conditions are not exceeded 50% of the time.

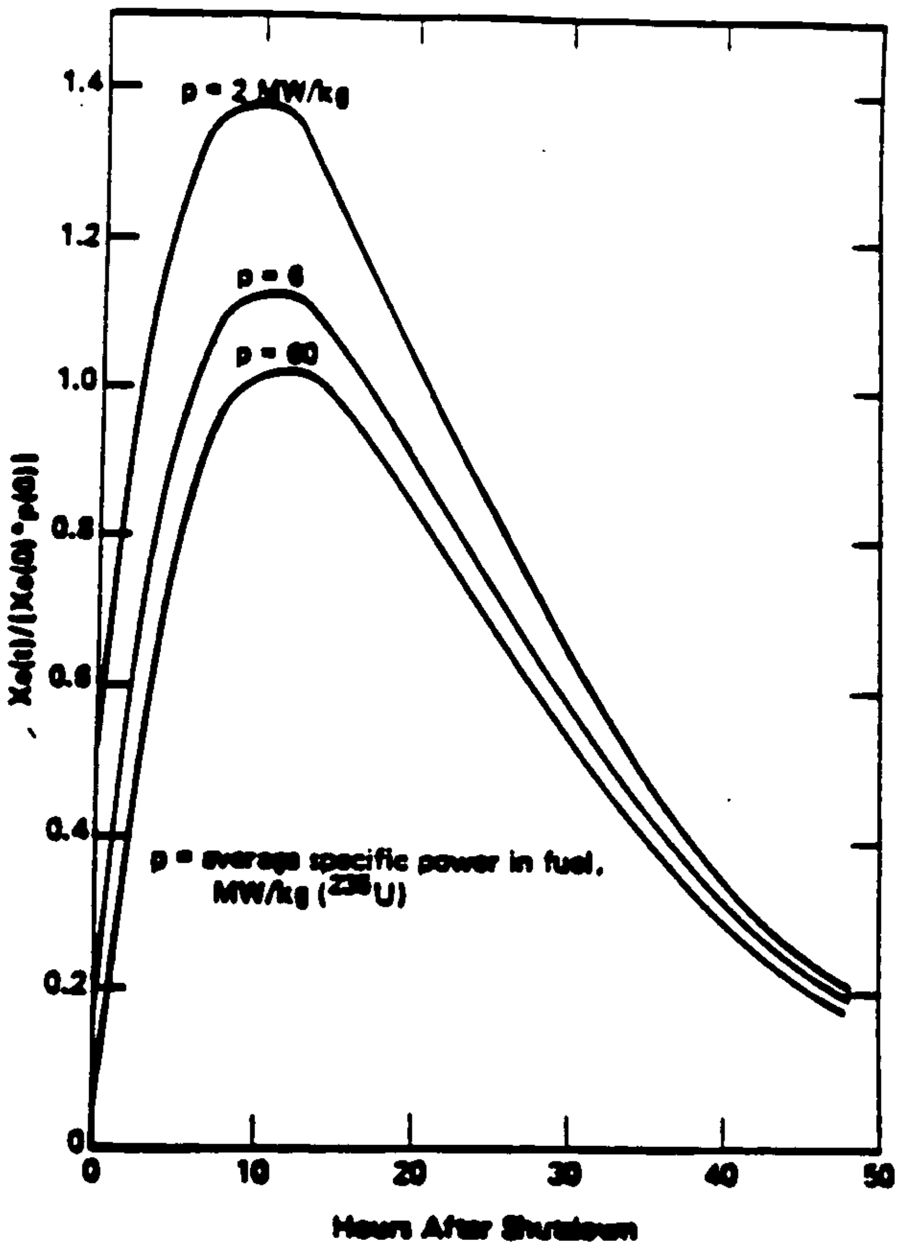

Figure 2–16

Relative Concentration of Xe–135 After Shutdown

2.5 Increase in Process Water System Inventory

Several events that could potentially result in an increase in the process water system (PWS) inventory or reactor coolant inventory may be postulated for the Savannah River Site (SRS) reactors. These events are:

- Inadvertent actuation of the emergency cooling system (ECS)
- Inadvertent actuation of the supplemental safety system (SSS)

Each of these events may result in the inadvertent of unwanted addition of coolant to the PWS.

2.5.1 Inadvertent Actuation of the Emergency Cooling System

2.5.1.1 Identification of Causes and Accident Description

The inadvertent actuation of the ECS may be caused by either a spurious signal from the automatic incident action (AIA) section, or by an operator error in manually initiating light water addition.

2.5.1.2 Analysis of Effects and Consequences

During normal power operation, the pressure in the ECS is less than the pressure in the reactor at the location where the ECS is delivered. Thus, regardless of how the ECS is actuated, either by spurious signal from the AIA or by operator error, no light water will flow from the ECS to the reactor. Operators will, however, be made aware of ECS actuation by means of illumination of incident action indication lights and graphic displays that are located both in the reactor control room and in the remote control center.

During charge and discharge operations, coolant is circulated through the reactor by means of six dc motors (one per pump) at a rate equal to about 30 percent of the full flow at power operation. At this reduced flow rate, the pressure in the reactor is reduced such that the ECS, if inadvertently actuated, can deliver light water to the reactor. The increase in reactor coolant inventory resulting from inadvertent operation of the ECS is vented and discharged from the reactor tank to prevent structural damage. Much of the moderator is driven out of the reactor through the vent paths ahead of the ECS flow.

There are three vent systems for the discharge of the water: (1) two vacuum breakers, (2) guide tubes and three relief tubes in the forest, and (3) a below-grade U tube. The vacuum breakers and the U tube provide a vent path from the gas space above the top shield. The guide tubes and three relief tubes in the forest provide a vent path from below the top shield. All light water supplied by the ECS is injected into the water plenum, and therefore must pass through the core assemblies before reaching the reactor vent paths.

During refueling operations, the blanket gas pressure is reduced to near atmospheric values. Upon inadvertent actuation of the ECS, any residual gases would be released through the vacuum breakers and the U tube vent. Then, about half of the excess cooling water would spill over the vacuum breakers, three forest relief tubes, and the forest guide tubes into the process room. The remaining excess coolant would be discharged through the U tube to the two–40–foot level pumproom sumps.

Each of the two pumproom sumps is provided with two sump pumps; a submersible 2500–gpm pump, and a 4500–gpm pump that has an elevated drive motor. The sump pumps will automatically actuate on high sump water level to move released moderator and, as the event progresses, light water added to the PWS by the ECS, into storage locations outside the reactor building. Fluid discharged by the 4500 gpm sump pumps will flow first to a closed 60,000–gallon underground storage tank in Building 106. Piping upstream of the 60,000–gallon tank is arranged such that, when the tank is full, the discharge from the 4500 gpm sump pump overflows into piping leading to a 500,000–gallon storage tank. Fluid dischanged by the 2500 gpm sump pumps, as well as any liquid that may be collected by the reactor room floor drain, flows directly to the closed 500,000–gallon storage tank. Piping upstream of the 500,000–gallon storage tank is arranged such that, when that tank is full, excess fluid supply overflows into the 50,000,000–gallon earthen basin. This system of sump pumps, tanks, and associated piping is part of the Contaminated Water Removal and Disposal System.

Inadvertent actuation of the ECS during refueling operations will result in the displacement of radioactive moderator from the reactor system. The majority of the radioactivity in the moderator will be contained within the reactor building and in the 60,000– and 500,000–gallon closed tanks, which are vented back to the reactor building. Some radioactivity, however, will be carried up the reactor building exhaust stack with water vapor generated from the displaced moderator. Also, any reactor inventory overflow from the 500,000–gallon tank into the 50,000,000–gallon earthen basin provides a direct path for the release of radioactivity to the environment.

The flow of excess reactor inventory for the inadvertent actuation of the ECS event is the same as for the loss of pumpage accident (LOPA) described in Section 2.2.4. The rate at which the ECS flow is delivered to the reactor is smaller for the inadvertent operation of the ECS than for the LOPA event because the reactor is at a higher pressure due to forced circulation being maintained by a minimum of three d.c. motors with throttled rotovalves. Thus, the radiological consequences of inadvertent operation of the ECS during refueling conditions are bounded by those of a LOPA.

As was the case with the inadvertent actuation of the ECS at power operation, operators will be made aware of ECS actuation by means of illumination of incident action indication lights and graphic displays that are located both in the reactor control room and in the remote control center. Also, radiation alarms will advise the operators of increased levels of radiation in the reactor building, the 60,000-gallon tank, and the 500,000-gallon storage tank. Thus, the operators will be given enough information to enable them to take corrective action to terminate the accident in a timely manner, thereby minimizing its potential radiological consequences.

2.5.1.3 Conclusions

Inadvertent operation of the ECS can result from either a spurious signal from the AIA section, or an operator error in manually initiating light water addition. At normal power operation, the reactor system is at a higher pressure than the discharge of the ECS; thus, no ECS water will discharge into the reactor. During refueling operation, the reactor system is at a reduced pressure, and inadvertent actuation of the ECS will result in additional inventory being supplied to the reactor. This excess inventory will be vented from the reactor, driving radioactive moderator in front of it, before damage to the reactor due to overpressurization results. Radiological consequences of this accident are bounded by those for the LOPA accident. Numerous alarms, signals, and status reports are available to the operators to alert them to any inadvertent operation of the ECS and provide for their timely action to minimize the consequences of this accident.

2.5.2 Inadvertent Operation of the Supplementary Safety System

2.5.2.1 Identification of Causes and Accident Description

Each reactor is provided with a SSS, which is an independent system that can be used to shut down the reactors. The SSS can be activated automatically by the safety computers, or by an operator in the control room at either the nuclear console of the SSS panel. One use of the SSS would be to shut down the reactor in the event that a reactor scram is not initiated even though a scram signal is generated.

The SSS actually consists of two identical systems. Each system consists of a tank and three injection paths into the reactor through three sparjets. A total of six injection paths are provided. Each tank contains approximately 32 gallons of gadolinium nitrate solution, commonly called "ink," which is a poison that will shut down the reactor. The tanks are pressurized to approximately 300 psig with nitrogen to provide the driving force for injecting the "ink" into the reactor. The volume of "ink" contained in either of the two independent systems is adequate to shut down the reactor and keep it subcritical.

Each "ink" tank discharge can be routed through any of three parallel valves to the sparjets. The "ink" may be injected by firing either of the two explosive valves on each system, or by relieving pressure on the pneumatic valve on each system. One of the two explosive valves can be fired at the nuclear console by the reactor operator. The same explosive valve can be fired by the safety computer. Actuation of the pneumatic valve is from the SSS panel.

Inadvertent operation of the SSS may be caused by either of the following actions; a spurious signal from the safety computer, or an inadvertent operator action at either the nuclear console or the SSS panel.

2.5.2.2 Analysis of Effects and Consequences

At normal conditions, each of the two "ink" tanks that comprise the SSS is isolated from the reactor system by means of three valves located in parallel flow paths between the reactor and the "ink" tank. At power operation, inadvertent actuation of the SSS would result in the injection of the "ink" into the reactor system, thereby placing the reactor in a shutdown condition. The volume of inventory increase from the SSS is small, being about 32 gallons for one tank, and about 64 gallons if both tanks should discharge. This increase in inventory can be readily accommodated by the reactor overflow system.

During refueling operations, the reactor is already shut down. The addition of the gadolinium nitrate solution only serves to increase the shutdown margin associated with the reactor. Again, the increase in reactor inventory due to the addition of the SSS solution is small, being no more than about 64 gallons for the discharge of both SSS tanks. As was true with the inadvertent discharge of the SSS at power operation, this increase in inventory can be readily accommodated by the reactor overflow system.

The effect of the inadvertent operation of the SSS is to either shut down the reactor, or increase the shutdown margin of the reactor; there are no power excursions associated with this event. Also, the increase in reactor coolant inventory due to this event is small and readily accommodated by normal overflow systems. Thus, there are no radiological consequences associated with the inadvertent actuation of the SSS.

2.5.2.3 Conclusions

The inadvertent actuation of the SSS may be postulated to occur as a result of any of the following: (1) spurious signal from the safety computer, (2) spurious signal from the gang temperature monitor, or (3) operator error. The actuation of the SSS results in either placing the reactor in a shutdown condition if the reactor is at power, or increasing the shutdown margin of the reactor if the accident occurs during refueling. The volume in reactor coolant inventory increase associated with this accident is small, being no more than 64 gallons per reactor, and may be readily accommodated by the normal operation of the reactor overflow system. Thus, there are no radiological consequences associated with this accident.

2.6 Decrease in Process Water System (PWS) Inventory

Several events that would result in a decrease in the inventory of the process water system (reactor coolant system) may be postulated for the Savannah River Site (SRS) reactors. These events are:

- Leaks and failure of small lines
- Process water heat exchanger tube leak
- Loss–of–coolant accidents

Each of these events may result in an inadvertent or unwanted decrease of coolant in the PWS with a subsequent radiological release to the environment.

2.6.1 Leaks and Failure of Small Lines

2.6.1.1 Identification of Causes and Accident Description

Each reactor is provided with a leak detection system that is designed to identify the source of moderator release from the PWS.

Each reactor is also provided with two liquid collection systems designed to collect and contain moderator and, in the event of ECS actuation, light water released from the PWS. The Moderator Recovery System (MRS) is designed to collect and return to the PWS moderator from small to intermediate PWS pipe breaks resulting in releases of 5 to 1000 gpm. Use of the MRS for moderator releases over this range precludes the need to activate the ECS. A description of the MRS is provided in Section 1.2.4.6.

For moderator releases in excess of 1000 gpm, the ECS is activated to assure adequate core cooling is maintained. Moderator and, as the event progresses, light water provided by the ECS that is released through the breech in the PWS is collected by the Contaminated Water Removal and Disposal System (CWRDS).

Seal leaks, flange leaks, and leaks from small cracks in piping are the most likely sources of leakage. Such leakage has occurred in the past. These leaks are a result of the normal operational loading on the reactor(s) such as thermal cycling, corrosion, flow, and neutron embrittlement. Typically, most leakage rates observed in the past have ranged from a few drops per hour to about 0.5 gpm. Over the operational history of all three reactors, only two leaks have approached a rate of 20 gpm.

A postulated break in one of the smaller lines carrying moderator between the PWS and auxiliary or support systems may result in a significant decrease in moderator inventory in the reactor. An example of these smaller lines includes, but is not limited to, the impulse lines extending from the monitor pin at the tank bottom.

2.6.1.2 Analysis of Effects and Consequences

Generally, up to a few gallons of moderator that might be lost from the PWS due to small seal and flange leaks or cracks in piping can be replaced by the process water overflow system and normal reactor operation can be continued. The moderator recovery system (MRS), can replace moderator that might be lost from the PWS due to a somewhat larger release at the rate of between 5 gpm to 1,000 gpm, and can also be used to maintain moderator inventory in the PWS. The reactor is shut down if the MRS is activated.

If activated, it is possible that the MRS might supply light water collected in the sumps to the moderator space of the reactor causing an unanticipated reactivity increase. This event was evaluated for the Mark 16–31B and Mark 22 charges and found to be bounded by the gang rod withdrawal incident. Other charges are evaluated on a case-by-case basis.

If the moderator release is small enough that the release of tritium to the environment through the stack was considered have negligible consequences (less than a few mrem), reactor operation continues. Otherwise, the reactor would be shut down following normal procedures, and the leak found and repaired before returning the reactor to operation.

If the moderator release was large enough to reduce the reactor coolant flow by a measurable amount, the reactor would be shut down to repair the release. If the release was large enough, the reactor would scram before the operators could diagnose the event and take corrective action. The scram would be initiated due to any of the following reactor trip setpoints being exceeded: low moderator level in the reactor, low blanket gas pressure, low plenum pressure, or low assembly flow.

The largest lines carrying moderator between the primary system and an auxiliary system are the lines supplying the control rod coolant ring header. A postulated rupture of one of these lines could cause a 7,200 gpm loss of moderator from the primary system, 4,400 gpm from the higher pressure moderator supply side of the break, and 2,800 gpm from the reactor tank. This event results in the maximum loss for postulated leaks or breeches in small lines carrying moderator to or from the reactor, and is therefore the bounding event for this class of events.

For the analysis of this event, the reactor is assumed to be initially operating normally at maximum power consistent with safety limits for a given charge. The break is postulated to occur in the ring header that supplies coolant to the control rod septifoils. Moderator from the break is released to the floor of the pumproom at the –40 foot elevation. The septifoil coolant monitor would detect the flow of coolant from the pipe break within 1.7 seconds after it is postulated to occur. The reactor would scram on either a low moderator level in the reactor or a low blanket gas pressure. This would occur approximately two minutes before coolant flow through the fuel would be significantly reduced. The release from the higher pressure moderator supply could be isolated within minutes after initiation of the event by closing supply valves in external loops, but release from the reactor tank through the break cannot be prevented.

When the moderator level in the reactor tank drops to 12.8 feet, the AIA initiates incident action by opening ECS isolation valves and starting the ECS booster pump, but does not add light water to the PWS. When the reactor tank moderator level drops to 10.8 feet, the AIA activates ECS injection into three PWS loops just upstream of the water plenum inlet nozzles. ECS flow to a fourth loop can be initiated manually if the flow path to any of the other three loops is inoperable. If the AIA is activated and the reactor tank moderator level drops to five feet, ECS injection to the fourth PWS loop is automatically initiated.

Operating procedures direct the operators to trip the process water pump a.c. motors as soon as possible. This lowers pressure in the system, which reduces flow out of the break and also permits emergency coolant to flow into the reactor. (Emergency coolant supply pressure is not great enough to inject water against the discharge head of the process water pumps when delivering approximately 80 percent or more of normal flow.)

In order to provide adequate cooling, at least one process water pump must remain in operation with cooling water supplied to at least one of the heat exchangers in the same loop. Procedures require that the operators establish flow in three loops. A minimum amount of coolant circulation promotes mixing within the core and reduces the injection flow required for core cooling. This also reduces the rate at which coolant is lost from the break.

If circulation cannot be maintained in at least one loop, operators must align the PWS for once-through cooling. In this mode, the ECS water flows through the core and then directly out of the break. Loss of circulation can be caused by loss of pumping, valve closure, flow blockage, or a loop piping break. Operator action consists of closing certain heat exchanger rotovalves to prevent excessive core bypass flow. Valve closure requirements depend on the status of emergency coolant injection. If full injection flow is provided to two loops, then the operator must close at least eight of the twelve rotovalves. No rotovalves need to be closed if full injection flow is provided to two loops.

Moderator and, as the event progressed, light water supplied by the ECS would be collected in the two –40 foot elevation pumproom sumps, where it is removed by the contaminated water remove and disposal system. Each sump is provided with two sump pumps; a submersible 2500–gpm pump and a 4500–gpm pump that has an elevated drive motor. The AIA will initiate action to align a three way diversion valve to provide flow paths from the sump pumps to storage locations. The sump pumps will automatically actuate on high sump water level to move released moderator and subsequent ECS supplied light water to storage locations outside the reactor building.

Fluid discharged by the 4500-gpm sump pumps will flow first to a closed 60,000-gallon underground storage tank. Piping upstream of this tank is arranged such that, when the tank is full, the discharge from the 4500-gpm sump pumps overflows into piping leading to a closed 500,000-gallon storage tank.

Fluid discharged by the 2500-gpm sump pumps flows directly to the closed 500,000-gallon storage tank. Piping upstream of the 500,000-gallon storage tank is arranged such that, when that tank is full, excess fluid supply from the sump pumps overflows into a 50,000,000-gallon earthen basin.

Almost all of the released moderator would be collected in the two closed tanks. The only vent path from these tanks is back to the reactor building. Thus, any tritium released from the moderator due to evaporation would be discharged only through the 200-foot high reactor building stack. Little evaporation from the closed storage tanks is expected.

As the event continues, ECS coolant will overflow into the 50,000,000-gallon earthen basin. This excess coolant is not expected to contain much radioactivity, as the excess flow bypasses the two closed tanks that receive moderator initially discharged from the reactor. The flow of moderator and coolant from the PWS for this event is similar to that for a loss of coolant accident (LOCA), except that the process occurs much more rapidly for the LOCA than it does for the small line break. Thus the radiological consequences for a postulated LOCA bound those that would result from a postulated break in small pipes carrying moderator between the reactor and support or auxiliary systems.

2.6.1.3 Conclusions

The loss of inventory from the PWS may be postulated to occur from leaks through seals, flanges, and small cracks in piping. Such leaks have occurred in the past. The reactor may continue to operate with some leakage, provided that the leakage can be made up by the reactor overflow system and that radiological release consequences are acceptable. If the reactor must be shut down due either to excessive leakage or to unacceptable resulting radiological consequences, efforts to locate, isolate, and repair the leak would begin immediately. The moderator recovery system would be used to collect released moderator and return it to the PWS for break flows ranging from 5 gpm to 1000 gpm, thus precluding actuation of the ECS.

Larger losses of inventory from the PWS may result from postulated breaks in small lines transporting moderator to and from the reactor and associated support systems. The largest such break that may be postulated would be in a line supplying the septifoil cooling system. Such a break would result in a reactor trip, actuation of the ECS, and a possible overflow of coolant into the 50,000,000-gallon earthen basin. Almost all of the moderator would be collected in two closed tanks outside the reactor building. These tanks are vented only to the reactor building; thus, any tritium release by evaporation would eventually be discharged through the 200-foot high stack. The flow of moderator and coolant discharged from the PWS is similar to that for a LOCA event, but at a slower rate. The radiological consequences of the postulated break of small lines carrying moderator to and from the reactor and its support systems, if bounded by the LOCA event are described in Section 2.6.3.

2.6.2 Process Water Heat Exchanger Tube Leak

2.6.2.1 Identification of Causes and Accident Description

Each of the six recirculating loops of the PWS is provided with two single-pass heat exchangers in parallel. These heat exchangers transfer heat from the moderator flowing through tubes to the light water flowing through the cooling water system. A brief discussion of the process water heat exchangers and the cooling water system is presented here to facilitate understanding of the event.

The process water heat exchangers are located at the -20 foot elevation. Each heat exchanger is about 30 feet long, 7 1/2 feet in diameter, contains about 9,000 stainless steel tubes having a 1/2 inch OD, and has a total heat transfer surface area of about 33,000 square feet.

The source of cooling water for the process water heat exchangers is the Savannah River or PAR Pond. River water is pumped into the area reservoir, then pumped from the reservoir to the reactor building through two large inlet headers. Effluent coolant leaves the reactor building in two large effluent headers and is ducted to the effluent sump. The water overflows a weir in the effluent sump and, for the L and K Reactors, flows back to the river by means of streams and swamps or, for the P Reactor, to a cooling pond. Cooling water effluent from P Area normally flows by gravity back to PAR Pond. Cooling water effluent from L Area returns to the Savannah River by way of L Lake, which allows the effluent to cool before discharge into the river.

The maximum pressure difference across the heat exchanger tubes is about 115 psi. Any leakage, if it occurs, would be from the PWS to the cooling water system. The principal mechanisms for initiating leaks in the process water heat exchangers are corrosion, erosion, vibration fatigue, and cracking of the stainless steel tubing. Although any of the preceding failure mechanisms may result in leakage of process water into the cooling water system through the heat exchangers, no specific failure mechanism is defined. Any leakage is assumed to occur deterministically.

2.6.2.2 Analysis of Effects and Consequences

Under normal operating conditions, the pressure difference between the shell (cooling water) side and tube (process water) side of the process water heat exchanger is a maximum of about 115 psi. The moderator flow carried by any given heat exchanger tube is small, no more than about 1.4 gpm on average. Thus, the addition of moderator to the cooling water flow due to a through–wall crack in a single process water heat exchanger tube is small.

Tritium carried into the cooling water system, as a result of the leakage of moderator through the process water heat exchanger, would be detected by radiation monitors in the cooling water effluent of the heat exchangers. This leakage would also contribute to a decrease in the moderator level in the overflow tank of the reactor overflow system (there may be leakage from the PWS other than through the heat exchangers; the drop in moderator level of overflow tank is a measure of total PWS leakage).

The release of radioactivity to the environment that would result from a postulated leakage of moderator from the PWS to the cooling water system through the process water heat exchanger is small, because the postulated leakage rates are small. The cooling water discharge effluent is monitored by gamma monitors to detect leakage from the PWS. If the radiological consequences of a leak to the cooling water are unacceptable, the reactor will be shut down and the cooling water effluent line isolated, thereby isolating the radiological release to the environment. Thus, the radiological consequences of this event are bounded by those for a moderator spill accident, as described in Section 2.6.3.4.

2.6.2.3 Conclusions

The integrity of the process water heat exchanger may be postulated to degrade due to corrosion, erosion, vibration fatigue, and cracking of stainless steel tubing. Such degradation has, in the past, resulted in small leaks of moderator into the cooling water. If reactor operation is unaffected by the leak, and the radiological consequences of the leak are acceptable, reactor operation can continue. If the radiological consequences of the leak are unacceptable, the reactor will be scrammed, and efforts to locate, repair, or replace the faulty equipment will be initiated immediately. The radiological consequences of this event are bounded by the LOCA event described in Section 2.6.3.

2.6.3 Loss of Coolant Accident (LOCA)

2.6.3.1 Identification of Causes and Accident Description

A LOCA is the result of a pipe rupture in the PWS. For the analysis discussed here, the design basis LOCA is defined as a double-ended guillotine break of a pipe in one of the primary recirculation loops of the reactor. Breaks in small pipes up to six inches in diameter carrying process water are described in Section 2.6.1.

The design basis LOCA is considered a limiting event or fault in that it is not expected to occur during the life of a plant, but is postulated as a conservative design basis. For the reactors at the SRS, the acceptance criterion for the consequences of a postulated LOCA has been defined as:

- The core shall remain amenable to cooling at all times following the postulated piping break.

This acceptance criterion is currently ensured by operating limits placed on the reactors such that, in the unlikely event of a design basis LOCA, no bulk boiling occurs in the core assemblies.

The double-ended pipe break in an inlet plenum line causes the plenum pressure and coolant flow through the core to rapidly decrease. The reduction in core coolant flow reduces heat transfer from the core assemblies to the coolant, causing assembly temperatures to rise. The drop in plenum pressure causes a decrease in the saturation pressure within the assemblies. As the moderator tank level drops, the blanket gas pressure also drops. The loss of blanket gas pressure and the decrease in static head of the moderator in the tank decreases the saturation temperatures at the tank bottom, and other locations within the primary recirculating loops of the PWS and decreases the suction of the Bingham pumps.

A decrease in saturation temperature below local moderator temperatures will allow for the following phenomena to occur within the reactor system:

- Cavitation of the moderator in the primary recirculation loops of the reactor, impeding flow through the intact loops
- Flashing of moderator as it enters the moderator space of the tank, reducing moderator effectiveness
- Boiling of moderator as it flows through core assemblies, resulting in a reduction of both flow through, and heat removed from, the affected assembly

Each of the preceding phenomena, if they occurred, would increase the severity of the design basis LOCA for the reactor. The acceptance criterion for the consequences of a postulated design basis LOCA, however, precludes the occurrence of bulk boiling in the core.

Each reactor has six plenum inlet lines, three of which can supply emergency coolant to the P Reactors; the L and K Reactor has four such lines. The consequences of a postulated LOCA where the break location is in one of the inlet plenum lines (between the heat exchanger and moderator plenum) that also receives flow from the ECS is bounding for all other design basis LOCAs for the following reasons:

- A break in the plenum inlet line causes a large, abrupt drop in plenum pressure, resulting in an immediate drop in flow through core assemblies, lower saturation temperature of the coolant in the core assemblies, and reduced heat removal from the core assemblies.
- A break in the plenum inlet line provides for the maximum possible rate of release of moderator from the reactor, calculated to initially be 66,500 gpm, thereby defining the minimum time after reactor shutdown that the ECS must deliver flow to the reactor to remove fission product decay heat from the core.
- Postulating a break in a plenum inlet line that receives ECS flow diverts that ECS flow from the plenum, thereby reducing the emergency cooling flow to the core.
- The bounding single failure for the design basis LOCA is a failure that precludes operation of tone ECS delivery line, thereby providing minimum emergency cooling flow to the core.

The preceding set of conditions arbitrarily maximizes the coolant needed by the core while limiting the amount available to it.

2.6.3.2 Sequence of Events and Systems Operations

Before the postulated break occurs, the reactor is assumed to be operating normally in an equilibrium condition; that is, process parameters such as temperature, pressure, and flow are stable.

The double-ended pipe break in an inlet plenum line causes the plenum pressures and coolant flow through the core to rapidly decrease. Typical normalized time histories of the flow and pressure decay resulting from this event, generated with the conservative assumption that the postulated break causes the pressure in all six plenum inlets to immediately drop to atmospheric pressure, are given in Figure 2-17. Reactor shutdown would begin about one second after the postulated break occurred. This one-second response time consists of the following components:

• Response time of plenum pressure monitor	0.2 second
• Relay response time	0.1 second
• Time for safety rods to reach mid-core	0.7 second
Total response time reactor shutdown	1.0 second

The reactor power transient following a reactor scram is both charge- and exposure-dependent. A sample of a reactor power transient after a scram signal is generated is given in Figure 2-18 for a Mark 22 charge at 20.5 MWD/ft exposure. This figure is provided for illustrative purposes only; it does not represent a generic response to a safety rod scram for reactor charges. Absolute assembly power limits are established such that no bulk boiling occurs within the assembly during the event.

When the moderator level in the reactor tank drops to 12.8 feet, the AIA initiates incident action by opening ECS isolation valves and starting the ECS booster pump, but does not add light water to the PWS. When the reactor tank moderator level drops to 10.8 feet, the AIA activates ECS injection into three PWS loops just upstream of the water plenum inlet nozzles. For the purposes of analysis, it is assumed that all ECS flow to one PWS loop is lost out the break, and that ECS flow through a second injection path is ineffective due to an arbitrary single failure of an active component in that line. ECS flow to a fourth PWS loop can be manually initiated immediately or, when the reactor tank level drops to 5 feet, the AIA will automatically initiate ECS flow to this fourth PWS loop.

The AIA will automatically initiate action to align dampers and fans of the airborne activity confinement system (AACS) such that a negative pressure will continue to be maintained in the process areas during a reactor incident and that all exhaust air from the process areas will be filtered. This action will provide for removal of a large fraction of particulate and halogen activity that may be released as a consequence of the event.

The process room spray system (PRSS) is actuated manually by procedure. The (PRSS) spray will remove fission products at some characteristic efficiency that may be released to the confinement atmosphere as a consequence of the event.

The longer term flow history in core, normalized to initial flow conditions, that is predicted to result from a postulated design basis LOCA is shown in Figure 2-19. During the first second after the break is postulated to occur, the flow drops to about 70 per cent of this normal full-flow value. That flow rate is maintained for about 12 seconds. During this time, the moderator level in the tank is decreasing due to spillage through the break. This causes the back pressure to the PWS pumps resulting from the static head of moderator in the tank to drop. The plenum pressure that the pump must work against, however, decreases faster than the static head of the moderator does in the tank. Thus, these pumps continue to move liquid at or near their maximum capacity.

As the moderator level in the tank approaches the top 5-foot level, above the reactor effluent nozzles the Bingham pumps driven by the a.c. motors begin to draw air, causing cavitation of the pumps and decreasing pump flow precipitously. Although this occurrence reduces coolant supplied to the plenum by the operation of the PWS pumps in the intact primary circulation loops, the coolant spilled through the break also decreases. Shortly afterward, as flow from the plenum drops in response to reduced delivery of coolant from the pumps, the plenum becomes vented. The plenum venting, which occurs before 16 seconds into the transient, reduces the spilling of coolant from the plenum side of the pipe break. The rate at which coolant is spilled from the reactor through the break of the design basis LOCA after the plenum is vented is calculated to be about 20,000 gpm.

At about 16 seconds into the event, the liquid level in the tank drops to about one foot above the bottom of the reactor tank outlet nozzles. At this time, the liquid level in the reactor tank determines the flow out of these nozzles, not the [illegible] characteristics. The liquid level in the reactor tank, in turn, is determined by the leak rate.

With the ECS activated, an equilibrium state of flow would be achieved in which the rate that coolant spills from the reactor equals the rate that coolant is delivered by the ECS. With five Bingham pumps driven by d.c. motors recirculating water through the intact primary recirculation system, the total flow rate through the core would be greater than the ECS addition rate alone. As an example, for a current Mark 16B–31 charge with 5,000 gpm of ECS flow, the total coolant flow through the core is calculated to be about 22,000 gpm.

The moderator and, as the event progressed, light water supplied by the ECS would flow from the break in the PWS piping and be collected in two pumproom sumps located at the -40 foot elevation of the reactor building. Each sump is provided with two sump pumps; a submersible 2500–gpm pump and a 4500–gpm pump that has an elevated drive motor. The AIA will initiate action to align a three way diversion valve to provide flow paths from the sump pumps to storage locations. The sump pumps will automatically actuate on high sump water level to move released moderator and subsequent ECS supplied light water to storage locations outside the reactor building. These sump pumps, storages tanks, and associated piping, valves, controls, and power supplies comprise the Contaminated Water Removal and Disposal System.

Fluid discharged by the 4500–gpm sump pumps will flow first to a closed 60,000–gallon underground storage tank in Building 106. Piping upstream of this tank is arranged such that, when the tank is full, the discharge from the 4500–gpm sump pumps overflows into piping leading to a closed 500,000–gallon storage tank.

Fluid discharged by the 2500–gpm sump pumps flows directly to the closed 500,000–gallon storage tank. Piping upstream of the 500,000–gallon storage tank is arranged such that, when that tank is full, excess fluid supply from the sump pumps overflows into a 50,000,000–gallon earthen basin.

The minimum flow of coolant through the core occurs when the flow added by the ECS equals the flow lost through the break. When equilibrium between break flow and ECS flow [illegible] achieved, the maximum level of fission product decay power [illegible] the core flow must be capable of removing is determined. [illegible] product decay power decreases as the transient progresses. [illegible] design basis LOCA assumes that this flow condition [illegible] achieved in a minimum time after initiation of the postulated break, thereby maximizing the fission product power that must be removed later from the core. Thus, it is both the time at which equilibrium is achieved between break flow and ECS flow, and the equilibrium core flow rate, that are used as input parameters to set operational limits for the reactors.

Two dominant factors affecting the onset of bulk boiling in the core are assembly flows and power levels. Assembly flows are determined by total core flow and the distribution of that flow to the assemblies in the plenum. Total core flow strongly depends on the rate that the ECS delivers emergency coolant, and only weakly on the flow resistance of the core. The rate that the ECS delivers emergency coolant depends on the number of ECS supply lines to the reactor that maybe assumed to be operable. A summary of the emergency cooling delivery capabilities of the ECS is given in Table 2-11. As stated in Section 2.6.5.1, a minimum emergency cooling flow is provided for by assuming that the postulated design basis LOCA disables one of the ECS addition lines, and that a single failure disables a second delivery line. This leaves a single operable ECS delivery line for the P and K Reactors capable of delivering about 5,000 gpm of emergency coolant to the core. For L Reactor, the preceding assumption provides for two ECS delivery lines to be operable and capable fo delivering about 9,000 gpm of emergency coolant to the core.

The distribution of flow among core assemblies is calculated with a computer code that has been benchmarked against empirical plenum flow distribution data measured in the reactors. A typical flow distribution for the delivery of 5,000 gpm of emergency coolant to a Mark 16B-31 charge following a postulated design basis LOCA is shown in Figure 2-20.

In 1987, a program was initiated to improve the analytical tools used to evaluate the thermal hydraulic transient that would result from a postulated double-ended pipe rupture LOCA. To achieve the objectives of the program, the TRAC computer program was selected to evaluate system behavior resulting from a design basis LOCA, and the FLOWTRAN code was developed to evaluate the resulting thermal transient of the heat generating assemblies in the core, using the system behavior predicted by the TRAC code. Descriptions of the TRAC and FLOWTRAN codes are given in Section 2.1.5. This computational approach has recently been used to set the transient protection limits (TPL) for a new core charge.

Current operating limits imposed on the SRS reactors preclude operation at power levels that would result in bulk boiling in the core at any time during a postulated design basis LOCA, including the time of minimum core flow/maximum fission product decay power.

2.6.3.3 Core and System Performance

Assembly power is defined as the product of total assembly deposited power and power distribution. For a given assembly, the total deposited power starts from the power level the reactor is operating at when the postulated pipe break is initiated. That power level undergoes a near step–like decrease, as a result of the insertion of control rods, to a level that is the result of both short– and long–term fission decay products. The power will continue to decrease until it is only a result of long–term fission decay products.

A sample power history calculated for a Mark 22 charge operating at a reactor power of 1,600 MW and an exposure of 20.5 MWD/ft to a postulated design basis LOCA is shown in Figure 2.6–2. There is a slight power decrease during the first second of the event, prior to the safety rods scramming. The postulated break reduces the pressure driving the flow between the plenum and the core bottom such that the assembly flow is calculated to be about 70 percent of full flow within about 0.2 seconds after the break is initiated. As a result of the core flow reduction, coolant temperatures begin to rise immediately. This causes a slow and slight power decrease as a result of the negative feedback caused by the negative temperature coefficients of reactivity. Insertion of the safety rods at about one second causes an abrupt power decrease.

Within 20 to 30 seconds, the power reduction from the safety rods and the decreasing water level in the tank guarantee that the only source of assembly power remaining is fission product decay heat. The calculation of emitted assembly powers is based on ANSI Standard 5.1 (1979 edition), corrected for gamma energy distribution that occurs in the drained tank environment. The correction for gamma energy is small for Mark 22 charges under drained tank conditions, but can be significant (a multiplication factor of approximately 1.5) for enriched–depleted charges under similar conditions.

To complete the calculation of the consequences of a postulated design basis LOCA on the core assemblies, the temperature of the core assembly as a function of assembly flow and power is required. The basic data required to establish this relationship have been experimentally developed and expanded for use in more general cases by calculations.

The dominant parameters affecting fuel damage calculations for the SRS reactors are assembly power and ECS delivery rate. Thus, the results of an analysis of the consequences of a postulated design basis LOCA may be presented as the degree of core assembly damage as a function of ECS flow. Current operating limits imposed on the SRS reactors provide for no core damage to occur by precluding bulk boiling in the core at any time during a postulated design basis LOCA.

2.6.3.4 Radiological Consequences

The results of the analysis presented in this section demonstrate that the radioactivity released to the environment by a LOCA does not result in doses exceeding the guidelines of 10 CFR 100.

The mathematical models used to calculate the activity releases during the course of the accident and the resultant doses are described in Chapter 3.

2.6.3.4.1 Release of Radioactivity

Tritium is a natural consequence of neutron capture by deuterium. Tritium in the reactor moderator could be partially released to the confinement atmosphere following ECS actuation or any LOCA. As discussed below in Section 2.6.3.4.2, this report uses a conservative value of five-million curies of tritium present in the moderator of each reactor. This value is 40 to 60 percent larger than levels observed during recent reactor operating experience, and is about 20 percent larger than the maximum value ever observed during the operating history of the SRS reactors.

The confinement system has no mechanism for tritium retention. Thus, any tritium released into the confinement system is discharged from the stack. In the current analysis, it is assumed that tritium is released to the airborne activity confinement system and then discharged from the 200-foot stack. This is assumed to occur over a two-hour period. Only a small part of the affected reactor's inventory of tritium would actually be released; the rest would remain in solution in the two (60,000- and 500,000-gallon) storage tanks of the contaminated water removal and disposal system. It is conservatively assumed that about three percent of the tritium becomes airborne.

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 contaminated water removal and disposal system storage tanks.

2.6.3.4.2 Method of Analysis

The Transient Protection Limit (TPL) precludes melting of fuel in any credible LOCA. The main contributor to offsite doses resulting from a postulated LOCA is tritium in the moderator (formed from neutron capture by deutrons). The tritium is released mainly by evaporation. The amount released depends on the tritium concentration, the size of the leak, and the disposition of D_2O leaking from the reactor to the reactor building, the air and water temperatures, and wind speed.

The tritium inventory of a reactor varies with its operating history and is currently about 2 to 3 million curies. To be conservative, a higher value of 5 million curies is assumed for the current analysis.

In the event of any moderator release, almost all of the D_2O would be contained in the two closed tanks of the contaminated water removal and disposal system that are located 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 200–foot stack. Not much of the water in these storage tanks would be expected to evaporate. Some water would evaporate in the process room or below grade area as the water drained to the collection tanks. To be conservative, high water and air temperatures were assumed as well as low humidity and high wind speed. Based on these conditions 3.3% of the tritiated water is assumed to become airborne and pass out the stack (Ref. 3–6).

The design objective of the closed–tank system discussed here is to contain the portion of expelled coolant that is most likely to contain significant quantities of radioactivity (including fission products, if assemblies were to melt). Any coolant flowing to the open-air basin would be the excess after the first 560,000 gallons. The excess flow bypasses the closed tanks, rather than flowing through them. That excess volume should not contain much radioactivity based on the expected accident sequence.

The ECS and the contaminated water removal and disposal system that serves as a liquid activity confinement system are expected to cope with any credible LOCA. However, if the ECS flow exceeds the capacity of the contaminated water removal and disposal system (e.g., a sump pump fails to operate), then ECS injection line valves are throttled to balance the sump pump capacities with the ECS flow and reactor core cooling requirements. Throttling is done in increments according to written procedures that require continuous monitoring of sump levels and reactor liquid discharge temperatures.

2.6.3.4.3 Dose from Moderator Release

In siting a reactor (10 CFR100), regulatory guidelines specify reference values for the maximum dose an individual might receive at the outer boundaries of the plant and at a low population zone. The reference dose values are 25 rem to the whole body and 300 rem to the thyroid. An exposed individual is assumed to be at the plant boundary for two hours immediately following the postulated release of fission products. This represents the same exposure to an individual as being at the outer boundary of the low population zone for the entire passage of the radioactive cloud.

The present analysis does not take credit for emergency plans to evacuate personnel beyond the plant boundary. The plant boundary and the low population zone for the SRS are assumed to be identical for this analysis. Thus, for this safety analysis report all offsite doses are computed at the SRS boundary.

The maximum dose calculation uses specified meteorology that also has a very low probability of occurrence. At the SRS site, the very unlikely meteorology determining the maximum two-hour.

whole-body dose occurs less than 1.1 percent of the time (not exceeded 99.5% of the time). This and other parts of the calculation are discussed in Chapter 3.

As discussed in Section 2.6.3.4.2, 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 3.3 percent of the tritium inventory (five million curies assumed) in the moderator over a two-hour period. (The more realistic value of three million curies tritium inventory reduces the computed release by 40%.) Most of the tritium would probably remain in solution in the two holding tanks (60,000- and 500,000-gallon) for the expected accident sequence. The calculated maximum dose to an individual at the plant boundary is shown in Table 2-12 for typical and very unlikely conditions. Table 2-13 shows the calculated dose to an individual onsite at various distances from the accident.

2.6.3.5 Conclusions

The design basis LOCA is the double-ended guillotine break of a line in the primary recirculating loop of the PWS. The design basis LOCA is an event that is not expected to occur during the life of the plant, but is postulated to establish a conservative design basis. The acceptance criterion for the design basis LOCA is the maintenance of a coolable core geometry. This acceptance criterion is currently satisfied by imposing operational limits on the reactor that preclude bulk boiling in the core, should a design basis LOCA occur. The ability of the ECS to supply sufficient emergency coolant to support the implementation of the one bulk boiling requirement is demonstrated by calculation for each charge design. The subsequent radiological consequences of a design basis LOCA are demonstrated, also by calculation for each charge design, to be within the limits prescribed by 10 CFR 100.

Table 2-11

Emergency Cooling System
Sample Flow Delivery Comparison Summary

Number of Systems[a]	Source[b]	Reactor	Flow, gpm: Loss of Coolant[c]	Flow, gpm: Loss of Pumping
1	H_2O header	P	4,850	5,750
2	H_2O header	P	7,860	9,900
3	H_2O header	P	——	11,800
1	Booster pump	P	5,120	6,630
2	Booster pump	P	8,040[d]	10,710
3	Booster pump	P	——	12,390
1	H_2O header	L, K	4,840	5,750
2	H_2O header	L, K	8.030	10,200
3	H_2O header	L, K	——	12,300
1	Booster pump	L, K	5,340	6,680
2	Booster pump	L, K	8,520	11,280
3	Booster pump	L, K	——	13,200

(a) The ECS is capable of adding H_2O through four of the plenum inlet lines.

(b) The ECS has three independent (primary) sources of H_2O available: the two H_2O supply headers and the booster pump. The booster pump is capable of supplying somewhat higher flows to the ECS than obtained with either H_2O header.

(c) The single failure criterion, as applied to the ECS, says that if one of the ECS systems is assumed to fail (column 1), then the booster pump may be assumed to operate. If the booster pump is assumed to fail, then all of the ECS systems may be assumed to operate (with the exception of the leaking system for loss-of-coolant accidents). The lowest flow for either of these two cases is then assumed for the accident.

(d) Minimum flows with fourth system (calculated).

Table 2-12

Calculated Radiation Dose to a Person at the Site Boundary from a Process Water System Moderator Release

Accident	Operating and Meteorological Condition (a)	Effective Dose Equivalent (2-HR) (rem)
Reference values for reactor siting in 10 CFR 100		25
D_2O release from process water system	Typical Very unlikely	0.002 0.027

(a) Typical conditions are 2,500 MW reactor power and average (50 percent) meteorology. Very unlikely conditions are maximum anticipated reactor power of 3,000 MW and very unfavorable meteorology as specified in NRC Regulatory Guide 1.145. Values shown are maximum for any of P, L, and K Reactors.

Table 2–13

Maximum Dose (REM) for Individuals Onsite at Various Distances[a]

Distance From Reactor (Miles)	Moderator Spill Accident (No Core Damage)[b]	
	Very Unlikely	Typical
0.5	0.047	0.015
1.0	0.049	0.014
2.0	0.058	0.011
4.0	0.043	0.005
6.0	0.032	0.003
Site Boundary	0.027	0.002

(a) This table represents the maximum effective dose equivalent at locations spanning the site directly under the plume. Very unlikely meteorological conditions are meteorological conditions not to be exceeded 99.5 percent of the time. Typical conditions are not exceeded 50% of the time.

(b) Doses are the maximum of all 16 directions for a two hour exposure for the reactor with the largest dose.

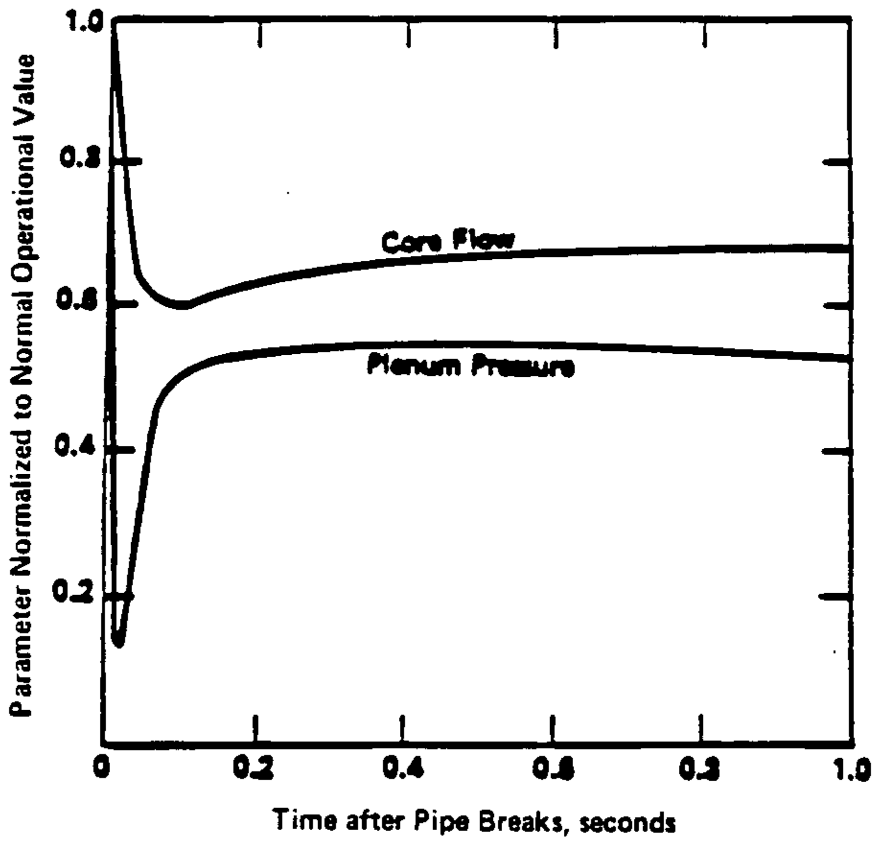

Figure 2–17

Sample Representation of Flow and Pressure Transients During First Second of Maximum Rate Loca

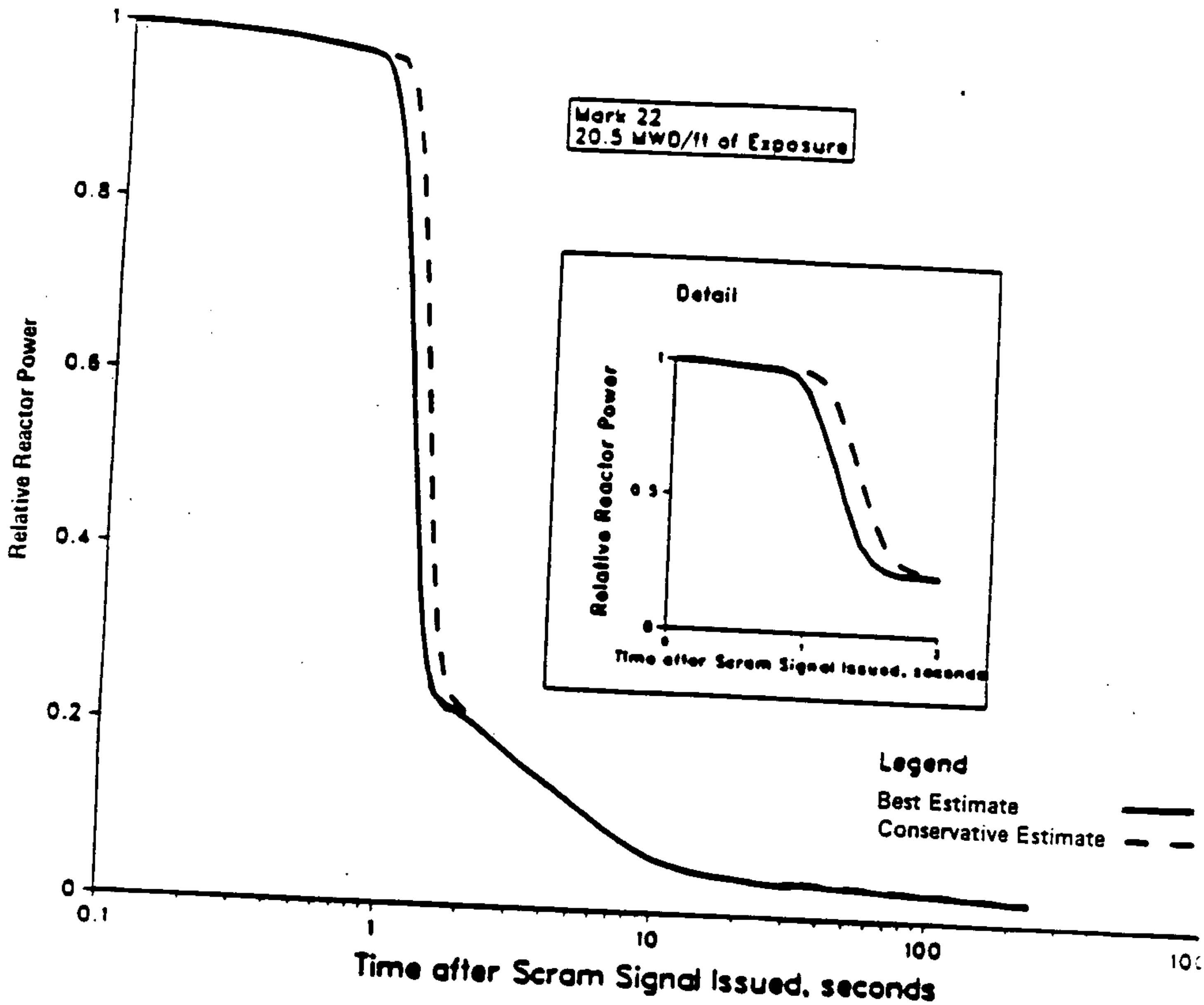

Figure 2–18

Sample Representation of
Reactor Power Transient for Safety Rod Scram

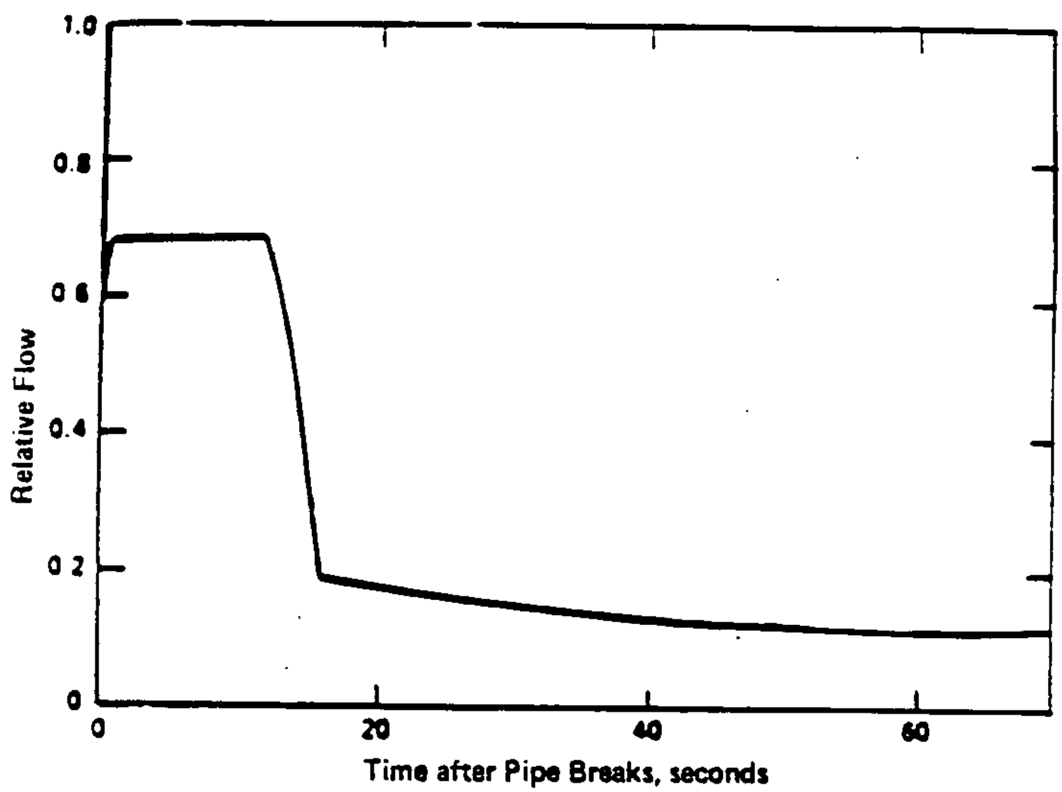

Figure 2-19

Sample Representation of
Flow Transient for Maximum-Rate Loss of Coolant

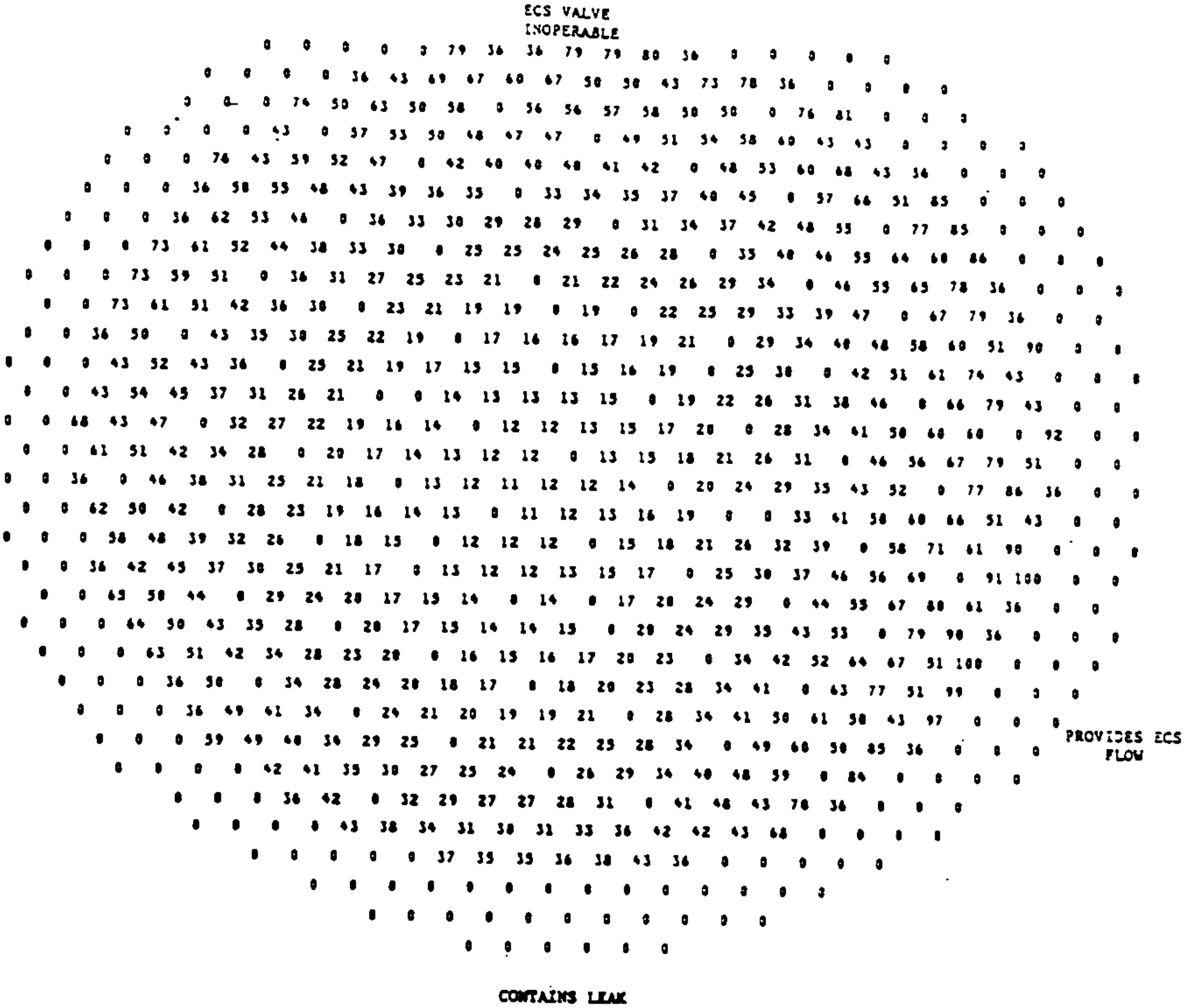

Figure 2–20

Sample Representation of ECS Flow Distribution in Assemblies for Loss of Coolant Accident (Total ECS Flow is 5,000 GPM)

2.7 Radioactive Releases from a Subsystem or Component

2.7.1 Identification of Causes and Accident Description

The fuel handling accident evaluated is the accidental dropping of an irradiated assembly onto the process room floor due to a failure of the latches on the discharge machine. This is considered to be the limiting accident in this category. The assembly is assumed to have sufficient decay heat and fall to a position such that the process room spray system cannot adequately cool the assembly. The assembly is assumed to melt and release noble gases, iodine and tritium to the airborne activity confinement system and out the stack.

The first step in fuel handling is shutdown and cooldown of the reactor. After reactor shutdown, irradiated fuel and target assemblies continue to generate heat as a result of activation and fission product decay. To prevent overheating during transport from the reactor to the water-filled canal, the assemblies while in the discharge machine are cooled with a continuous flow of water. This water is supplied via one or both of the two independent cooling systems on the discharge machine.

Before a heat generating assembly can be discharged from the reactor, the heat generation rate must be low enough so that the assembly will be adequately cooled if it is dropped horizontally underwater in the disassembly basin. The heat generation rate must also be low enough so that the discharge machine cooling system is adequate to cool the assembly; an assembly is adequately cooled if water exits the assembly at less than 100°C.

The operation is performed with charge and discharge machines, which are operated remotely from the crane control room because of the radiation levels in the reactor room. The discharging operation consists of taking an assembly from the reactor and placing it in a conveyor for transfer to the disassembly basin.

The fuel handling accident is assumed to occur after a fuel assembly has been removed from the core, but before it has been placed in its designated location in the disassembly basin.

The process room spray system would be manually activated by the operators to cool a hot assembly in discharge mishaps and prevent melting if the discharge machine cooling systems fail or the assembly were dropped.

The fuel handling accident in the reactor confinement area does not impact the integrity of the core.

Any fission product release from the fuel handling accident would be in the process room. The air from the process room is exhausted through a set of confinement filters before release to the stack.

2.7.2 Analysis of Effects and Consequences

One irradiated fuel or target assembly is assumed to release noble gases, iodine and tritium. The fission products in the assembly would have decayed between shutdown of the reactor and the discharge operation.

Because assembly discharge operations do not begin before at least 14 hours after shutdown, fission products would have experienced at least 14 hours decay from their equilibrium values. This accident could release no more than 0.0246 percent of the core inventory at the time of shutdown because of the decay of fission products.

The reactor room emergency spray system would be used to cool a hot assembly in most discharge mishaps and 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 and tritium 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.

The calculated maximum dose from this accident to an individual at the plant boundary is shown in Table 2-14 for typical and very unlikely conditions.

Onsite doses and the major assumptions and parameters assumed in the analysis are itemized in Chapter 3.

2.7.3 Conclusions

The limiting accident in this class of accidents for the SRS reactors is considered to be the accidental dropping of an irradiated assembly during refueling. The irradiated fuel assembly is assumed to fall into a position such that it is not adequately cooled by the emergency spray system. The calculated maximum dose to an individual at the site boundary resulting from the assumed accident is a very small fraction of the reference doses used by the NRC.

Table 2-14

Calculated Radiation Dose to a Person at the SRS Boundary Following a Discharge Mishap[a]

Operating and Meteorological Conditions[b]	Calculated Dose, Rem: Whole-Body (2-hr)	Calculated Dose, Rem: Thyroid (2-hr)
Reference values[c]	6	75
Typical	0.0042	0.0032
Very unlikely	0.048	0.033

(a) One fuel assembly melted.

(b) Typical conditions are 2,500 MW reactor power, average (50 percent) meteorology, and 19-month service age carbon filters (carbon filter age is discussed in Section 2.3.2.2). Very unlikely conditions are maximum anticipated reactor power of 3,000 MW, very unfavorable meteorology as specified in RG 1.145 (95 percent site, 99.5 percent worst sector), and 19-month aged carbon filter. Values shown are maximums for any of the P, L, and K reactors.

(c) U.S. NRC Standard Review Plan, NUREG-0800, Section 15.7.4, Radiological Consequences of Fuel Handling Accidents, July 1981.

2.8 Anticipated Transients Without Scram (ATWS)

The effects of anticipated transients without scram (ATWS) are not considered in the design basis of modern licensed commercial nuclear power plants since the likelihood of such hypothetical events is negligibly small. Even though ATWS are beyond the design basis, these events are addressed from a safety perspective in more recent SARs because of interim licensing requirements established in NUREG–0800 and RG 1.70 relative to SAR Chapter 15 accident information. These requirements include:

- The reactor protection system is to be designed with appropriate margin to ensure that acceptable fuel design limits are not exceeded during normal operational, including anticipated transients.
- The reactor protection system is to be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operations, including anticipated transients.
- Two independent reactivity control systems are to be used.
- The reactivity control systems are to have the combined capability of reliably controlling reactivity changes to ensure that under postulated accident conditions, and with appropriate margin for stuck rods, the core can be cooled.
- The protection and reactivity control systems shall be designed to ensure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

A probabilistic risk assessment is being performed for the SRS reactors to determine the relative importance of all postulated accident scenarios, including ATWS. This work is scheduled to be completed in the next fiscal year.

Even though the ATWS type events are expected to have extremely low probabilities, the SRS reactors have the supplementary safety system (SSS) to protect them against postulated failure during scram conditions. This has been accomplished through the application of the confinement protection limits (CPL) described in Section 2.3. The intent of these limits is to maintain the integrity of the airborne activity confinement system during specific postulated transients not terminated by safety rod action. To protect the confinement system, the criterion of preventing a major breach in the reactor tank and/or primary boundary was adopted. This is achieved by limiting the rate of steam formation in that the pressure forces that develop are less than those required to lift the plenum and break the roll anchors. This limit is implemented by requiring the automatic backup shutdown system to activate the SSS in time to prevent attainment of this pressure level. When actuated, the SSS injects a solution of gadolinium nitrate into the reactor moderator through six spargers near the center of the reactor. Moderator circulation and diffusion distribute the nuclear poison through the reactor core.

Anticipated transients are those of Condition II: Incidents of Moderate Frequency, described in Section 2.2. The Chapter 2 events evaluated relative to the CPL include:

- Loss of power to cooling water pumps (Section 2.2.1)
- Loss of cooling water inventory (Section 2.2.4)
- Loss of process water a.c. pumping power (Section 2.3.1)
- Combined loss of both process water and cooling water a.c. pumping power (Section 2.3.2)
- Closure of rotovalves in process water loops (Section 2.3.3)
- Loss of blanket gas pressure (Section 2.3.5)
- Gang rod withdrawal (Section 2.4.2)

Each of these accidents has been evaluated and the resultant transient performance was found to be within that allowable CPL.

CHAPTER 3.0
ONSITE DOSES FROM DESIGN BASIS ACCIDENTS

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3.0 ONSITE DOSES FROM DESIGN BASIS ACCIDENTS

3.1 Introduction

This chapter presents the calculational methods used for estimating onsite doses resulting from design basis accidents. The results are summarized in Tables 3-1 through 3-10. Site-specific meteorology, topography, and release elevation were used in the calculations.

3.2 Summary

Table 3-1 summarizes potential dose to personnel during post-accident evacuation from a reactor area. If personnel follow the evacuation routes, the calculated dose is less than 1 rem for the worst assumed accident. In the unlikely event that released activity passes over the evacuation route, the dose could be as high as 44 rem. Tables 3-1 through 3-10 summarize the maximum calculated effective dose equivalent, thyroid, and bone doses on the site at increasing distance from the reactor area. As shown in these tables, the maximum two hour whole body dose is less than 6 rem; the maximum thyroid dose is less than 4 rem at the SRS boundary.

Within the reactor building, doses from radioisotopes in the reactor process room, in the ventilation filter banks, and in the plume exiting from the stack would be negligible. With certain very unlikely meteorology, some of the radioisotopes released from the reactor building stack could infiltrate personnel areas through the ventilation system. In such an instance, portable breathing equipment would be necessary to limit the doses for essential personnel not evacuated from the building. Procedures require evacuation of all personnel before their exposure reaches 25 rem. After evacuation, control of the reactor would be from the remote control room.

Outside the reactor areas, except for minor perturbations caused by topography, effects decrease as distance from the accident site increases.

Doses were calculated using methodology described in Ref. 3-2. Meteorological data was collected for a five-year period 1982-86 (Ref. 3-3), and dispersion calculations conformed to criteria required by the NRC (Ref. 3-4). Doses were not calculated for releases from the discharge areas of the 105 Buildings or from the 50-million-gallon basins because such releases are considered improbable under existing operating requirements.

3.3 Reactor Accidents that Result in Doses to Personnel

The radioactive material can be released after a reactor accident and can be dispersed over the SRS, exposing personnel to radiation. The personnel dose was calculated at various locations for exposure to radiation from noble gases, iodine, and particulates.

This report examines three types of reactor accidents which represent the spectrum of credible design basis accidents. In order of decreasing severity of consequences they are:

- Misloading accident. The postulated worst-case misloading of a fuel assembly during reactor refueling could cause an inadvertent criticality and release up to 3 percent of the reactor core fission products.
- Discharging accident. A single irradiated fuel assembly, could release fission products as a result of a mishap during discharging.
- Moderator spill accident. A large process water (D_2O) leak or activation of the emergency cooling system could spill the entire moderator and release tritium.

A three percent release of fission products from the misloading accident bounds the consequences of the other two accidents. The moderator spill accident differs from fuel damage accidents in that the moderator tritium is the only significant radioactive species released. Therefore, two bounding accidents will be discussed in this chapter.

3.3.1 Releasing Fission Products from Three Percent of the Core Inventory

Fuel damage accidents in the core could result in the release of fission products and other radioactive isotopes into the process room. Experiments show the sequence of release of fission products can occur in as little as 15 minutes (Ref. 3-5). The fission products that contribute the most doses to personnel are noble gases and iodine. Tritium and particulates are also released.

Fifty percent of the iodine in the affected core material is assumed to become airborne. Only 0.05 percent of the airborne iodine bypasses the filters, and the remainder of the iodine is trapped by the carbon filters. The iodine on the filters is eventually released from the exhaust stack of the reactor building as it slowly desorbs.

Several fissionable and non-fissionable isotopes that become particulates after melting could be released to the process room. The dominant particulate isotope, Pu-239, is the only particulate isotope considered in this chapter. Other particulates present fewer risks because they are produced in lesser quantities or have less effect on the body.

3.3.2 Moderator Spill Accident

Moderator spills are less severe to personnel than core accidents because they release less radioactivity. The total moderator inventory in the reactor contains less than 5 MCi of tritium. The moderator, diluted by emergency cooling system flow, would collect in the below-grade sumps and drain to a 500,000-gallon storage tank. Some of the tritium would evaporate in the 105 Building and be released from the stack.

The amount of tritium released during a moderator spill depends on the temperatures of the moderator and air, the emergency cooling system flow rate, wind velocity, and humidity. A previous study (Ref 3-6) examined the amount of tritium that evaporates for various conditions and concluded that 3.3 percent of the tritium in the moderator could be released through evaporation within a 2-hour period under the worst expected conditions.

3.4 Dose to Personnel in the Reactor Area

The radioactive isotopes released from the stack form a plume that disperses across the plant site depending on the wind speed, direction, and variability. The frequency of particular wind speeds and directions was collected at weather towers near the reactors and averaged to indicate the probability of conditions at an averaging time of 60 minutes or two hours. A 5-year record was used. (Ref. 3-3).

Using the wind data and a Gaussian plume assumption, as recommended in NRC RG 1.145 (Ref. 3-4), relative concentrations across the plant site were calculated for meteorological conditions not exceeded 99.5 and 50 percent of the time. The resulting doses are indicated in the Tables 3-2 through 3-7.

Doses in the reactor areas are potentially the largest of any on the SRS. Doses are minimized by personnel assembling in the reactor building or evacuating the area, as current procedures require (Ref. 3-7). Doses were calculated for both situations (Ref. 3-8).

Emergency procedures require personnel in the reactor building to stay inside, and for personnel within the inner fence to seek shelter in the reactor building. Other personnel in a reactor area are to evacuate along one of two designated routes to a location outside the reactor area for evacuation by vehicle. A public address announcement from the 105 Building control room will designate which evacuation route is to be used (Ref. 3-7).

Selection of a route is based on local wind direction as indicated by instruments in the control room. Personnel exit the reactor area through the nearest crash gate away from the direction of the plume.

3.4.1 Dose During Evacuation

The irradiation of personnel during a planned evacuation away from the plume was calculated. The irradiation of personnel was also calculated for an evacuation under the plume, which is the worst case. Table 3-1 indicates the effective dose equivalent for each case.

When personnel evacuate away from the plume, exposure generally decreases as they move farther away. For this situation, the dose cannot be calculated easily, because the available codes only calculate the dose immediately below the plume. An upper limit calculation was made that indicates the maximum dose personnel could receive during the evacuation. The upper bounding dose assumed that the release was at height equal to the distance from the plume centerline to the closest evacuation path. This upper bounding dose is considerably higher than that expected for planned evacuation because during evacuation the plume should be much farther from personnel moving away from the reactor area. The upper bounding dose is reported in Table 3-1 as "evacuation effective."

The dose for the worst-case condition, evacuation under the plume, was calculated by assuming the exposure occurred 0.5 miles from the reactor for a two hour period. The dose at 0.5 miles represents the maximum dose that might occur to personnel evacuating along a route from the reactor to the pick-up point. The dose for a two-hour exposure for a plume overhead is given in Tables 3-2 through 3-4 (whole-body) and Tables 3-5 through 3-7 (thyroid).

3.4.2 Dose in the Reactor Building

The dose inside the reactor building is insignificant in the lunchroom and the control room for radiation from either inside the process room or from the filter compartments. In all cases, several feet of concrete shielding limit the dose rate to less than 0.5 mrem/hr (Ref. 3-8).

Radiation in the reactor building affects personnel only if the air intake system brings radioactivity into the building personnel areas. When the wind is blowing from the stack toward the actuator tower, the relative concentration is high on the downwind side of the tower where the building air intakes are located. Emergency procedures require stopping air inflow to personnel areas following an accident.

3.5 Dose to Personnel on the SRS

The release of radioactivity from the stack of the reactor building would allow a plume of radioactivity to disperse across the SRS. For a three percent core release accident, the whole-body, thyroid, and bone doses were calculated for a two hour exposure period. For a moderator spill accident, the whole-body dose for very unlikely meteorology was calculated for a two hour exposure. The doses were calculated for concentric rings centered on each of the three reactors at distances of 0.5, 1.0, 1.5, 2, 4, 6, 8, and 10 miles and at the SRS boundary. Calculations were made at each distance for sixteen 22.5° sectors around the site perimeter; the maximum value of the 16 directions is reported.

3.5.1 Doses from the Three Percent Core Release Accident

The whole-body dose was calculated at the nine concentric rings in 16 directions for a two-hour exposure. Tables 3-2 through 3-4 indicate the maximum whole-body dose for each reactor for very unlikely and typical conditions at each of the nine distances. For distances greater than three miles, the dose is less than the 25-rem reactor siting criteria (Ref. 3-

The thyroid doses were calculated at the same locations as the whole-body doses and for a two-hour exposure. All thyroid doses are less than 10 rem; this is considerably below the 300 rem NRC siting criteria. Tables 3-5 through 3-7 list the maximum thyroid doses in very unlikely meteorological conditions at each concentric ring for a 60-minute exposure period.

The inhalation of particulate Pu-239 released after an accident would produce a bone dose. The relative concentrations in very unlikely and typical conditions at concentric rings centered on each reactor were used to calculate the bone doses (Ref. 3-8). Tables 3-8 through 3-10 list the maximum bone doses at various distances for a two hour exposure period. There is no NRC limit for exposure to bone, but all exposures were less than 300 mrem for typical conditions.

3.5.2 Dose from a Moderator Spill Accident

The inhalation of tritium released after a moderator spill accident produces a whole-body dose. The relative concentrations in very unlikely and typical conditions at concentric rings centered on each reactor were used to calculate the whole-body doses from tritium (Ref. 3-8). Tables 3-8 through 3-10 list typical doses at various distances for a 60-minute exposure. The-25 rem whole-body dose siting criterion was not approached. All exposures were less than 60 mrem.

3.6 Calculational Method

This section describes detailed assumptions, methods, and parameters used in analyzing the radiological consequences of postulated accidents. The method is described in more detail in Ref. 3-2. Table 3-11 lists the equilibrium iodine and noble gas inventory of the core at 3,000 MW. The site-specific, short-term atmospheric dispersion factors X/Q are based on NRC RG 1.145 methodology (Ref. 3-4) and represent the 0.5 percent worst-sector meteorology; these factors are given in Tables 3-2 through 3-10, and 3-12. Release fractions for iodine loaded on confinement system carbon beds are presented in Table 3-13. Inhalation dose factors for tritium and radioiodines are given in Table 3-14.

The maximum offsite doses are based on:

(1) Specifying the source of radioactivity. Specifying the source designates the release rate and isotope type.

(2) Computing the transport of the isotope by the wind. The calculational procedure and pertinent meteorological data determine the transport of radioactivity.

(3) Computing the amount of radioactivity absorbed by an individual at the plant boundary. This calculation requires specifying a standard man, breathing rates, and several parameters related to absorption of energy from a particular isotope.

These calculations do not indicate the precise dose an individual would receive if he were at the exposure point following a reactor accident. The result does provide some indication of an upper limit on the probable dose to an individual. Individual body characteristics, time of exposure, and wind behavior are important variables that are generalized in computing a maximum individual dose at the boundary. In an actual accident, the WIND computer system of SRS would predict the release path and indicate appropriate action to minimize exposure to people offsite (Ref. 3-9). Evacuation procedures would reduce the actual dose to an individual (Ref. 3-7).

3.6.1 Sources of Radioactive Release

The three sources of radioactivity considered are tritium in the D_2O moderator, fission products in the fuel, and tritium in the targets of tritium-producing charges. The two types of accidents considered are moderator spills and assembly melting accidents. For the moderator spill accident, the maximum amount of radioactivity available for release is conservatively assumed to be 5 MCi of tritium (the actual value is probably less: 2 to 4 MCi). About three percent of this would evaporate in the first two hours. For the melted assembly accidents, the damage to the core is specified as the fraction of the fission product inventory that is available for release (e.g., for a three percent core damage accident, three percent of the core inventory of radioactivity is assumed to be available for release). All releases are via the 200-foot stack in the confinement system of each reactor.

Any tritium or noble gas activity released into the confinement system is discharged from the stack because the confinement system has no mechanism for removal of these activities.

If any fuel or target assemblies melt, fission products become available for release. Depending on the type of assemblies melting and the accident circumstances, the radioactivity released would include noble gases (xenon, krypton), iodine, tritium, and radioactive particulates (fission products, Co-60, Pu-239, etc.). The concentration of these isotopes (Table 3-11) in the core is a function of reactor power, which is typically 2,500 MW, but has approached 3,000 MW. The total inventory of noble gases at shutdown is about 300 MCi; the total inventory for iodine is about 1,000 MCi.

The total inventory of all fission products at shutdown is a little more than seven times the total of iodine and noble gases combined. Some of these isotopes decay rapidly following shutdown and, depending on the expected accident sequence, some isotopes may not contribute significantly to potential doses. The full power iodine and noble gas inventories (major contributors to dose) are listed in Table 3-11. Tritium is present in the lithium-containing assemblies and control rods, as much as 70 MCi of tritium may be present in some charges. The radioactive particulates include several different isotopes that would be captured by the HEPA filters. The amount of particulates that would penetrate the filters would not contribute significantly to the offsite dose. Two releases of radioactivity from the core have been considered: release of fission products from a single fuel assembly and three percent melting of the core during a misloading accident (Ref. 3-2).

For the release from a single assembly during discharge, 14 hours of decay of fission products is assumed as the minimum time to satisfy other discharge constraints.

Prior to the misloading accident involving three percent core damage, the fission products would also have decayed for a minimum of 14 hours. However, more fission products would be formed during this postulated criticality accident. To be conservative, no credit is taken for decay prior to the accident.

3.6.2 Transport of Activity Released

To standardize calculations as much as possible, the NRC has suggested an analysis technique for a release that occurs during a short period (Ref. 3-4). The release from the stack is assumed to propagate as a Gaussian plume and the irradiation of an individual is treated as a time-integrated calculation.

Noble gases and tritium are assumed to be released to the atmosphere very rapidly (Ref. 3-5). Unfavorable wind conditions are also assumed. These stable, slow (1 to 4 m/sec) winds would transport the atmospheric releases to the plant boundary in about one hour. Higher wind speeds would have the material to the plant boundary sooner but would cause more dispersion and result in lower offsite doses.

The methodology assumes a continuous plume and does not account for a traveling wave front. The irradiation period begins when the radioactive material is released. Both the noble gas and iodine source terms are assumed to have decayed during transport. Decay during the exposure is not included in the calculation.

The downwind concentrations of iodine, tritium, and noble gases are calculated according to an integral technique using the computer code NRC-145-2 (Ref. 3-10). This code was developed at the SRS and uses a Gaussian plume model based on RG 1.145.

The meteorological data used in the dose calculations were collected from 1982 through 1986 in accordance with the NRC Safety Guide 23 (Ref. 3-11). The data were obtained at towers near P and K Reactors. Calculations for L Reactor used data from the closest tower (K Area). The meteorological data from each tower are averaged for two-hour periods and sorted into 16 sectors, 6 wind speeds, and 7 stability classes. (Stability classes are based on the standard deviation of the mean wind azimuth.)

Doses are computed by two methods. The first method computes, for the entire site (all 16 sectors), a dose (either inhalation or whole-body) that would be exceeded only five percent of the time. The result is referred to as the 95th percentile value. The second method computes for each sector a dose value that would be exceeded only 0.5 percent of the time (99.5th percentile procedure). The maximum dose for all sectors is then compared to the 95th percentile dose for the whole site, and the higher of the two values is reported.

The doses reported in Tables 3-2 through 3-10 for conditions not exceeded 50% of the time use the first method mentioned above and are based on doses calculated for the entire site. The doses in those tables not exceeded 99.5% of the time use the second method and are for a particular sector.

For the SRS, the second method (99.5th percentile worst sector) calculates doses (both thyroid and whole body) at the site boundary that are about a factor of two higher than the value obtained with the first method (95th percentile whole site). The 99.5th percentile worst sector doses approximate whole site percentiles of 98.0 to 99.4 (depending on the reactor and meteorological tower). The fact that these equivalent whole-site percentiles are near 99.5 (the worst sector percentile) means that the meteorological data are very asymmetric for the stability classes and wind speeds that produced the highest doses.

Corrections for topography and jet rise of the released plume are applied. The topography correction is prescribed by RG 1.145 (Ref. 3-4) 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 NRC computer code 145-2 is described in Ref. 3-12.

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

The relative downwind concentration factors for each reactor are given in Table 3-12. These X/Q values are obtained by dividing the downwind concentration of an isotope in Ci/m^3 by the release rate at the stack (Ci/sec) without regard to subsequent dose effect.

The effective dose equivalent is composed of an inhalation component from iodine and tritium and a skyshine component from the gamma emission of the noble gases. The inhalation component is computed by multiplying the relative concentration by (X/Q) the source strength breathing rate, and dose conversion factor. The shine component integrates the gamma dose from the entire (finite) radioactive cloud. The thyroid dose is mainly due to inhalation of iodine. The effective dose equivalent for the entire body was not added to the thyroid dose in this chapter.

The source term for iodine is the amount that would desorb in the first two hours following the incident. The average iodine retention efficiency assumed for the carbon is that for carbon aged 19 months. This is 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. Iodine release as a function of carbon service age is given in Table 3–13 (Ref. 3–13).

3.6.3 Dose Conversion

The radioactive release is transported to the plant boundary based on the above techniques. At the boundary, the release is assumed to irradiate a standard man to determine the maximum dose received. For iodine and tritium, a reference man and breathing rate are used to calculate an inhalation dose. The dose conversion factor considers skin absorption as well as inhalation in the case of tritium.

Inhalation dose factors for tritium and radioiodines are given in Table 3–14. These factors are from Ref. 3–15.

3.6.4 Dose Calculations

The methods used for offsite dose determination are summarized below (Ref. 3–2).

3.6.4.1 Calculation of Dose from Tritium

The dose due to tritium includes whole body contributions from inhalation and skin absorption. The release time and the exposure time are assumed to be equal.

The equation for dose equivalent is:

$$\text{Dose equivalent} = \text{dose factor}\left(\frac{\text{rem}}{\text{Ci}}\right) \times \text{source (Ci)}$$

$$\times \text{ breathing rate } \left(\frac{\text{m}^3}{\text{sec}}\right) \times \text{dilution factor, } \frac{\chi}{Q}\left(\frac{\text{sec}}{\text{m}^3}\right)$$

The dilution factor is the calculated downwind dilution based on local meteorology. The breathing rate is 3.47×10^{-4} m^3/sec as specified by NRC guidelines. The dose factor is based on information presented in ICRP-30.

3.6.4.2 Calculation of Dose from Noble Gases

The model used for the noble gas contribution to dose is explained in Appendix B of Ref. 3-10. It is a subroutine called ERGAM in the computer code NRC-145-2. The method is based on irradiation of a subject on the ground by a source of radioactivity passing overhead. The dose to the individual from the finite source is integrated to find the total dose. Build-up factors contribute to the air absorption of radiation; they are based on the assumption of 1-MeV gamma rays.

ERGAM uses the same meteorology and terrain values as the inhalation dose calculation. A finite cloud meteorological model is used. The source released is based on the percentage of the core inventory that melts.

3.6.4.3 Calculation of Dose from Iodine

The inhaled and skin absorbed iodine isotopes migrate to the thyroid and are the contributors to the thyroid dose. The iodine is trapped by the carbon filters in the airborne activity confinement system. The iodine desorption values are explained below.

Fifty percent of the inventory of melted fuel is assumed to become airborne. This amount is twice that specified by the NRC; this is because of the high flow rate in the confinement system preventing deposition (or plate-out) of the iodine. The offsite dose is:

$$D = N \times S\,(Ci) \times \frac{X}{Q}\left(\frac{sec}{m^3}\right) \cdot B\left(\frac{M^3}{sec}\right)$$

where:

N = the dose conversion factor (rem/Ci) from ICRP-30

S = the curies of iodine isotopes released from the stack

B = breathing rate $\left(\frac{m^3}{sec}\right)$

After the iodine is released to the process room from assembly overheating, it travels in the confinement system until it reaches the carbon filters. Only 0.05 percent of the iodine bypasses the filter; the rest is trapped. Subsequently, some iodine desorbs and is released via the stack. Desorption depends on the age of the carbon filters. While the maximum age of the carbon filters is 30 months, the average age is 19 months because the filters are replaced regularly. The dose calculations use 19 months as the carbon filter age. Desorption rates are given in Table 3-13.

The source of radioiodine released from the stack is 50 percent of the iodine released from the fuel multiplied by the fraction desorbed. The calculation input includes the desorption fraction and the inventory.

3.7 References

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3-10 Pendergast, M. M., NRC-145-2, a Computer Code to Assess Dose from Accidental Pollutant Releases, DP-1646, May 1983.

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3-14 Pillinger, W. L., and W. L. Marter, Standardized Dose Factors for Dose Calculations, DPST-82-708, Revised, August 16, 1982.

3-15 DOE/EH-0071, Internal Dose Conversion Factors for Calculation of Dose to the Public, U. S. Department of Energy, July 1988, Derived from ICRP-30, limits for Intakes of Radionuclides by Workers, International Committee on Radiation Protection, July 1978.

Table 3–1

Maximum Whole–Body Dose During Evacuation from Reactor Areas

	Dose, Rem		
	P[a]	L	K
Plume dispersing opposite direction of evacuation (evacuation effective[b])	0	0.1	1
Plume overhead during evacuation (evacuation not effective[c])	22	22	21

(a) The dose at both rally points in P Area is too small to calculate for effective evacuation.

(b) When an effective evacuation occurs, the plume is assumed to be as close to personnel as the evacuation plan allows. Each reactor has two or more rally points so that personnel will exit from the area as far away from the direction of the plume as possible. An upper limit estimate of the dose is reported above using an increased stack height equal to the distance from the rally point to the plume centerline, assuming the entire 15–minute exposure takes place at this point. This dose is the maximum that would occur to personnel evacuating correctly. The expected dose should be much lower. The plume will likely be farther from personnel than assumed. Dose is for "very unlikely" meteorological conditions (not exceeded 99.5% of the time).

(c) "Evacuation not effective" means the dose calculation assumes the plume passed over personnel during evacuation from the reactor area. This dose is the same as the dose at 0.5 mile from the reactor directly under the plume. Dose is for meteorological conditions not exceeded 50% of the time.

Note: During evacuation, the dose from the carbon filter on the 105 Building is not added to the dose from the plume. The dose from the carbon filters is not included because personnel should spend only a short time in transit near the 105 Building, where they might be exposed to radiation from the carbon filters

Table 3-2

K Reactor Effective Dose Equivalent, 3% Core Melt

Distance (mi.)	Dose, rem 99.5%	Dose, rem 50%	X/Q, in units of 10^{-6} (sec/m^3) 99.5%	X/Q, in units of 10^{-6} (sec/m^3) 50%
0.5	33.84	20.85	8.00	2.45
1.0	28.35	13.32	5.73	2.23
1.5	25.35	9.42	5.56	2.38
2.0	23.13	7.08	5.36	1.92
4.0	12.57	2.94	5.83	0.92
Boundary	5.91	0.47	4.56	0.42

Note: These calculations used dose conversion factors from ICRP-30, in which the effective dose equivalent replaces the whole body dose. Meteorological data measured from 1982 through 1986 was used to calculate the *X/Q* and dose values. The exposure time is two hours. The percentages represent the percent of the time for which the meteorological conditions producing the dose are not exceeded. The dose not exceeded 99.5% of the time is calculated for each of 16 sectors; the reported dose is the maximum dose from the 16 sectors at the specified distances. The dose not exceeded 50% (typical) of the time is calculated by considering doses from all 16 sectors (entire site) together. The dose is reported at successive concentric rings to the SRS boundary. This method of reporting doses is used in Tables 3-2 through 3-10.

Table 3-3

P Reactor Effective Dose Equivalent, 3% Core Melt

Distance (mi.)	Dose, rem 99.5%	Dose, rem 50%	X/Q, in units of 10^{-6} (sec/m^3) 99.5%	X/Q, in units of 10^{-6} (sec/m^3) 50%
0.5	32.46	21.57	7.27	2.73
1.0	28.89	13.53	6.00	2.34
1.5	23.70	9.54	5.34	2.31
2.0	21.99	7.14	5.04	1.95
4.0	12.48	2.96	4.16	0.92
Boundary	4.80	0.44	3.86	0.35

Note: These calculations were made using dose conversion factors from ICRP-30, in which the effective dose equivalent replaces the whole body dose. Meteorological data measured from 1982 through 1986 was used to calculate the *X/Q* and dose values. The percentages represent the percent of the time for which the meteorological conditions producing the dose are not exceeded. The dose not exceeded 99.5% of the time is calculated for each of 16 sectors; the reported dose is the maximum dose from the 16 sectors at the specified distances. The dose not exceeded 50% (typical) of the time is calculated by considering doses from all 16 sectors (entire site) together. The dose is reported at successive concentric rings to the SRS boundary.

Table 3–4

L Reactor Effective Dose Equivalent, 3% Core Melt

Distance (mi.)	Dose, rem 99.5%	Dose, rem 50%	X/Q, in units of 10^{-6} (sec/m^3) 99.5%	X/Q, in units of 10^{-6} (sec/m^3) 50%
0.5	43.86	22.02	8.68	2.45
1.0	37.65	13.32	9.05	2.58
1.5	33.60	9.75	10.31	2.58
2.0	27.75	7.20	10.75	2.05
4.0	15.84	2.98	7.87	0.97
Boundary	4.11	0.51	5.05	0.42

Note: These calculations were made using dose conversion factors from ICRP-30, in which the effective dose equivalent replaces the whole body dose. Meteorological data measured from 1982 through 1986 was used to calculate the *X/Q* and dose values. The percentages represent the percent of the time for which the meteorological conditions producing the dose are not exceeded. The dose not exceeded 99.5% of the time is the maximum dose from any of the 16 sectors at each specified distances. The dose not exceeded 50% (typical) of the time is calculated by considering doses from all 16 sectors (entire site) together. The dose is reported at successive concentric rings to the SRS boundary.

Table 3-5

K Reactor Thyroid Dose, 3% Core Melt

Distance (mi.)	Dose, rem 99.5%	Dose, rem 50%	X/Q, in units of 10^{-6} (sec/m^3) 99.5%	X/Q, in units of 10^{-6} (sec/m^3) 50%
0.5	7.62	2.46	8.00	2.45
1.0	5.37	2.19	5.73	2.23
1.5	4.98	2.22	5.56	2.38
2.0	4.77	1.83	5.36	1.92
4.0	4.89	0.87	5.83	0.92
Boundary	3.54	0.39	4.56	0.42

Note: The thyroid dose is based on an accident releasing radioactive iodine (fission products) from the melting of three percent of the core. The iodine is released from the stack after desorbing from the carbon filters. Percentages are the percent of the time for which the meteorological conditions are not exceeded.

Table 3–6

P Reactor Thyroid Dose, 3% Core Melt

Distance (mi.)	Dose, rem 99.5%	Dose, rem 50%	X/Q, in units of 10^{-6} (sec/m^3) 99.5%	X/Q, in units of 10^{-6} (sec/m^3) 50%
0.5	6.96	2.73	7.27	2.73
1.0	5.58	2.35	6.00	2.34
1.5	4.80	2.25	5.34	2.31
2.0	4.38	1.86	5.04	1.95
4.0	3.54	0.87	4.16	0.92
Boundary	3.03	0.33	3.86	0.35

Note: The thyroid dose is based on an accident releasing radioactive iodine (fission products) from the melting of three percent of the core. The iodine is released out the stack after desorbing from the carbon filters. Percentages are the percent of the time for which the meteorological conditions are not exceeded.

Table 3–7

L Reactor Thyroid Dose, 3% Core Melt

Distance (mi.)	Dose, rem 99.5%	Dose, rem 50%	X/Q, in units of 10^{-6} (sec/m^3) 99.5%	X/Q, in units of 10^{-6} (sec/m^3) 50%
0.5	8.31	2.52	8.68	2.45
1.0	8.22	2.52	9.05	2.58
1.5	9.09	2.43	10.31	2.58
2.0	9.21	1.95	10.75	2.05
4.0	6.51	0.91	7.87	0.97
Boundary	4.04	0.39	5.05	0.42

Note: The thyroid dose is based on an accident releasing radioactive iodine (fission products) from the melting of three percent of the core. The iodine is released from the stack after desorbing from the carbon filters. Percentages are the percent of the time for which the meteorological conditions are not exceeded.

Table 3–8

K Reactor Particulate Dose from Melting (3%) and Moderator Spill Dose

Moderator Spill

Distance (mi.)	Eff. Dose Equiv. rem 99.5%	Eff. Dose Equiv. rem 50%	X/Q, in units of 10^{-6} (sec/m^3) 99.5%	X/Q, in units of 10^{-6} (sec/m^3) 50%
0.5	0.044	0.013	8.00	2.45
1.0	0.031	0.012	5.73	2.23
1.5	0.030	0.013	5.56	2.38
2.0	0.029	0.010	5.36	1.92
4.0	0.032	0.005	5.83	0.92
Boundary	0.025	0.002	4.56	0.42

Dose from Particulate Pu^{239}

Bone & Effective Dose Equivalent

Distance (mi.)	Bone Dose 99.5%	Bone Dose 50%	Eff. Dose Equiv. 99.5%	Eff. Dose Equiv. 50%	X/Q, in units of 10^{-6} (sec/m^3) 99.5%	X/Q, in units of 10^{-6} (sec/m^3) 50%
1.0	0.610	0.237	0.033	0.013	5.73	2.23
1.5	0.592	0.253	0.032	0.014	5.56	2.38
2.0	0.571	0.204	0.031	0.011	5.36	1.92
4.0	0.621	0.098	0.034	0.005	5.83	0.92
Boundary	0.486	0.045	0.027	0.002	4.56	0.42

The effective dose equivalent for the moderator spill accident is from tritium released from the stack after a moderator spill accident.

The release of particulate Pu^{239} is in the release from the stack after a three percent core melting accident. Through inhalation the particulates affect the bone and cause an effective dose equivalent. X/Q is the downwind dilution factor in sec/m^3. The bone dose refers to bone surface.

Table 3-9

L Reactor Particulate Dose from Melting (3%) and Moderator Spill Dose

Moderator Spill

Distance	Eff. Dose Equiv. rem		X/Q, in units of 10^{-6} (sec/m^3)	
(mi.)	99.5%	50%	99.5%	50%
0.5	0.047	0.013	8.68	2.45
1.0	0.049	0.014	9.05	2.58
1.5	0.056	0.014	10.31	2.58
2.0	0.058	0.011	10.75	2.05
4.0	0.043	0.005	7.87	0.97
Boundary	0.027	0.002	5.05	0.42

Dose from Particulate Pu^{239}

Bone & Effective Dose Equivalent

Distance	Bone Dose		Eff. Dose Equiv.		X/Q, in units of 10^{-6} (sec/m^3)	
(mi.)	99.5%	50%	99.5%	50%	99.5%	50%
0.5	0.924	0.261	0.051	0.014	8.68	2.45
1.0	0.964	0.275	0.053	0.015	9.05	2.58
1.5	1.098	0.275	0.060	0.015	10.31	2.58
2.0	1.145	0.218	0.063	0.012	10.75	2.05
4.0	0.838	0.103	0.046	0.006	7.87	0.97
Boundary	0.538	0.045	0.029	0.002	5.05	0.42

The effective dose equivalent for the moderator spill accident is from tritium released from the stack after a moderator spill accident.

The release of particulate Pu^{239} is in the release from the stack after a three percent core melting accident. Through inhalation the particulates affect the bone and cause an effective dose equivalent. X/Q is the downwind dilution factor in sec/m^3.

Table 3-10

P Reactor Particulate Dose from Melting (3%) and Moderator Spill Dose

Moderator Spill

Distance (mi.)	Eff. Dose Equiv. rem 99.5%	Eff. Dose Equiv. rem 50%	X/Q, in units of 10^{-6} (sec/m^3) 99.5%	X/Q, in units of 10^{-6} (sec/m^3) 50%
0.5	0.040	0.015	7.27	2.73
1.0	0.033	0.013	6.00	2.34
1.5	0.029	0.013	5.34	2.31
2.0	0.027	0.011	5.04	1.95
4.0	0.023	0.005	4.16	0.92
Boundary	0.021	0.002	3.86	0.35

Dose from Particulate Pu^{239}

Bone & Effective Dose Equivalent

Distance (mi.)	Bone Dose 99.5%	Bone Dose 50%	Eff. Dose Equiv. 99.5%	Eff. Dose Equiv. 50%	X/Q, in units of 10^{-6} (sec/m^3) 99.5%	X/Q, in units of 10^{-6} (sec/m^3) 50%
0.5	0.774	0.291	0.042	0.016	7.27	2.73
1.0	0.639	0.249	0.035	0.014	6.00	2.34
1.5	0.569	0.246	0.031	0.013	5.34	2.31
2.0	0.537	0.208	0.029	0.011	5.04	1.95
4.0	0.443	0.098	0.024	0.005	4.16	0.92
Boundary	0.411	0.037	0.023	0.002	3.86	0.35

The effective dose equivalent for the moderator spill accident is from tritium released from the stack after a moderator spill accident.

The release of particulate Pu^{239} is in the release from the stack after a three percent core melting accident. Through inhalation the particulates affect the bone and cause an effective dose equivalent. X/Q is the downwind dilution factor in sec/m^3.

Table 3-11

Iodine and Noble Gas Inventory of Core at 3,000 MW
(Major Contributors to 2 Hour Offsite Dose)

Isotope	t 1/2	Inventory MCi	Isotope	t 1/2	Inventory MCi
I-131	8.04 d	74	Kr-87	76 m	35
I-132	2.28 h	95	Kr-88	2.8 h	75
I-133	20.8 h	170	Xe-133	5.3 d	172
I-134	52.6 m	122	Xe-133m	2.2 d	25
I-135	6.58 h	153	Xe-135	9.16 h	20
		614 MCi Iodine	Xe-135m	15.3 m	30
					332 MCi Noble Gases

Table 3–12

Relative Concentration Factors at the Plant Boundary for the Worst Sector [a]

X/Q, in units of 10^{-6} sec/m^3 99.5%	50%	Sector	Reactor
3.86	0.35	N	P
5.05	0.42	W	L[b]
4.56	0.42	NW	K

(a) The X/Q values in this table were calculated with the computer code NRC 145–2. These X/Q values are "undecayed" values; that is, they do not account for decay during transit from release point to receptor point. Meteorological data from 1982 through 1986 was averaged for a one-hour period.

(b) Estimated based on values from K Reactor since there is no meteorological tower at L Reactor. Percentages represent the percent of the time (99.5 and 50%) that meteorological conditions are not exceeded.

Table 3–13

Cumulative Release Fractions for Iodine Loaded on Confinement System Carbon Beds Due to Radiolytic Desorption Mechanisms [a]

Time After Loading, hours	Cumulative Fraction of Iodine Release as a Function of Carbon Service Age, months: 6	14	19	30
0	0.00050	0.00050	0.00050	0.00050
1	0.00073	0.00084	0.00140	0.00220
2	0.00084	0.00116	0.00190	0.00400
3	0.00093	0.00138	0.00230	0.00503
4	0.00102	0.00156	0.00256	0.00575
5	0.00108	0.00171	0.00280	0.00630
9	0.00130	0.00200	0.00341	0.00760
18	0.00178	0.00248	0.00422	0.00963
24	0.00210	0.00280	0.00476	0.01053
48	0.00338	0.00408	0.00722	0.01593
72	0.00465	0.00535	0.00908	0.02133
96	0.00593	0.00663	0.01124	0.02673
120	0.00721	0.00791	0.01340	0.03213

(a) Data include assumed 0.05 percent instantaneous filter bypass and observed desorption rates for GX-176 carbon service aged in the SRS confinement system.

Note: Values given are fractions of filter inventory, not core inventory. Filter inventory is assumed to be 50 percent iodine released from the core.

Table 3–14

Inhalation Dose Factors for Tritium and Radioiodines

Nuclide	Dose Factor [a] (Rem/Ci inhaled) Effective Dose Equivalent	Thyroid
H–3	0.95×10^2	0.95×10^2
I–131	32×10^3	1.1×10^6
I–132	3.3×10^2	0.63×10^4
I–133	5.4×10^2	1.8×10^5
I–134	11×10^1	1.1×10^3
I–135	11×10^2	3.1×10^4

Adult dose commitment factors from ICRP–30 (Ref. 3–15). Calculations assumed an active man breathing rate of 3.47×10^{-4} m^3/sec.

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SEVERE ACCIDENTS

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4.0 SEVERE ACCIDENTS

4.1 Introduction and Purpose

The purpose of this chapter is to summarize the risks to the general public and site staff from hypothetical severe accidents at the Savannah River reactors. Chapter 2 presented information on the predicted response of the Savannah River reactors to postulated design basis events. Design basis events consist of those events which are postulated as part of the design process, with subsequent plant response engineered to prevent or minimize fuel damage and to mitigate the consequences of fuel damage. A single failure of an active system or component is assumed in design basis analyses.

Severe accident analyses go beyond design basis analyses by assuming multiple failures of plant equipment or systems. Typical frequencies of these failure combinations are on the order of 2 to 3 per 100,000 years of reactor operation. Any number of initiating events and subsequent equipment failures can be considered, leading to hypothetical conditions which can result in severe consequences such as extensive fuel damage and large releases of radionuclides.

The risks to the Savannah River Site work force and to the population in the vicinity of the Savannah River Site due to hypothetical severe accidents at the Savannah River reactors have been determined using probabilistic risk assessment (PRA) techniques. The PRA methodology is composed of three distinct parts, which when combined, estimate the overall severe accident risks.

The first part of a PRA, termed the Level 1 analysis, includes investigations of the response of the reactors and their safety systems to a wide range of accident initiators. These accident initiators include internal initiators such as pipe breaks and plant transients, as well as a wide range of external initiators such as earthquakes, fires, and floods. The principal result of the Level 1 analysis is the determination of the likelihood of severe accidents which involve damage to the reactor core and subsequent release of the radionuclides from the fuel and target materials. The Level 1 analysis identifies the accident sequences and equipment failures which dominate the potential for a severe accident and estimates their frequency of occurrence.

The second part of a PRA, termed the Level 2 analysis, is an investigation of the response of the reactor safety systems designed to mitigate the consequences of a severe accident. The results of the Level 2 analysis are expressed as radionuclide source terms released to the environment as a result of severe accident sequences identified in Level 1. The conditional probability of release is also calculated. The radionuclide source terms are the quantities of radioactive materials that are predicted to be released and the time and duration of the release.

The third part of a PRA, termed the Level 3 analysis, is an investigation of the consequences resulting from a release of radioactive material during a hypothetical severe accident. The consequences can be expressed as predicted radiation exposure to the population or as the resultant predicted health effects to persons in the vicinity of the plant.

The mathematical combination of the results from the three levels of the PRA yields the predicted risk due to severe accidents associated with operation of the reactors. Risk may be defined as the product of the frequency of a severe accident sequence (Level 1), the conditional probability of radioactive release for that sequence (Level 2), and the consequences of the release (Level 3), summed over all sequences.

The ground rules established for performing this assessment of severe accident risk for the Savannah River reactors are:

(1) The analyzed reactor configuration represents the configuration of the reactors at the time of restart, based on planned and implemented safety improvements.

(2) The initiating events to be considered in the assessment include both internal events, such as pipe breaks, and external events, such as earthquakes.

(3) The assessment is based on best estimate methods wherever possible. Conservatism in analysis and evaluation methods is acceptable if no best estimate methods are available.

(4) The reactor power is assumed to be 3000 megawatts thermal, a level which is not expected to be exceeded.

(5) The modelling of severe accidents used for this assessment is based on the K Reactor. A review of the important features of the K Reactor design that control the severe accident frequencies and/or consequences has been conducted, and those features have been compared to the L Reactor and P Reactor designs. The conclusion from this comparison is that the severe accident risk results for the K Reactor are directly applicable to the L Reactor and P Reactor.

(6) The predicted severe accident frequencies and consequences are based on fuel assemblies used for the production of tritium in the reactors. A qualitative description is provided at the end of Section 4 of differences in the predicted frequencies and consequences of severe accidents for reactor operation with assemblies used for the production of other special nuclear materials.

4.2 Systems Analysis and Core Damage Frequency Determination

4.2.1 Systems Description

The Savannah River reactors produce nuclear materials by irradiating target materials with neutrons produced from the fission of uranium. The fission process also produces heat which must be removed from the uranium fuel. The heat is removed by circulating heavy water (deuterium oxide) through the fuel and then through heat exchangers which transfer the heat to light water which is pumped from the Savannah River. Accidents which would cause the release of radioactive materials from the fuel or targets must involve either the production of more heat than the water can remove, or a significant reduction in the flow of water. Accidents of the former type are called reactivity or power transients and the latter type are called flow reduction events. Emergency shutdown and cooling systems are installed in the reactors to prevent any power transients or flow reduction events from leading to fuel or target damage.

Two independent and redundant reactor shutdown systems are provided to prevent reactivity or power transients from damaging the reactor fuel or targets. The primary shutdown system, the safety rod system, is effective in reducing reactor power within one second after initiation. If the safety rod system should fail, a backup system, called the supplementary safety system, is effective within less than five seconds. These systems are actuated by redundant actuation systems which monitor the process variables, such as temperature, pressure, and neutron flux, needed to detect any condition which could lead to fuel damage.

Under normal circumstances, after a shutdown in response to a transient, the full reactor cooling flow continues to remove radionuclide decay heat without any valve operations or adjustments. This includes the flow of heavy water through the reactor and the flow of light water to the secondary side of the heat exchangers. If the normal full flow is interrupted by a loss of normal electrical power, the flow of light water will continue by gravity alone in a quantity sufficient to remove the fission product decay heat. Heavy water flow is provided by diesel–electric powered motors which are always operating when the reactor is operating. Thus, these systems do not require any change in power sources, or that any emergency equipment be started to provide cooling following a transient event.

If heavy water circulation through the reactor is interrupted by a pipe break or by flooding of the circulating pump motors resulting from a pipe break, the emergency cooling system (ECS) will cool the fuel with light water. This system can be powered by emergency diesel–generators if normal electrical power is lost.

A system, called the moderator recovery system (MRS), is provided to recover heavy water lost from small pipe breaks or other leaks and return it to the reactor so that the ECS would not be required. It consists of two sump pumps which can be manually actuated.

In addition, a number of sump pumps are provided as part of a water disposal system (WDS) to remove water from the lower level of the reactor building to prevent flooding of the primary system pumps. Water discharged from the reactor building by the WDS is collected in tanks with a capacity of more than 500,000 gallons. Water volumes in excess of this amount would be collected in a retention basin with a capacity of 50 million gallons.

Each reactor has a concrete basin with a capacity of 25 million gallons to store cooling water for use if the supply of water from the Savannah River should fail. Pumps are provided to recirculate the effluent water from the reactor heat exchangers to the basin to provide long–term shutdown cooling if supply from the river is interrupted. It is also possible to provide shutdown cooling water to all three reactors from Par Pond if necessary.

The confinement system consists of several subsystems: the airborne activity confinement system (AACS), the confinement heat removal system (CHRS), and the reactor room spray system (RRSS). These subsystems are designed to work together to confine radionuclides which may be released during severe accidents.

The AACS filters the air exhausted from the reactor building through moisture separators, high efficiency particulate air (HEPA) filters, and activated charcoal filters before it is discharged from a stack 62 meters (200 feet) in height. Retention of particulate radionuclides and radioiodine by these filters is very efficient.

The CHRS has two purposes. It is intended to protect the fans and filters of the AACS by limiting the temperature of air or steam reaching the filters from the areas below grade level, and to retain radionuclides and other aerosols in water within the below grade areas of the reactor building. This is accomplished by assuring that any core debris reaching the reactor building lower level floor would release heat to water rather than to air, and assuring that radionuclides escaping from core debris on the floor would have to pass through water, with a likelihood of retention in the water. This is accomplished by flooding the lower level floor with water from the disassembly basin if clear evidence is present that fuel melting has occurred or will occur.

The RRSS is intended to provide cooling to the reactor process room to protect the AACS filters from heat. In addition, the RRSS provides some scrubbing of airborne radioactivity and other aerosols from air exhausted from the process room. Water from the RRSS drains from the process room floor to the tanks of the WDS.

4.2.2 Identification of Initiating Events

To estimate the frequencies of occurrence of severe accidents, it is necessary to estimate the frequencies of initiating events which can cause such accidents, and to combine these with the probability of failure of the systems which are provided to prevent the events from becoming accidents. Initiating events are those occurrences which, in combination with protective system failures or other occurrences, can lead to accidents. The initiating events considered in a PRA include internal initiators such as pipe breaks and plant transients, and external initiators such as earthquakes, fires, and floods. In practice, some events are known to be so improbable that they need not be considered. Meteorite strike is perhaps the best example of such an event.

In the ongoing PRA effort, an extensive large list of internal initiators has been considered. It was determined that, for the purposes of this analysis, most could be condensed into five general classes; a heavy water or primary system loss–of–coolant accident (LOCA), a light water or secondary system LOCA, called a loss–of–pumping accident (LOPA) because of the flooding of the cooling water pumps, a loss-of-heat-sink accident (LOHSA), a loss-of-river-water (LORW) supply, and reactivity and coolant flow transients. The remaining initiators were not considered in this analysis because of their negligible contribution to risk. The grouping of initiators into classes was possible and practical because the initiators in each of the above groups require the same responses from the reactor systems and operators. Table 4.2-1 shows the individual initiators that were condensed into these five initiator classes. Also shown are several examples of low risk initiators that were dismissed because of extremely low probability, or because the consequences would be so small that they represent a negligible contribution to risk.

The "negligible risk" initiators were considered and determined unimportant in this analysis because, although they may be more probable than accidents which would lead to melting of a large fraction of the fuel in a reactor at power, the quantity of radioactive materials released would be far less than that released from a major fuel melt accident. For example, criticality during charging of fuel to a reactor can happen only if a large number of errors are made and remain undetected. This accident is somewhat more likely to occur with a charge intended to produce plutonium rather than tritium, but is still not a significant contributor to risk because of the low inventory of radionuclides available for release.

It is important to note that the Savannah River reactors do not produce electrical power and, thus, do not have steam turbines or electrical generators. A great many of the initiating events which must be considered in a PRA for a nuclear power generating plant, therefore, need not be considered for the Savannah River reactors. Also, because the Savannah River reactors do not produce electrical power, a reactor shutdown has no effect on the availability of electric power for safety purposes. The fact that the Savannah River reactors operate at low temperature and pressure also simplifies the analysis because some events are less likely to cause damage to equipment needed to cope with the accident. For example, the "high energy line break" concern in power generating reactors has no applicability to the Savannah River reactors. Published PRA documents for nuclear power plants were reviewed, however, to assure that no pertinent initiating events were overlooked.

The external initiators that were explicitly considered were earthquake and flood. The seismic PRA is discussed in Section 4.2.4. The only relevant result of external flooding is the loss of the supply of river water by flooding of the river water pump houses, the only important structures less than one hundred feet above river level. This event is explicitly considered as an internal initiator. The only relevant result of high wind is loss of normal electric power supply to the reactor building, which was explicitly considered as an initiator and found to be insignificant. Therefore, flooding and high wind are not addressed as separate initiators. Toxic gas releases were not considered because no toxic gases are used in or near the reactor areas and no public transportation routes are near the reactors. Volcanic eruptions and meteorite impacts are so improbable in the area of Savannah River Site that they do not significantly contribute to risk.

4.2.3 Initiating Event Frequencies

The frequencies of initiating events were determined, whenever possible, from the actual records collected in the 110 reactor-years of operation of the Savannah River reactors. Where data were lacking, such as for pipe rupture, data generic to the nuclear power industry were used. For external events, historical data were used where possible.

Reactivity and power transients are the most frequent initiating events, but require no response other than reactor shutdown and continuation of normal or shutdown fuel cooling. Single control rod withdrawal events have occurred in the reactors, but were usually of such short duration that the safety rod system was not demanded. The frequency of single control rod withdrawal events is 4.0×10^{-1} per reactor year. The control rods are divided electrically into three "gangs", and occasionally an entire gang of rods will withdraw partially from a reactor. The frequency of such events is conservatively estimated as 1.0×10^{-1} per reactor-year. The component motion initiator refers to hypothetical horizontal motion of reactor fuel or a control rod guide housing. This could occur if such a component is not properly seated in the reactor. While no transient has occurred as a result of this, there have been six instances of improperly seated components in 110 reactor years of operation. The initiator frequency is conservatively determined from six occurrences in 110 reactor years to be 5.5×10^{-2}. Flow reduction transients caused by failure of a primary coolant pump or associated drive motor (including loss of electrical power) have occurred at a frequency of 8.7×10^{-1} per reactor-year. The corresponding frequency for the secondary system pumps or motors is 5.4×10^{-1} per reactor-year. The historical frequency of valves inadvertently closing to any degree at all in the primary coolant system is 4.8×10^{-1} per reactor-year. The corresponding frequency for valves in the light water cooling system is 2.1×10^{-2}. These frequencies are considered conservative because not all of the historical events were sufficiently large to challenge the safety rod system. Plugging of the heat exchangers was considered, but all conceivable mechanisms would occur so slowly that the shutdown systems would not be challenged. The inadvertent addition of light water to the fuel coolant channels could cause a large reactivity increase and consequent power increase, but no credible mechanisms could be identified and thus the frequency is estimated to be about 1×10^{-8} per reactor year. The total frequency of power and reactivity transients is about 2.5 per reactor year

The probability of pipe rupture in the primary coolant system was estimated by the use of the leak-before-break methodology as applied to nuclear power plant assessments. An important difference in the piping for the Savannah River reactors is the use of expansion joints. These joints were incorporated in the original design and serve to minimize the stress in the piping. The bellows of these joints are now thought to be the most probable part of the system to rupture, although the resulting leak rate would be far less than that which would result from a double-ended guillotine pipe break. The frequencies for primary system LOCAs vary from 9.5×10^{-7} to 2.8×10^{-3} per reactor year, depending on the location and size of the leak. The total frequency of primary system leakage large enough to require safety system actuation is 5.6×10^{-3} per reactor year.

The probability of pipe rupture in the secondary coolant system was estimated by the use of the same models typically used in PRA for nuclear power reactors. Like the primary system, the secondary system also contains expansion joints which are thought to be the most probable part of the system to rupture, although the resulting leak rate would again be much less than that from a double-ended guillotine pipe break. The resulting frequencies for secondary system LOCAs vary from 1.3×10^{-5} to 1.0×10^{-3} per reactor year, depending on the size of the leak, with a total frequency of 3.3×10^{-3} per reactor year.

The initiator described as loss of river water supply refers to the condition in which the river water flow to the inlet basin stops, but the reactor continues to operate, exhausting the water supply from the 25-million-gallon basin. The frequency of this initiator, 1.2×10^{-3} per year, is the sum of the frequencies for the complete loss of the plant electrical grid (where power to the river water pumps and the reactor area is lost) and for initiators that affect only the river water supply, where all other reactor operating facilities remain intact.

The loss of heat sink initiator is an event in which the heat removal capability is suddenly removed, by closure of a large number of valves or other circumstances, without a pipe break. The frequency of loss of heat sink, 1.2×10^{-4} per reactor year, is the sum of the frequencies of many conditions which would cause complete loss of primary water circulation, or complete loss of secondary water circulation.

4.2.4 Systems Analysis

To estimate the frequency of occurrence of severe accident sequences it is necessary to combine the estimates of the frequencies of the various initiating events with the probability of failure of the systems which are provided to prevent the events from becoming accidents. The accident sequences for internal initiators were quantified by the use of event trees and fault trees. An event tree is a logic model of the possible combinations of system responses to an initiating event. The model produces a list of the possible outcomes, or success states, that can result from the initiating event. The tree is read from left to right, beginning with the initiating event and following through the success or failure of the relevant systems as denoted in the "top events" along the tree. An upward branch denotes system success, and a downward branch denotes failure of the system to perform its function. Each unique pathway through an event tree is called a sequence. Thus, a sequence is a unique combination of system responses that would lead to one of two plant states at the end of the sequence: success in preventing fuel damage, or failure to prevent fuel damage.

A fault tree is a logic model of an individual system, and is used to find the combinations of component failures and human errors that would cause failure of the system to perform adequately. The "top event" in a fault tree is the system failure as defined in the relevant event tree. Below the top event are all of the possible combinations of failures which can lead to the top event, connected by combinatorial logic symbols known as "and gates" and "or gates". The intermediate events are those which can cause the failure immediately above in the tree. The bottom events are the "basic events" in the sense that they need not be developed further because a probability value is available for the event. The probability of system failure may be found from the probabilities of component failures and human errors by reducing the fault tree logic to Boolean algebra and providing the component failure probabilities and human error probabilities. Fault trees are used to determine the system failure probabilities for the system failures represented in the event trees.

The probability of fuel damage resulting from an initiating event may be found from a knowledge of the failure probabilities of individual components in the protective systems. If event trees are prepared for each of the classes of initiating events, and the frequencies of the initiating events are determined, the total frequency of fuel damage can be found.

The component failure probabilities were determined whenever possible from the actual records collected in the 110 reactor-years of operation of the Savannah River reactors. Where data were lacking, such as for newer components, data generic to the nuclear power industry were used. Human error probabilities were determined by methods which have become standard in the nuclear power industry, including the Technique for Human Error Rate Prediction (THERP) methodology, and time-reliability correlations, as described in NUREG/CR-1278 and NUREG/CR-2254.

Seismic events could lead to severe accidents by causing one or more of the initiators previously discussed. In addition, a seismic event could cause failure of the confinement systems as discussed later in this chapter. A seismic PRA differs from a PRA performed for internal initiators only in that many degrees of seismic excitation must be considered, requiring in essence that the PRA be performed many times. The response of the reactors to each level of seismic excitation was determined in a manner very similar to that used for internal initiators with the exception that the failure probabilities for components of the safety systems are a function of seismic acceleration in addition to the probability of failure from random causes. The seismic dependencies are represented by fragility curves. A fragility curve for a component defines the failure probability of the component as a function of seismic acceleration.

Degrees of seismic acceleration and associated frequency of occurrence are provided by a seismic hazard model. The seismic hazard model used for this work was developed by the Electric Power Research Institute (EPRI) and was applied specifically to the SRS reactors. The EPRI hazard curve was developed to enable calculation of the seismic risk at any location in the eastern United States. This EPRI model is one of several alternative representations of the frequency of recurrence of seismic events.

An analysis of the core damage frequency which would result from fires internal to a reactor building was also performed. This analysis is performed by identifying the zones in the reactor building which have a significant probability of occurrence of a fire, and have equipment or electrical cables important to safe long-term shutdown. For building zones which meet these criteria, fire temperature response is determined, component fire fragilities are determined, fire barrier failure probabilities are determined, and fire recovery analysis is performed.

A fire in the reactor building is very unlikely to occur simultaneously with any internal initiating event other than a transient, and the only actions required to respond to either a fire or transient are to shut the reactor down and maintain adequate cooling of the fuel. Therefore, only one event tree was required for this analysis. The initiating event frequency for fires was based on the experience of commercial nuclear power plants, combined with the Savannah River reactor experience. The resulting fire initiating event frequency was 1.2×10^{-1} per reactor year.

4.2.5 Summary of Core Damage Frequency Analyses

The total core damage frequency from all initiators is estimated to be 2.0×10^{-4} per reactor year. Sixty-five percent of this total is attributed to internal initiating events, and nearly thirty-five percent to seismic events. Contributors to core damage frequency are shown in Table 4.2-2 and are discussed in following sections. The core damage frequency resulting from internal initiators is 1.3×10^{-4} per reactor year.

The frequency of severe accidents resulting from seismic events is 6.8×10^{-5} per reactor year. The seismically induced failures which would lead to a severe accident are dominated by breaks of cooling water pipes, accounting for seventy-nine percent of the total frequency. The next largest contributor is loss of river water supply, accounting for thirteen percent of the total. The remainder involve breaks in the primary cooling system.

The frequency of severe accidents resulting from a fire is 1.4×10^{-7} per reactor year, which is a negligible contribution to the total severe accident frequency. The low frequency of severe accidents from fire results from the widely separate, redundant supplies of primary and secondary cooling water provided in SRS reactors, and the reactor's minimal post-shutdown cooling requirements. No credible single fire event can defeat both principal and emergency cooling water systems concurrently.

4.2.5.1 Primary System Pipe Rupture (LOCA)

Primary system pipe ruptures are divided into two categories, large LOCAs which are major pipe breaks, and small LOCAs that are of the size that might be expected of leaks from the expansion joint bellows. Small LOCAs are further divided into leaks greater than 88 liters/sec (1400 gpm), leaks less than 63 liters/sec (1000 gpm), and leaks intermediate to those two categories. This division is based on the requirements for human action in response to the different leak sizes. In the case of leaks exceeding 88 liters/sec (1400 gpm), actuation of the ECS is automatic. For leaks less than 63 liters/sec (1000 gpm), human action is required for the actuation of either the ECS or the MRS. The probability of human error is known to be highly influenced by the degree of stress to which the human is subjected. Human actuation of the ECS is considered to be a high stress condition because unnecessary actuation would result in light water degradation of the heavy water, and the need to recover the degraded heavy water from the WDS tanks. Human actuation of the MRS is considered a low stress condition because actuation of the system has no undesirable consequences, even if the actuation occurs unnecessarily. For the intermediate leaks between 63 liters/sec (1000 gpm) and 88 liters/sec (1400 gpm), the MRS flow is not adequate to maintain the reactor tank inventory and the ECS must eventually be actuated manually. This manual actuation is not needed for at least thirty minutes and, thus, is considered a low stress situation.

For all large leaks, and for smaller leaks if the WDS should partially fail, it is necessary to manually throttle the ECS flow to match the capacity of the WDS so that the primary water circulating pump motors do not become flooded. Over-throttling can result in fuel damage. Over-throttling is the cause of about ten percent of the hypothetical LOCA-induced fuel damage frequency. Failure to throttle the ECS flow results in flooding of the d.c. drive motors, which complicates recovery from this accident, and requires that fuel cooling be provided by the ECS for the long term.

Primary system pipe rupture is a major contributor to the total core damage frequency, accounting for thirty-four percent of the total. Of this, small pipe or bellows breaks account for seventy-four percent of the LOCA-induced core damage frequency, because ECS is not automatically actuated for small breaks, and because a conservative estimate is used or operator error in actuating either the ECS or the MRS.

4.2.5.2 Secondary System Pipe Rupture (LOPA)

Secondary system pipe ruptures constitute about seventeen percent of the total core damage frequency. Of these, the most important are breaks of very large cooling water headers, for which the leak rate will overwhelm the WDS and flood out the primary water circulating pump motors. Fuel damage resulting from these severe accident sequences requires that the operators fail to isolate the leak to prevent flooding, and also fail to actuate the ECS, or else involve the long term failure of the ECS. Proper operator action will terminate the accident.

4.2.5.3 Reactivity or Power Transient

The only system response required to terminate a reactivity or power transient is that either one of the reactor shutdown systems function properly. Either the safety rod system or the supplementary safety system will terminate a transient without fuel damage. Transients contribute significantly to the total core damage frequency only because of the large initiating event frequency. However, a large fraction of the historical initiating events which contribute to this frequency were terminated long before the reactor shutdown system was required. Thus the estimated risk from transients is known to be conservative, but no further effort was made to reduce this estimate.

4.2.5.4 Loss of River Water Supply

The required actions in response to loss of river water supply are to shut the reactor down, reduce cooling water flow to a minimum consistent with required heat removal, and recirculate water from the outlet sump to the inlet basin. It is also possible to recirculate cooling water through the two cooling water headers in such a way that it will return to the inlet basin without using the outlet sump or the pumps in that sump. Approved procedures are in place for this operation. It is further possible to provide cooling water from Par Pond or wells, if the river pump house outage is expected to continue. The sequences that lead to fuel damage involve operator failure to recognize the event and to take corrective action on a timely basis. Proper operator action as described in approved procedures would prevent this accident. Two separate inlet basin level alarms are provided in the control room to warn of this event, and approved procedures are in place to provide notification by several different telephone systems if this event should occur.

4.2.5.5 Loss of Heat Sink

This class of accidents includes those sequences which could lead to a loss of circulation in either the primary or secondary systems, without a loss of water through a pipe break. Because of the definition of this class, ECS is required to mitigate this event. The most important failures in these sequences are a failure of the ECS to deliver adequate water to cool the fuel, or an operator error in over-throttling the ECS while trying to prevent flooding of the primary coolant pumps.

Table 4.2-1

Internal Initiating Events

Primary System Pipe Rupture (LOCA)
Nine Sizes and Types of Breaks, up to and Including Double-Ended Guillotine Break of the Largest Pipe
Secondary System Pipe Rupture (LOPA)
Nine Sizes and Types of Breaks, up to and Including Double-Ended Guillotine Break of the Largest Pipe
Reactivity or Power Transient
Single Control Rod Withdrawal
Gang Control Rod Withdrawal
Component Motion
Loss of Primary Pump or Motor
Primary System Valve Closure
Loss of Secondary Pump or Motor
Heat Exchanger Plugging
Light Water Addition
Loss-of-River-Water Supply (LORW)
Loss of Electrical Power
Flood
Loss-of-Heat-Sink Accident (LOHSA)
Loss of All Primary Flow
Loss of All Secondary Flow
Plugging of Heat Exchangers
Freezing of Heat Exchangers
Low Risk Initiators (not included in the analyses)
Melting of Fuel During Discharge
Criticality During Charging
Criticality Outside Reactor

Table 4.2-2

Core Damage Frequency By Accident Class

	Frequency (per Reactor Year)
Internal Events Core Damage Frequency	1.3 E-04
LOCA	6.8 E-05
LOPA	2.2 E-05
Transient	1.6 E-05
LORW	1.3 E-05
LOHS	9.9 E-06
Seismic Core Damage Frequency	6.8 E-05
Fire Core Damage Frequency	1.4 E-07
TOTAL Core Damage Frequency	2.0 E-04

4.3 Confinement Performance and Radionuclide Behavior

4.3.1 Introduction

The Savannah River reactors have confinement structures and safety systems to minimize the quantities of radioactive materials that would be released to the environment as a result of postulated severe accidents. The confinement and the safety systems have been designed to be effective in minimizing the releases of airborne radioiodine and "solid" (i.e., aerosol) fission product materials, but noble gases would be released unmitigated. These systems were designed to handle limited damage to the reactor core. For some of the severe accidents discussed in this chapter, these systems may be only partially effective and, in the extreme, their effectiveness could be considerably degraded. Their effectiveness is highly dependent on the extent of core damage and other characteristics of the sequence, in particular, the amount of water available.

4.3.2 Severe Accident Progression

This section provides an overview of the predicted progression of hypothetical severe accidents in the Savannah River reactors and is based on the results of research programs carried out at the Savannah River Site over the past several years. These programs have led to an improved understanding of severe accident behavior with the fuel types used in these reactors. The purpose of this discussion is to summarize the progression of hypothetical severe accident classes including the performance of the confinement system and the predicted resultant releases of radionuclides to the environment.

The accident classes which are the dominant contributors to the core damage frequency were outlined in earlier sections of this chapter and are shown in Table 4.2-2. The early progression of a severe accident can differ significantly among these accident classes. For the severe accident classes identified, the time interval between the beginning of the accident and the beginning of core melting is estimated to vary from 5 minutes to over one hour. The time period between accident initiation and the completion of core melting is governed by the accident class. For this study, complete core melting is assumed to occur within 10 to 30 minutes after the beginning of melting, and it is conservatively assumed that all severe accidents lead to complete core melting. The temperature of the core material when melting and downward relocation occurs is on the order of 900 K. This is below the temperature at which significant quantities of the volatile and semi-volatile radionuclides would be released from the fuel matrix. Core debris remains within the primary system until a failure in the primary system boundary occurs. Failure is most likely to occur when debris relocates into the coolant piping. Thus, failure of the primary system may occur in the piping or pumps rather than in the reactor vessel, itself. The reactor core is not sufficiently massive to cause failure of the bottom shield structure or of the coolant piping, so the bottom shield and/or coolant piping must reach a highly elevated temperature (1300 K or greater) before failure occurs. Studies indicate that the time required for the structures to heat to failure temperature is approximately 15 to 30 minutes after core debris begins to accumulate on the lower shield. During this time, the temperature of the core debris may exceed 1700 K. At these elevated temperatures, the release of volatile and semi-volatile radionuclides can be substantial. The debris heat-up phase of the accident progression would result in the most substantial release of volatile and semi-volatile radionuclides from the fuel that would occur during a severe accident.

After primary system failure, physical and chemical interactions between core debris and concrete (referred to as core-concrete interactions) may occur. Occurrence of these interactions depends on the rate of debris release from the reactor tank or piping, the amount of superheat in the debris, and the availability of water on the lower level floor of the reactor building. The greatest importance of core-concrete interactions is that they have the potential to create aerosols which could be transported to the filters and block air flow through the filters so that overheating would occur.

The Savannah River reactors use highly enriched uranium-aluminum fuel, so there is a possibility that transient nuclear criticality could be established during a severe accident. Studies indicate that the relocation of the core debris in the early stages of melting may result in "re-criticality" events in the reactor vessel when water is present. Re-criticality could alter or exacerbate the accident progression as a result of rapid heating of the debris or rapid steam generation. Re-criticality in the absence of water has been found to be highly improbable, and re-criticality of debris on the confinement floor is very unlikely.

The direct contact between a relatively cold volatile coolant (such as water) and a much hotter liquid (such as molten core debris) can potentially develop into a molten-fuel-coolant-interaction (MFCI), or a steam explosion. Rapid vaporization of the coolant as a result of a postulated MFCI could potentially exert significant loads on the reactor vessel or confinement.

Re-criticality and steam explosions are considered to be possible in the severe accident analyses which follow, and conservative estimates of the probability and consequences of such events are used. Since re-criticality in these reactors necessarily results in molten fuel being dispersed into saturated liquid the consequences are assumed to be identical to that of a steam explosion. The potential for large scale steam explosions is very unlikely unless re-criticality occurs.

4.3.3 Plant Damage State Development

Analysis of confinement system response to a severe accident requires a knowledge of the conditions under which fuel melting has occurred as opposed to a knowledge of the initiating event which caused the accident. The sequences from the core damage frequency analysis are, therefore, sorted into "bins" or "plant damage states" (PDSs) depending on the conditions at the time of melting. The criteria used to bin the sequences are: the amount of water in the reactor tank at the time of melting, the depth of water on the lower level floor of the reactor building, the availability of makeup water to supply the RRSS and the CHRS, and the availability of the AACS. These criteria were selected since operation of the AACS and the availability of water to scrub fission product releases from fuel debris have been shown to have the greatest effect on confinement response and radionuclide release to the environment. Since the AACS must be operating when the reactor is at power, for the internal events, it is assumed the AACS is always available at accident initiation. Each of the core damage sequences was evaluated to determine the appropriate PDS for that sequence.

The binning results in eight PDSs, four for the internally initiated sequences and four for the seismically induced sequences. The seismic sequences required separate PDSs because they include failure of the AACS. The definitions of these plant damage states are given in Table 4.3-1, and the logic of the binning process is shown by the event tree (called the bridge tree) shown in Figure 4.3-1. The frequency associated with each PDS is the sum of the frequencies of the sequences binned into that PDS. The frequencies are shown in Table 4.3-2.

4.3.3.1 Plant Damage State 1

This damage state is characterized by water in the reactor vessel at the onset of core damage and a minimum of one meter of water on the lower level floor. A "wet" reactor vessel and a "wet" confinement floor require that releases from the fuel at every stage of the accident pass through a significant quantity of water before reaching the confinement atmosphere. Thus, there is significant scrubbing of all radionuclides by the water. The AACS functions at nominal conditions at the onset of core damage. For this damage state, operation of the RRSS is not considered since it is assumed that any additional fission product retention provided by the sprays would be negligible in comparison to that afforded by water in the reactor vessel. Typical sequences in this damage state would include a LOPA initiated by flooding of the primary cooling system pump motors followed by a failure to actuate ECS, and a LOCA in which the ECS was activated but a mismatch between ECS addition rate and water removal rate that led to pump flooding.

4.3.3.2 Plant Damage State 2

This damage state is characterized by no water in the reactor vessel at the onset of core damage and less than one meter of water on the confinement floor. Make-up water to the CHRS is unavailable, as is the RRSS. A "dry" reactor vessel, a "dry" confinement floor, and the loss of water supply means that there will be minimal retention of fission product releases due to scrubbing. The AACS functions at nominal conditions at the onset of core damage. A typical sequence binned into this damage state would be a LOHSA initiated by a loss of river water supply.

4.3.3.3 Plant Damage State 3

This damage state is characterized by the reactor vessel being partially full of water, or having water supply from degraded ECS, at the onset of core damage. Further, there is less than one meter depth of water on the confinement floor. Since there is water in the reactor vessel, iodine removal from above-grade releases due to RRSS operation can be neglected. Operability of the CHRS make-up is also neglected since it is assumed that the eight-inch water level that this system would maintain will not provide significant scrubbing of below-grade fission product releases. A "wet" reactor vessel and a "dry" confinement floor means that there will be significant retention of in-vessel releases due to scrubbing but that there will be minimal retention of ex-vessel releases (e.g., from core debris interaction with concrete). The AACS functions at nominal conditions at the onset of core damage. A typical sequence in this damage state would be a LOCA in which only one ECS injection pathway is available and the confinement building sump pumps operate normally.

4.3.3.4 Plant Damage State 4

This damage state is characterized by no water in the reactor vessel at the onset of core damage, and more than 20 cm (8 inches) of water on the confinement floor. Water availability to the RRSS and to the CHRS is characteristic of this PDS. The water level considered by this damage state is, at a minimum, that which results from CHRS actuation. A "dry" reactor vessel, but some water on the confinement floor plus operability of the RRSS means that there will be limited scrubbing of releases both above and below grade. The AACS functions at nominal conditions at the onset of core damage. A typical sequence in this damage state would be a LOCA with complete failure of ECS but with water from the 25-million-gallon basin available to supply CHRS make-up and the RRSS.

4.3.3.5 Plant Damage State 5

This damage state is characterized by seismically induced structural failure leading to large openings in the AACS exhaust ducts from both above and below grade areas of the confinement. Similar to PDS 2, this damage state is characterized by no water in the reactor vessel at the onset of core damage and less than one meter of water on the lower level floor. The CHRS is unavailable, as is the RRSS. The AACS is totally unavailable. Structural failure of the confinement building, a dry reactor vessel, a dry confinement floor, and the loss of water supply, means that there will be minimal retention of fission product releases. A typical sequence in this damage state would include a LOHSA initiated by a seismic event of sufficient severity to cause that portion of the confinement building which supports the AACS to fail.

4.3.3.6 Plant Damage State 6

This damage state is characterized by seismically induced delatching of the filters and subsequent closure of the exhaust dampers. Thus, the filters, the fans, and the stack are isolated from the confinement building. Similar to PDS 2 and PDS 5, this damage state is characterized by no water in the reactor vessel at the onset of core damage and less than one meter of water on the confinement floor. Make-up water to the CHRS and the RRSS is unavailable. The AACS is totally unavailable, but the confinement building does not have large leakage pathways to the environment. Isolation of the AACS, a "dry" reactor vessel, a "dry" confinement floor, and the loss of water supply means that there will be minimal retention of fission product releases by water and that the filters are unavailable. Typical sequences in this damage state would include a LOHSA initiated by a seismic event that breaks cooling water lines outside the confinement building.

4.3.3.7 Plant Damage State 7

This damage state is characterized by seismically induced failure of the confinement building leading to large openings in the AACS exhaust ducts from both above and below grade areas of the confinement. Similar to PDS 1, this damage state is characterized by water in the reactor vessel at the onset of core damage and a minimum of one meter of water on the lower level floor. The AACS is totally unavailable. For this damage state operation of the RRSS is not considered since it is assumed that any additional fission product retention provided by the sprays would be negligible in comparison to that afforded by water in the reactor vessel and on the confinement floor. Structural failure of the confinement building, a "wet" reactor vessel, and a "wet" confinement floor means that there will be significant retention of fission product releases in water owing to scrubbing, but minimal retention by the confinement system. Typical sequences in this damage state would include a LOPA initiated by a seismic event of sufficient severity to cause that portion of the confinement building which supports the AACS to fail.

4.3.3.8 Plant Damage State 8

This damage state is characterized by seismically induced delatching of the filters and subsequent closure of the exhaust dampers. Thus, the filters, the fans, and the stack are isolated from the confinement building. Similar to PDS 1 and PDS 7, this damage state is characterized by water in the reactor vessel at the onset of core damage and a minimum of one meter of water on the lower level floor. For this damage state, operation of the RRSS is not considered since it is assumed that any additional fission product retention provided by the sprays would be negligible in comparison to that afforded by water in the reactor vessel and on the confinement floor. These conditions mean that there will be significant retention of fission product releases in water due to scrubbing and some retention by the confinement building. A typical sequence in this damage state would be a LOPA initiated by a seismic event that breaks cooling water lines inside the confinement building.

4.3.4 Confinement Event Tree Development

The analyses of the response of the confinement system to a severe accident were guided by means of confinement event trees (CETs). A CET is a logic model of confinement system response, and one CET has been constructed for each PDS. The CET top events represent the important phenomena which might occur in the confinement during a severe accident that can significantly alter the radionuclide release to the environment. Each path through the CET represents a possible accident progression sequence. Each sequence is assigned to a release category (RC) with an associated characteristic release.

The CETs are divided into four general time periods: in-vessel, at primary system failure, ex-vessel, and long term. For the in-vessel period, conditions within the reactor vessel are defined by the PDS definition. Two potential operator actions during the in-vessel period that may influence the subsequent progression of the accident are considered in the CET. These actions are the operation of the RRSS and initiation of the CHRS. The next time period of the accident encompasses failure of the primary system. Only the possibility of an energetic event (steam explosion or re–criticality induced steam explosion) that either causes or accompanies primary system failure is considered. The third time period encompasses the accident progression from the time of primary system failure up to the time at which core debris interactions with concrete would cease. Conditions in the confinement during this period are established by the PDS definition and the CET sequence up to this point. Four events, principally related to the performance of the confinement system filters, are considered. These are: the possibility that the high–efficiency particulate air (HEPA) filters have failed, the possibility that the core debris is quenched, the possibility of desorption of iodine from the charcoal filters, and the possibility of ignition of the charcoal filters. The final time period encompasses the interval between termination of core debris interaction with the system to the time at which recovery from the accident and removal of the filter compartments to contain the captured fission products could be accomplished. The probability of failure of the AACS is considered within this period.

Figures 4.3–2 through 4.3–9 present the CETs for PDS 1 through 8 respectively. The figure show the tree logic and the assigned branching ratios. The three columns to the right of the CETs show the conditional probability of each sequence, the release category to which the sequence is assigned, and the sequence number. Informed judgement was used where an explicit phenomenological basis did not exist for estimating the branching ratios, and the numerical values and justification for their use are shown in Table 4.3–3. The rationale for each of these values, or the complement of that value, is the same in every appearance of that value in the CETs.

4.3.4.1 Description of Top Events

The top events of the CETs represent phases of plant response to the damage state which can have a significant effect on the radionuclide release from the confinement. A description of each top event follows.

"RRSS Actuated" means that the operators turn on the spray system in the reactor room. Spray water will absorb a substantial fraction of the molecular iodine that is released to the reactor room thus reducing the iodine burden on the charcoal filters. The spray will also have a secondary effect of slightly reducing the temperature of air drawn into the filter compartments. The effect of spray on aerosol releases transported to the filters is small because of the large size of water droplets produced by the RRSS

"CHRS Actuated" means that the operators turn on the confinement heat removal system. This system dumps water from the disassembly basin to the lower level floor. A pool of water 20 cm (8 inches) deep is formed on the floor. The presence of water means that a steam explosion in the confinement may be possible following primary system breach. The CHRS will provide cooling of the core debris if the debris is sufficiently dispersed or fragmented. It will also cool gases that are released should core debris remain sufficiently hot to interact with concrete. Cooling of these gases may enhance the survivability of the filters and fans. Availability, or unavailability, of the make-up water supply is indicated by the PDS definition.

"No Steam Explosion" means that an energetic event does not accompany primary system failure. In this context, a steam explosion is taken to mean steam generation at a rate that may be sufficiently large to produce pressures in the confinement high enough to threaten the integrity of the HEPA filters. An in-vessel steam explosion considered by this event could be sufficiently large to cause failure of both the primary system and the filters. This event subsumes re-criticality events that might accompany relocation of molten fuel. Re-criticality might occur in the reactor vessel during fuel melting. The likelihood that re-criticality will occur on the confinement floor following breach of the primary system has been assessed to be small but cannot be completely eliminated. Re-criticality of the fuel material in the absence of water has been found to be highly improbable. The consideration of these distinct phenomena by this single event is made under the implicit assumption that the effects of a re-criticality event are essentially that of a steam explosion. This assumption stems from the judgement that a re-criticality would occur only in water and that the resultant heating of the fuel would result in its dispersal into the water, terminating the chain reaction and triggering a steam explosion.

"Filters Intact" means that the HEPA and charcoal filters are not functionally damaged and continue to remove aerosols and iodine vapor from the effluent air stream in accordance with their design. Success for this event also implies operation of the exhaust fans. Failure for this event implies mechanical failure of the HEPA filters with an accompanying release of some fraction of the captured fission product aerosols. Failure for this event is also indicated if the filters remain intact but the confinement building sustains structural damage that results in leakage so large as to prevent the exhaust fans from maintaining a negative pressure within the building. The charcoal filters and the exhaust fans are also assumed to fail. Since the charcoal would not be cooled following this event, complete desorption of iodine is assumed. This assumption is conservative if the building fails and the fans continue to operate. This latter combination has a low probability, so the resulting over-estimation of risk is judged not to be significant.

"Core Debris Cooled" means that the fuel and target debris generated during core melt is cooled to solidification within the primary system or on the confinement floor. Cooling within the primary system has been shown to be likely only if the debris flows into the primary system piping when that piping is submerged as a result of below-grade flooding. Cooling on the confinement floor will occur only if the debris spreads out to a thin layer on the confinement floor or is fragmented and dispersed (i.e., a steam explosion in the context of these CETs). Because of the presence of drainage trenches in the lower level floor, spreading to a thin layer is judged to be unlikely. Cooling of the debris eliminates ex-vessel aerosol generation and eliminates further fission product releases.

"No Iodine Desorption" means that the charcoal filters retain all the iodine that they absorb. This implies that the charcoal bed temperatures are maintained below 500 K (440°F). It also implies that aerosol capture by the HEPA filters is sufficiently low to allow adequate air flow to remove the heat generated in the charcoal beds.

"No Filter Burn" means that the charcoal filters do not ignite. This implies that the charcoal bed temperatures are maintained below 560 K (550°F). Failure for this event is assumed to imply that all fission products captured in the filter compartments are released. This assumption is conservative but considers the possibility that heat from charcoal combustion causes the HEPA filters to melt. Fission products trapped on the HEPA filters would revaporize either during the melting or as a consequence of the self-heating that occurred following the loss of cooling (it is assumed that the melted geometry would not be adequately cooled).

"Fans Operate" means that the building exhaust fans continue to operate for thirty days following the core melt. Fan failure is assumed to result in total desorption of the iodine on the charcoal bed since essentially all cooling would be lost.

4.3.5 Release Category Descriptions

Each CET sequence has been assigned to a release category (RC) outcome. A release category has a unique source term, and thus a consequence, associated with it. The frequency of the various release categories can be calculated as the sum of the frequencies of the accident progression sequences assigned to it. The frequency of an accident progression sequence is the product of the sequence conditional probability, developed by the CET analysis, and the frequency of the associated PDS. Risk is evaluated as the product of the release category consequences and their frequencies, summed over all release categories.

Twenty-six release categories have been defined. In general, a release category has been defined for each CET sequence that was expected to result in a source term that was clearly different from the others based on judgement prior to the actual source term calculations. The number of release categories has been kept to a minimum while maintaining adequate distinction between possible accident outcomes. The process by which each CET sequence was assigned to a release category required estimation of the source term for a sequence and subsequent grouping of that sequence with others for which the estimated source terms are similar. The primary means of performing this grouping was through similarity in sequences. Source term similarity was inferred and, where necessary, engineering judgement was employed. This is similar to the binning process performed for the core damage frequency analysis described in Section 4.3.4.

The characteristics of the various release categories are defined in Table 4.3-4. Mechanistic calculations have been used to estimate the source terms for some of the release categories. These primary release categories are designated RC-X, where X is a number. Source terms for secondary release categories that are similar to a primary release category in most respects have been evaluated assuming a simple perturbation from the mechanistic result. These secondary release categories are designated RC-Xy, where y is a lower-case letter. Each of these secondary release categories differs from the one upon which their evaluation was based (i.e., RC-X) in only one important aspect. Release timing or presence of a set of particular radionuclide groups are examples of differences. For example, RC-10 differs from RC-10a only in that RC-10a has no release of fission products due to the interaction of molten core debris with concrete. The release fractions of radionuclide groups associated with this interaction are manually deleted from the source term associated with RC-10 to form RC-10a.

4.3.5.1 Release Category 1

Release Category 1 provides the minimum source term to the environment. It is based on a postulated LOPA initiated by a guillotine break in a secondary-coolant effluent-header pipe. The assumed progression of the accident results in the reactor vessel being full of water and the lower level flooded at the time core damage begins. Thus, fission product releases from fuel debris at each stage of the accident are scrubbed by a significant depth of water before reaching the confinement atmosphere.

It is assumed that water from the broken cooling-water pipe floods the primary system pump motors approximately 150 seconds after the break occurs. Fuel melting begins 5 minutes after the break occurs. Consistent with thermal-hydraulic analyses of this accident performed with best-estimate codes, it is assumed that fuel melting occurs with the vessel filled with water. In the assumed scenario, thirty minutes after the cooling-water pipe break hot core debris melts through one of the primary system pipes. The liquid metal debris is assumed to flow down to the floor, down the trenches in the floor, and into the sump. The core debris then interacts with the concrete floor of the sump beneath a 3 meter depth of water. Fission product releases from the interaction are assumed to occur over a 90 minute time span.

4.3.5.2 Release Category 1a

Release Category 1a assumes the same accident progression as does RC-1 except that long-term failure of the confinement building exhaust fans is assumed. It is further assumed that this failure results in the desorption of all the iodine trapped in the charcoal beds. Consistent with the CET analysis for the sequences leading to this release category, fan failure could occur any time up to thirty days following the accident. To reduce the complexity of the consequence analysis, this iodine release is conservatively assumed to begin twelve hours into the accident and to be two hours in duration.

4.3.5.3 Release Category 2

Release Category 2 provides the maximum source term to the environment in the absence of charcoal filter ignition. It is based on a postulated LOHSA initiated by draining of the 25-million-gallon inlet water basin. With respect to accident timing, it is conservatively assumed that the accident begins (i.e., reactor shutdown occurs) at the time the basin drains. A more realistic assumption would be that reactor shutdown would occur at least 4 hours earlier. The accident on which this release category is based is thus assumed to begin with the loss of cooling water flow to the heat exchangers. Calculations show that it would require 80 minutes, following the loss of cooling, to boil-off the primary system coolant inventory. The steam from this boil-off is vented to the reactor room. Core damage is assumed to begin at that time. The reactor vessel is assumed to have been depleted of water, so fission product retention by the vessel is limited to that deposited (primarily by sedimentation) on the vessel interior surfaces. Fission products not deposited inside the vessel are assumed to escape into the reactor room.

It is assumed that the CHRS is not activated, so that when the core debris melts through the primary system piping at 110 minutes after basin drain, it flows onto a dry concrete floor. The molten debris is assumed to flow to the sumps and the molten core–concrete interaction (MCCI) begins. There is no scrubbing of the fission product releases associated with the MCCI since there is no overlying water. Fission products are assumed to be released from the core debris on the confinement floor for a period of 140 minutes. Throughout the scenario, the reactor vessel interior continues to heat and revolatilize a substantial fraction of the fission products that were deposited during the in–vessel phase of the accident. These revolatilized fission products are assumed to be released into the reactor room. These releases are also assumed to occur over a 140 minute period. Mechanistic predictions of the source term for this release category show significant thermal desorption of iodine from the charcoal filters late in the accident.

4.3.5.4 Release Category 2a

Release Category 2a assumes the same accident progression as does RC–2 except that long–term failure of the confinement building exhaust fans is included. It is further assume that this failure results in the desorption of all the iodine trapped in the charcoal filters. Consistent with the CET analysis for the sequences leading to this release category, fan failure could occur any time up to thirty days following the accident. To reduce the complexity of the consequence analysis, this iodine release is conservatively assumed to begin twelve hours into the accident and to be two hours in duration.

4.3.5.5 Release Category 2b

Release Category 2b assumes the same accident progression as RC–2 except that charcoal filter ignition is assumed to occur immediately after the release from the core debris is terminated (i.e., 250 minutes after the accident began). Based on the results of mechanistic calculations, charcoal filter ignition would occur only if more than 100 kg of aerosols were transported to and deposited on the HEPA filters. It is estimated, based on available charcoal combustion data, that the charcoal filters will burn for ten hours, and that combustion releases all the iodine that has been absorbed. In addition, it is conservatively assumed that all the volatile species on the HEPA filters are released (i.e., all captured CsI, CsOH, and TeO). The remaining fission product species are almost certainly deposited in their oxide form and only a relatively small fraction of these compounds are assumed to be released.

4.3.5.6 Release Category 3

Release Category 3 provides a source term to the confinement that is greater than that for RC–1 but significantly less than that for RC–2. It is based on a postulated LOCA, initiated by a primary coolant system pipe break. Flow to the vessel from one ECS pathway is assumed to enter the vessel, but this flow is assumed not to prevent a core melt. Core damage is assumed to begin 5 minutes after the pipe break occurs. The water injected into the vessel scrubs some of the fission products released from the melting fuel. Because the vessel is not full of water, as it is assumed to be in the accident that forms the basis of RC–1, less of the release is scrubbed. Corresponding to the most likely accident progression, RRSS operation is assumed. The source term to the confinement during core melting is thus greater than for RC–1 but less than for RC–2. As for RC–1, primary system failure as a result of debris melting through the primary system piping is assumed to occur 30 minutes into the accident. While there is likely to be water on the confinement floor for this accident (due to actuation of the CHRS plus whatever water has spilled from the vessel after the sump pumps are stopped), its presence is conservatively neglected in evaluating the source term for this release category. Core debris is assumed to flow across the floor, into the sumps, and MCCI begins, just as it does for RC–2. Fission product and aerosol releases are assumed to occur for 90 minutes (as they are for RC–1). Since ECS water is injected to the reactor vessel, revolatilization of any deposited fission products is assumed not to occur. Mechanistic predictions of the source term for this release category predict limited thermal desorption of iodine from the charcoal filters late in the accident.

4.3.5.7 Release Category 3a

Release Category 3a assumes the same accident progression as does RC–3. However for this release category, long–term failure of the confinement building exhaust fans is included. It is further assumed that this failure results in the desorption of all the iodine trapped in the charcoal filters. Consistent with the CET analysis for the sequences leading to this release category, fan failure could occur any time up to thirty days following the accident. To reduce the time required, and complexity of the consequence analysis, this iodine release is conservatively assumed to begin twenty-two hours into the accident and to be two hours in duration.

4.3.5.8 Release Category 4

Release Category 4 provides a source term to the confinement that is greater than that for RC-3 but less than that for RC-2. It is based on a postulated LOCA, initiated by a primary coolant system pipe break. Complete failure of the ECS is assumed. However, operation of both the RRSS and CHRS are assumed. The reactor vessel is assumed to be drained prior to the onset of core damage. Core damage is assumed to begin 5 minutes after the initiating event. There is no water in the vessel, as is true for RC-2, so releases to the reactor room are assumed to be the same. In the reactor room, however, spray water has a significant effect in absorbing iodine vapors. A small effect in washing out fission product aerosols is also expected. These effects are calculated mechanistically in developing a source term estimate for this release category. Vessel failure due to melt-through of the primary system piping is assumed to occur thirty minutes into the accident. While the depth of water assumed to be present in this scenario is no greater than that for RC-3, the source term calculation for this release category considers the presence of water on the confinement floor, unlike RC-3. Minimal scrubbing of the releases from the MCCI and some reduction in the temperature of the gases generated is assumed. As for RC-3, the interaction is assumed to proceed until 120 minutes after the initiating event. Revolatilization of fission products initially deposited in the reactor vessel is also considered. The treatment here is the same as that for RC-2. Iodine desorption from the filters is not predicted, owing to wash-out by the spray water and reduction in the temperature of gases drawn into the filter compartment.

4.3.5.9 Release Category 4a

Release Category 4a has the same accident progression as does RC-4. However, for this release category, long-term failure of the confinement building exhaust fans is included. It is further assumed that this failure results in the desorption of all the iodine trapped in the charcoal beds. Consistent with the CET analysis for the sequences leading to this release category, fan failure could occur any time up to thirty days following the accident. To reduce the complexity of the consequence analysis, this iodine release is conservatively assumed to begin twelve hours into the accident and to be two hours in duration.

4.3.5.10 Release Category 4b

Release Category 4b assumes the same accident progression as does RC–4. However, for this release category, charcoal filter ignition is assumed to occur immediately after the release from the core debris is terminated (i.e., 120 minutes after the accident began). Based on the results of mechanistic calculations, charcoal filter ignition would occur only if more than 100 kg of aerosols were transported to and deposited on the HEPA filters and the radioiodine loading on the charcoal beds was at a level comparable to that predicted for RC–2. It is estimated, based on available charcoal combustion data, that the charcoal filters will burn for ten hours, and that combustion releases all the iodine that has been absorbed. In addition, it is conservatively assumed that all the volatile species on the HEPA filters are released (i.e., all captured CsI, CsOH, and TeO). The remaining fission product species are almost certainly deposited in their oxide form and only a relatively small fraction of these compounds are assumed to be released.

4.3.5.11 Release Category 4c

Release Category 4c assumes the same accident progression as does RC–4. However, for this release category, it is assumed that a molten fuel–coolant interaction occurs immediately following vessel breach (i.e., 30 minutes after the initiating event). The core debris is assumed to be quenched on the confinement floor and continued supply of water from the CHRS is assumed to keep the debris cooled. As a result, there are no fission product releases from MCCI. All other aspects of this release category are the same as for RC–4. In producing the source term estimate for this release category only the fission product releases are neglected. Aerosol production (and filter plugging) and gas generation from concrete ablation are part of the source term estimate. The source term associated with this release category is conservative given the assumed accident progression.

4.3.5.12 Release Category 4d

Release Category 4d assumes the same accident progression as does RC–4c. However, for this release category, long–term failure of the confinement building exhaust fans is included. It is further assumed that this failure results in the desorption of all the iodine trapped in the charcoal beds. Consistent with the CET analysis for the sequences leading to this release category, fan failure could occur any time up to thirty days following the accident. To reduce the complexity of the consequence analysis, this iodine release is conservatively assumed to begin twenty-two hours into the accident and to be two hours in duration.

4.3.5.13 Release Category 5

Release Category 5 is associated with seismic events that lead to breach of the confinement building. It is based on a postulated LOHSA initiated by failure of the cooling water pipes between the 25-million-gallon basin and the confinement building. Therefore, the accident on which this release category is based is assumed to begin with the loss of cooling water flow to the heat exchangers. Calculations show that it would require 80 minutes, following the loss of cooling, to boil-off the primary system coolant inventory. The accident progression is identical to that assumed for RC-2 with the exception that the building is not actively ventilated. Rather, hot gases flow from the building through the failed ventilation pathways that led to the filters prior to the initiating seismic event. The gas flow is driven solely by heat generated by the core-melt accident.

4.3.5.14 Release Category 5a

Release Category 5a assumes the same accident progression as does RC-5. However, for this release category, it is assumed that the CHRS is actuated (but the make-up system is failed). Subsequently, a molten fuel-coolant interaction occurs immediately following vessel breach (i.e., 30 minutes after the initiating event). The core debris is assumed to be quenched on the confinement floor. Because there is no make-up to the water on the floor, this water will boil away and MCCI will begin. As a result, fission product releases are delayed. A ten hour delay (i.e., the time required to evaporate the water dumped by the CHRS) is assumed. All other aspects of this release category are the same as for RC-5. In formulating the source term for this release category, the gas sources associated with the MCCI are assumed to accompany both the release from revolatilization of fission products deposited in vessel and the delayed releases assumed for the interaction. This is conservative since the gas generation rate would be lower prior to the interaction and thus the revolatilization releases may be more effectively retained in the confinement.

4.3.5.15 Release Category 6

Release Category 6 is associated with seismic events that lead to delatching of the filters and subsequent closure of the filter isolation dampers. For this release category, all normal release pathways from the confinement building are closed and fission product releases occur through ex-filtration. The source term for this release category is based on postulated LOHSA initiated by failure of the cooling water pipes between the 25-million-gallon inlet water basin and the confinement building. The accident on which this release category is based is thus assumed to begin with the loss of cooling water flow to the heat exchangers. Calculations show that it would require 80 minutes, following the loss of cooling, to boil-off the primary system coolant inventory. The accident progression is identical to that assumed for RC-5 with the exception that the AACS is not vented to the atmosphere. Rather, gases leak from the building via normally available pathways. The gas flow is driven solely by heat generated by the core debris and the gases and heat generated by MCCI.

4.3.5.16 Release Category 6a

Release Category 6a assumes the same accident progression as does RC-6. However, for this release category, it is assumed that the CHRS is actuated (but the make-up system is failed). Subsequently, a molten fuel-coolant interaction occurs immediately following vessel breach (i.e., 30 minutes after the initiating event). The core debris is assumed to be quenched on the confinement floor. Because there is no make-up to the water on the floor, this water will boil away and MCCI is then assumed. As a result, fission product releases that occur from the MCCI are delayed. A ten hour delay (i.e., the time required to evaporate the water dumped by the CHRS) is assumed. All other aspects of this release category are the same as for RC-6. In formulating the source term for this release category, the gas sources associated with the MCCI are assumed to accompany both the release from revolatilization of fission products deposited in vessel and the delayed releases assumed for the interaction. This is conservative since the gas generation rate would be lower prior to the interaction, and the revolatilization releases may be more effectively retained in the confinement.

4.3.5.17 Release Category 7

Release Category 7 is associated with seismic events that lead to breach of the confinement building. It is based on a postulated LOPA initiated by failure of the secondary cooling water header within the building. Apart from damage to the confinement structure, the accident progression on which the source term calculation is based is similar to that for RC-1. The assumed progression of the accident results in the reactor vessel being full of water, and the lower level flooded, at the time core damage begins. Thus, fission product releases from fuel debris at each stage of the accident are scrubbed by a significant depth of water before reaching the confinement atmosphere. It is assumed that water from the broken cooling-water pipe floods the primary system pump motors approximately 150 seconds after the break occurs. Fuel melting begins 5 minutes after the break occurs. In the assumed scenario, thirty minutes after the cooling-water pipe break, hot core debris melts through one of the primary system pipes beneath the vessel. The liquid metal debris is assumed to flow down to the floor, down the trenches in the floor, and into the sumps. MCCI on the floor of the sump, beneath an assumed 3 m depth of water, is then modeled. Fission product releases from the interaction are assumed to occur over a 90 minute time span. Hot gases flow from the building through failed ventilation pathways that led to the filters prior to the initiating seismic event. The gas flow is driven solely by heat generated by the core debris and the heat and gases generated by MCCI.

4.3.5.18 Release Category 7a

Release Category 7a assumes the same accident progression as does RC-7. However, for this release category, it is assumed a molten fuel-coolant interaction occurs immediately following vessel breach (i.e., 30 minutes after the initiating event). The core debris is assumed to be quenched in the deep water pool on the confinement floor. As a result, fission product releases as a result of MCCI are prevented. All other aspects of this release category are the same as for RC-7.

4.3.5.19 Release Category 8

Release Category 8 is associated with seismic events that lead to delatching of the filters and subsequent closure of the filter isolation dampers. For this release category, all normal release pathways from the confinement building are closed and fission product releases occur through ex-filtration. The source term calculation for this release category is based on a postulated LOPA initiated by failure of the secondary cooling water header within the building. Apart from the isolation of the filters, the accident progression on which the source term calculation is based is the same as that for RC-7. The assumed progression of the accident results in the reactor vessel being full of water, and the lower level flooded, at the time core damage begins. Thus fission product releases from fuel debris at each stage of the accident are scrubbed by a significant depth of water before reaching the confinement atmosphere. It is assumed that water from the broken cooling-water pipe floods the primary system pump motors approximately 150 seconds after the break occurs. Fuel melting begins 5 minutes after the break occurs. Thirty minutes after the cooling-water pipe break hot core debris melts through one of the primary system pipes beneath the vessel. The liquid metal debris is assumed to flow down to the floor, down the trenches in the floor, and into the sumps. MCCI beneath a 3 m depth of water is then assumed. Fission product releases from the interaction are assumed to occur over a 90 minute time span.

4.3.5.20 Release Category 8a

Release Category 8a assumes the same accident progression as does RC-8. However, for this release category, it is assumed a molten fuel-coolant interaction occurs immediately following vessel breach (i.e., 30 minutes after the initiating event). The core debris is assumed to be quenched in the deep water pool on the confinement floor. As a result, fission product releases as a result of MCCI are prevented. All other aspects of this release category are the same as for RC-8.

4.3.5.21 Release Category 9

The assumed accident progression for RC-9 is similar to that used as the basis for the source term calculation for RC-1. The source term evaluation for this release category is based on a postulated LOPA initiated by failure of a secondary-coolant effluent-header pipe. The reactor vessel is assumed to be full of water, and the lower level is flooded at the time core damage begins. Thus, fission product releases from fuel debris at each stage of the accident are scrubbed by a significant depth of water before reaching the confinement atmosphere. Fuel melting begins 5 minutes after the pipe break occurs. In this scenario, thirty minutes after the break, hot core debris melts through one of the primary system pipes beneath the vessel and results in a steam explosion. The resulting pressure spike is assumed to cause structural failure of the confinement filters. This failure is assumed to release 50% of the aerosol mass deposited on the HEPA filters at that time. As a result of losing the fan-forced cooling through the charcoal beds, thermal desorption of the iodine inventory occurs over a two hour period. It is also assumed that the core debris collects in the lower level sumps, re-heats, and interacts with the concrete floor of the sump. Fission product releases from the MCCI are assumed to occur over a 90 minute time span.

4.3.5.22 Release Category 9a

Release Category 9a assumes the same accident progression as does RC-9. However, it is assumed that the steam explosion results in complete dispersal and quenching of the core debris. This prevents subsequent debris re-heating and fission product releases as a result of the core debris interacting with concrete. All other aspects of this release category are the same as for RC-9. Thus, by including the gases and sensible heat released during the MCCI, the source term for this release category is conservatively defined.

4.3.5.23 Release Category 10

Release Category 10 provides the maximum source term to the environment with the AACS operating for accident sequences involving a steam explosion. The source term estimate is based on a postulated LOHSA initiated by draining of the 25-million-gallon basin, similar to that of RC-2. The accident on which this release category is based begins with the loss of cooling water flow to the heat exchangers. It requires 80 minutes, following the loss of cooling, to boil-off the primary system coolant inventory. The resultant steam is vented to the reactor room. Core damage is assumed to begin at that time. Fission product retention by the vessel is limited to that deposited (primarily by sedimentation) on the vessel interior surfaces. Fission products not deposited inside the vessel are assumed to escape into the reactor room. For this release category, it is assumed that the CHRS is activated, so that when the core debris melts through the primary system piping (110 minutes after basin drain), it comes into contact with water resulting in a steam explosion. The pressure increase is assumed to structurally fail the confinement filters. This failure is assumed to release 50% of the aerosol mass deposited on the HEPA filters at that time. As a result of losing the fan-forced cooling through the charcoal beds, thermal desorption of the iodine occurs over a two hour period. The core debris collects in the lower level sumps and is assumed to reheat resulting in MCCI. Fission product releases from the MCCI are assumed to occur over a 90 minute time span.

4.3.5.24 Release Category 10a

Release Category 10a assumes the same accident progression as does RC-10. However, it is assumed that the steam explosion results in complete dispersal and quenching of the core debris. This prevents subsequent debris re-heating and prevents fission product releases as a result of the core debris interacting with concrete. All other aspects of this release category are the same as for RC-10. Thus, by including the gases released during the MCCI, and their sensible heat, the source term for this release category is conservatively defined.

4.3.5.25 Release Category 11

Release Category 11 provides a source term to the environment that is intermediate between RC-10 and RC-9. As for those release categories, AACS is assumed to operate initially and a steam explosion is assumed. The source term estimate is based on postulated LOCA, initiated by a primary coolant system pipe break. Complete failure of the ECS is assumed. However, operation of both the RRSS and CHRS are assumed. The accident progression used in the source term analysis is similar to that for RC-4. The reactor vessel is assumed to be drained prior to the onset of core damage. Core damage is assumed to begin 5 minutes after the initiating event. In the reactor room, spray water has a significant effect in absorbing iodine vapors, and a small effect in washing out fission product aerosols. Primary system failure due to molten core debris melting through the below-grade primary system piping occurs thirty minutes into the accident resulting in a steam explosion. The resulting pressure spike is assumed to fail the confinement filters. This mechanical failure is assumed to release 50% of the aerosol mass deposited on the HEPA filters at that time. As a result of losing the fan-forced cooling through the charcoal beds, thermal desorption of the iodine occurs over a two hour period. The core debris collects in the lower level sumps and is assumed to re-heat resulting in MCCI. Fission product releases from the MCCI are assumed to occur over a 90 minute time span.

4.3.5.26 Release Category 11a

Release Category 11a assumes the same accident progression as does RC-11. However, it is assumed that the steam explosion results in complete dispersal and quenching of the core debris. This prevents subsequent debris re-heating and fission product releases as a result of the core debris interacting with concrete. All other aspects of this release category are the same as for RC-11. Thus, by including the gases released during the MCCI, and their sensible heat, the source term for this release category is conservatively defined.

4.3.6 Source Term Calculations

For each release category (RC) identified, a source term has been estimated to form the basis of the consequence calculations. The assumptions and calculational techniques employed to produce source terms for each of the twenty-six release categories are described in this section. Mechanistic calculations were performed to estimate source terms for nine of the primary release categories. Source terms for the remaining release categories have been evaluated using engineering judgement.

Mechanistic calculations for the primary release categories were performed with the CONTAIN computer code. CONTAIN is a best-estimate control-volume thermal-hydraulics code that models one-dimensional single-phase flow. Multiple flow components, including water, steam, and a variety of gas mixtures, are considered. The code incorporates models for various phenomena that may occur during severe reactor accidents. The phenomena represented that are relevant to the current analysis include; hydrogen combustion, aerosol transport and deposition, fission product vapor dissolution in sprays and transport, and aerosol and fission product vapor capture by filters. CONTAIN is strictly limited to modeling physical processes that occur outside the reactor primary system.

The version used in this analysis, CONTAIN/SR, has been modified at Savannah River Laboratory, in cooperation with Sandia National Laboratories, to incorporate models for the fans and filter compartments used on Savannah River reactor confinements. The objectives of the analyses were to calculate fission product releases to the environment for the eleven primary release categories. This analysis required that the fission products released from the core debris to the confinement be developed and specified as part of the CONTAIN input. In addition to the fission product source term, water levels and gas releases were also specified.

The first four primary release categories correspond to: (1) LOPA due to a secondary cooling water line break; (2) LOHSA due to a basin drain; (3) primary system pipe break with degraded ECS addition; and (4) LOCA with no ECS. Release categories 5 and 6 are for seismically induced LOHSAs, similar to RC-2, whereas RC-7 and RC-8 are seismically induced LOPAs, similar to RC-1. RC-5 may be described as a LOHSA with concurrent stack building failure, while RC-6 is a LOHSA where the confinement filters delatch and the exhaust dampers close. The AACS remains intact with filter/fan isolation. Case 7 is a seismic loss of pumping scenario due to an internal cooling water line break with concurrent stack building failure. The eighth release category is a LOPA with minor seismic damage

Again, the AACS remains intact with filter/fan isolation. Three final release categories (9–11) superimpose steam explosions on LOPA, LOHSA, and LOCA scenarios. Source terms from the reactor accidents described above are calculated based on mechanistic computer models and engineering judgment.

Three mechanisms for the release of fission products from the reactor fuel to the confinement system were considered in developing the source term to the confinement. These are: (1) release from fuel in the reactor vessel, (2) release from the molten core debris when it interacts with the concrete confinement floor, and (3) release from molten fuel that participates in a molten fuel–coolant interaction. The release from the fuel debris does not define the release to the confinement because releases within the reactor vessel will be scrubbed if they must pass through water before reaching the confinement. In addition, the in–vessel releases may deposit on surfaces within the primary system. Releases from the molten core–concrete interaction (MCCI) may be scrubbed by overlying water pools. These mechanisms that reduce the confinement source term are considered for each release category.

4.3.6.1 In–Vessel Releases

In–vessel releases are assumed to begin when the fuel tube temperatures reach approximately 670 K. At this temperature, tritium is fairly mobile in aluminum and diffusion out of the assembly target tubes would be expected. Fuel damage is defined to occur when the highest fuel tube temperature reaches 920 K. This is approximately the solidus temperature of the fuel and cladding material, and grain boundary melting would occur at this temperature so that releases of fission gases from the fuel material would begin at this time. In–vessel releases are defined to end either when primary system failure allows fuel debris to escape from the vessel, or when the debris is cooled and re–solidified within the primary system. In calculating source terms for each of the release categories, it is assumed that the primary system fails when molten debris melts through a primary system pipe in the pump room. This has been assessed to be the most likely failure mode. It also leads to the highest in–vessel releases since the debris temperature must reach essentially the melting temperature of stainless steel (1700 K).

Based on thermal–hydraulic analyses, in–vessel releases begin approximately 5 minutes after cooling flow to the assemblies is degraded. (The time at which coolant flow to the fuel assemblies degrades is defined as the start of the accident in this analysis.) Melting of the fuel assemblies requires approximately 5 minutes. At this time the fuel is assumed to reside on the bottom of the reactor vessel as a pool of slurry (a mixture of liquid and particulate fuel debris). Experimental evidence, combined with the results of fuel assembly melting calculations, indicates that significant release of the volatile fission product elements does not occur until this time. Heat-up of the debris on the vessel bottom would proceed until the debris temperature reaches 1000 K to 1200 K, at which time it would flow through the vessel coolant outlet nozzles and into the below-grade piping. This migration of the molten debris is assumed not to impact fission product releases or retention of released materials by the primary system. It is estimated that melt-through of the primary system would require approximately 20 minutes following formation of the debris pool on the reactor vessel bottom. Therefore, primary system failure, and thus the cessation of in-vessel releases is assumed to occur 30 minutes after the start of the accident.

Fission product gas releases are assumed to occur during the five minute period from the start of fuel damage until the formation of the debris pool. These fission product gases are assumed to accumulate on grain boundaries within the fuel matrix during the irradiation process, and would form into bubbles as the fuel starts to melt. Consequently, the fuel material is expected to foam as it attains its melting temperature. This foaming is assumed to release all the fission gases. Volatile fission products (e.g., I, Cs, and Te) are not released in significant quantities during this period. The majority of volatile fission products will be released when the fuel debris is held at temperatures exceeding 1200 K for a few minutes. It is therefore assumed that all the volatile fission products are released during the 20 minute period between fuel assembly melting and primary system failure. Strontium, samarium, europium, and barium are fission product elements in the debris that exert significant vapor pressure at 1700 K (the maximum temperature that the debris is assumed to attain). It is assumed that the largely metallic melt in which the fission product elements are dissolved will not act to reduce the vapor pressure. Thus, these elements are assumed to be released during the in-vessel phase in approximate proportion to their vapor pressure at 1700 K. Table 4.3-5 lists the fraction of the initial core inventory assumed to be released from the fuel.

Two types of in-vessel fission product retention are possible for the accident scenarios considered. The first is retention of fission products on primary system surfaces. Two general processes contribute to this phenomena. The first is condensation from the vapor phase onto cool surfaces (temperature less than the fission product boiling point). The second is aerosol deposition, which is dominated by sedimentation (gravitational settling).

A second type of retention within the reactor tank is pool-scrubbing afforded by the presence of water. The amount of water present will depend on the particular accident sequence. Full reactor tank sequences, e.g. LOPA scenarios, imply that water: (1) surrounds fuel and target assemblies during the assembly heat-up and melting process; and (2) overlies the debris bed once formed in the bottom of the reactor tank. These sequences will yield the greatest retention of both fission product and aerosol species. Partially filled reactor tank sequences would be much less effective, since a depth of several feet of water would be available rather than fourteen feet in full-tank sequence. Late actuation of the ECS would be illustrative of the partial tank water depth.

Fission products and aerosols emerging from slumped fuel material would travel through overlying water levels carried by water vapor and gases. It is likely that bubbles would form and rise through the water depth. The primary retention mechanisms that are active as the bubbles transit the pool include: (1) sedimentation of aerosols within the bubbles, (2) impaction of particles on the bubble walls, and (3) diffusion of particles to the walls. The greater water depths allow these mechanisms to be active longer. Once reaching the water surface in the tank, the bubbles break up and release remaining contents. At this point, fission products and aerosols are assumed to enter the process room. Although additional retention may occur upon passage through various tank top penetrations, e.g., vacuum breakers, it is not considered.

To assess the efficiency of the ability of a process, an area or surface, or an engineering system to remove fission products, the decontamination factor (DF) concept is useful. It is defined as the ratio of the species mass entering a system/process to the mass leaving the system/process. Table 4.3.6 shows the in-vessel DFs assumed for fission products in: (1) a filled reactor vessel; (2) late ECS (several feet of water); (3) dry reactor vessel (condensation and sedimentation); and (4) a dry vessel, with condensation, sedimentation, and chemisorption active. The last mechanism accounts for vessel stainless steel surfaces retaining tellurium oxides. The DFs shown are best estimates, based on in-vessel mechanistic code calculations, commercial plant studies, and engineering judgment.

For the primary release categories assuming a dry vessel, (RC-2, -5, -6, and -10), a post-melt through period has been added for evolution of fission products to the process room. For these cases, condensed species (CsI, CsOH, and TeO) revolatilize upon reactor vessel heating in a dry condition. It is assumed that the release is only partially effective for CsOH- and TeO- class species. The fraction of the initially deposited CsOH and TeO released is 0.5 and 0.2, respectively, due to chemisorption reduction.

4.3.6.2 Ex-Vessel Releases

Fission products not released in-vessel are assumed to be retained by the core debris upon primary coolant system (PCS) failure. Two types of fission product releases are assumed to occur outside of the primary coolant system: molten core-concrete interactions (MCCI), and molten fuel-coolant interactions (MFCI, or steam explosions). Both releases are assumed to occur below grade in one of the pump rooms. The MCCI processes are assumed to start without delay time for debris travel and re-accumulation in one of the pump room sumps. MFCI fission product releases occur instantaneously upon breach of the PCS. A thirty-minute period is assumed in release categories 9, 10, 11 from the steam explosion until the initialization of MCCI processes. This is an estimate of the time necessary for reaccumulation of molten core debris as a pool in one of the sumps.

4.3.6.2.1 Molten Core Concrete Interaction

MCCI occurs when core debris accumulates in the pump room sump after breaching the primary system coolant piping (approximately 30 minutes after the start of the accident). Aerosols are created as the core debris reacts with the concrete and fission products hosted by the aerosols become airborne. Current experimental and analytical studies indicate that the melt temperature does not exceed 1700 K during the attack on concrete. The concrete is silicious and, therefore, yields negligible carbon monoxide. Only steam, hydrogen and carbon dioxide evolution are important.

The LWR version of the CORCON code has been modified to treat uranium-aluminum fuel and SRS concrete. The new code, designated CORCON/SR, was used to predict hydrogen gas and carbon dioxide evolution rates during the MCCI for each release category. These data were then specified directly as sources in each of the calculations performed with CONTAIN/SR.

Mechanistic code predictions for the aerosol generation and fission product evolution from the MCCI are not available. These sources are instead estimated. For the fission products, the maximum melt temperature of 1700 K limits the specific groups that can be released. The strontium, europium/samarium and barium fission product groups are released relative to their boiling point and vapor pressure data for this temperature regime. The noble gas halogens, cesium, and tellurium group are assumed to have been completely released during the in-vessel portion of the sequence (Table 4.3-7).

The MCCI process typically lasts over the course of several to tens of hours, depending on the core debris pool configuration and initial conditions. For this study, releases are compressed to one hour and two hours for wet floor and dry floor conditions, respectively. This is expected to approximate the actual process in which most of the fission products are released early in the vigorous stages of MCCI. In addition to release of radioactive and stable aerosol masses of the Sr, Eu/Sm, and Ba groups, 100 kg of concrete aerosols are released over the same one to two-hour period.

An overlying water layer is present over the melt debris for several of the primary release categories due to the initiating event, the CHRS, and/or make-up water. The height is variable and may extend from 20 cm (CHRS only) to 3 m (assumed for LOPA). The same scrubbing processes would be active as discussed for the in-vessel phase. For the specific primary release categories analyzed with CONTAIN/SR, the overall pool DF was kept conservatively low through selection of aerosol input parameters. This resulted in pool DFs of 1.1 to 3.0 rather than expected values of 1.2 to 10.0, for 20 cm to 3 m sump water levels. The net effect will be that the CONTAIN analyses will introduce more aerosol mass for transport through the AACS than would be predicted by engineering judgement supplemented by significant experimental evidence. Accordingly, the fission product release to the confinement is higher than expected.

4.3.6.2.2 Molten Fuel Coolant Interaction

Steam explosions may occur within the reactor vessel, in the primary system piping, or in the below grade confinement (pump room) However, the release of fission products to the AACS is assumed to always occur below grade in one of the pump rooms. Three primary release categories were considered with a common MFCI fission product source term occurring upon primary coolant system failure. For RC-9, RC-10, and RC-11 cases, the steam explosion occurs at 29.9 minutes, 109.9 minutes, and 29.9 minutes, respectively. The fission product inventory introduced into the AACS is due to effects of the explosion. Mechanical and thermal agitation of the molten debris is assumed to introduce 1.5% of the Sr group, and 1% of the other fission products and capture species. These releases are shown in Table 4.3-7 and include low volatility groups headed by ruthenium, lanthanum, and cerium species. The period of release to the confinement is 6 seconds (0.1 minute). An additional amount of 100 kg of aluminum aerosol is introduced during the same period.

4.3.6.3 Primary Release Category Input To CONTAIN

The source term to the confinement for each of the primary release categories is shown in Table 4.3-8. CONTAIN/SR used the fission product, aerosol, and other material sources as input to analyze the response of the confinement system and to establish the source term to the environment. Release categories 9 and 11 were not mechanistically calculated, but instead were based on Case 1 and Case 4 analyses with the effect of the steam explosion added based on RC-10 analyses.

The fission product masses for as many as ten groups were tracked in the CONTAIN/SR analyses for each of the nine cases. The ten groups and component elements are shown in Table 4.4-1.

4.3.6.4 MACCS Input Source Terms

In interpreting CONTAIN/SR results, the appearance of fission product groups in three environmental cells (stack, ground, and roof) indicated release from the AACS into one of three possible elevation bins. The masses released are normalized by the original radioactive mass present in the core inventory to derive the release fractions. Four plume segments are utilized to characterize the environmental release timing. The final set of twenty-six source terms developed for the consequence analysis are also shown by MACCS release groups in Table 4.3–8.

Tritium as a MACCS release group is assumed to transport in the AACS with the characteristics of a noble gas. This is conservative and will over-predict release rates to the environment since much of the tritium will be retained in water pools throughout the reactor and stack building as tritiated water. The sensible heat in a plume segment is added manually. It may be calculated once the mass flow rate to the environmental cell, the heat capacity of the gas exiting the confinement system, and the gas temperature are known for a specific time period.

Table 4.3-1

Definition Of Plant Damage States

Plant Damage State	Definition
1	Reactor Tank Flooded At Core Melt Lower Level Flooded Availability of Make-up Water Unimportant AACS Operating
2	Reactor Tank Dry At Core Melt Lower Level Dry Make-up Water Unavailable AACS Operating
3	Reactor Tank Flooded At Core Melt Lower Level Dry Availability of Make-up Water Unimportant AACS Operating
4	Reactor Tank Dry At Core Melt Lower Level Partially Flooded Make-up Water Available At Core Melt AACS Operating
5	Reactor Tank Dry At Core Melt Lower Level Dry Make-up Water Not Available AACS Ducts Open to Environment
6	Reactor Tank Dry At Core Melt Lower Level Dry Make-up Water Not Available AACS Isolated
7	Reactor Tank Flooded At Core Melt Lower Level Flooded Availability of Make-up Water Unimportant AACS Ducts Open to Environment
8	Reactor Tank Flooded At Core Melt Lower Level Flooded Availability of Make-up Water Unimportant AACS Isolated

Table 4.3-2

Core Damage Frequency by Plant Damage State

Plant Damage State	Core Damage Frequency (per Reactor Year)
PDS 1	2.9×10^{-5}
PDS 2	1.3×10^{-5}
PDS 3	6.7×10^{-5}
PDS 4	1.9×10^{-5}
PDS 5	2.9×10^{-6}
PDS 6	3.2×10^{-5}
PDS 7	2.8×10^{-6}
PDS 8	3.1×10^{-5}
TOTAL	2.0×10^{-4}

Table 4.3-3

Criteria for Assignment of Branch Point Probabilities

Probability	Descriptive Criterion
1.0	The branch point probability is known based on the definition of the accident sequence or fundamental physical/chemical principles. To avoid confusing complexity in the CETs, branching is omitted in these cases.
0.99	The branch point probability is very certain, based on analyses and/or physical and chemical principles.
0.90	The branch point probability is likely, based on analyses and/or physical and chemical principles. However, some uncertainty exists which should be considered in the assessment.
0.50	The branch point probability cannot be determined on the basis of analyses, and/or physical or chemical principles. Further study beyond this scoping assessment is indicated.
0.3	This probability is applied to specific cases in which the solidification of the core debris is in question. It is derived from the probabilities of other cases.
0.057	The probability the operator fails to actuate CHRS.
8.65×10^{-3}	The probability of failure of the AACS fans during a 30-day mission time. It includes the fan improvements to be made before restart.
1.0×10^{-3}	The probability of a filter burn for specific sequences in which analyses indicate that ignition is very unlikely.

Table 4.3–4

Summary of Release Category Characteristics

Release Category	Confinement Status	In-Vessel Release	Intermediate Filter Status	Ex-Vessel Releases	Filter Heat-Up
1	AACS On	Deep Water	Intact	Deep Water	None
1a	AACS On	Deep Water	Intact	Deep Water	Delayed Desorption
2	AACS On	Dry	Intact	Dry	Desorption
2a	AACS On	Dry	Intact	Dry	Delayed Burn
2b	AACS On	Dry	Intact	Dry	Burn
3	AACS On	Some Water	Intact	Dry	Small Desorption
3a	AACS On	Some Water	Intact	Dry	Delayed Desorption
4	AACS On	Spray Only	Intact	Some Water	Moderate Desorption
4a	AACS On	Spray Only	Intact	Some Water	Delayed Desorption
4b	AACS On	Spray Only	Intact	Some Water	Burn
4c	AACS On	Spray Only	Intact	None	None
4d	AACS On	Spray Only	Intact	None	Delayed Desorption
5	Failed	Dry	N. A.	Dry	N. A.
5a	Failed	Dry	N. A.	Dry	N. A.
6	Isolated	Dry	N. A.	Dry	N. A.
6a	Isolated	Dry	N. A.	Dry	N. A.
7	Failed	Deep Water	N. A.	Deep Water	N. A.
7a	Failed	Deep Water	N. A.	Deep Water	N. A.
8	Isolated	Deep Water	N. A.	Deep Water	N. A.
8a	Isolated	Deep Water	N. A.	Deep Water	N. A.
9	AACS On	Deep Water	Failed	Deep Water	Complete Desorption
9a	AACS On	Deep Water	Failed	Deep Water	Complete Desorption
10	AACS On	Dry	Failed	Dry	Complete Desorption
10a	AACS On	Dry	Failed	Dry	Complete Desorption
11	AACS On	Spray Only	Failed	Some Water	Complete Desorption
11a	AACS On	Spray Only	Failed	Some Water	Complete Desorption

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Table 4.3–5

Fission Product Releases In–Vessel

Time Period (min)	Fraction of Core Inventory Released by Fission Product Group						
	NG	1.0	1.0	1.0	0.4	0.2	0.13
0–5	0.0	0.0	0.0	0.0	0.0	0.0	0.0
5–10	1.0	0.0	0.0	0.0	0.0	0.0	0.0
10–30	0.0	1.0	1.0	1.0	0.4	0.2	0.13

Table 4.3–6

In–Vessel Decontamination Factors

Process/Mechanism	DF
Deep Water, In–Vessel	20
Late ECS, In–Vessel	4
Dry Vessel	2
Dry Vessel w/Chemisorption	4

Table 4.3-7

Fission Products Released Ex-Vessel

Molten Core – Concrete Interactions

Release Category	Time Period (min.)	Fraction of Inventory Released by Fission Product Group: Sr	Eu/Sm	Barium
1,3,4,7,8	30 - 120	0.50	0.20	0.13
2,5,6	110 - 250	0.50	0.20	0.13
9,11	60 - 120	0.50	0.20	0.13
10	140 - 250	0.50	0.20	0.13

Molten Fuel – Coolant Interactions

Release Category	Time Period (min.)	Halogens	Cesium	Te	Sr	Eu/Sm	Ba/Ru/La/Ce
		Fraction of Inventory Released by Fission Product Group					
9,11	29.9 - 30.0	0.01	0.01	0.01	0.015	0.01	0.01
10	109.9 - 110.0	0.01	0.01	0.01	0.015	0.01	0.01

Table 4.3–8, Sheet 1

Release Category Fission Product and Material Sources to the Confinement

PRIMARY RELEASE CATEGORY	COMMENTS	TIME INTERVAL (minutes)	RELEASE LOCATION[a]	FISSION PRODUCT SOURCES[b] (% OF CORE INVENTORY) Noble Gases	Molec. Iodine	I as CsI	Cs as CsOH	Te as TeO	Sr as SrO	Eu/Sm as Oxides	Ba as BaO	Ru/La/Ce as Oxides	Aerosol Size Mean[d] (μm)	SATURATED STEAM (kg)	HYDROGEN[c] (kg)	MCCI AEROSOLS (kg)
1 LOPA DUE TO SECONDARY LOCA																
		5 - 10	ABOVE GRADE	100	-	-	-	-	-	-	-	-	-	-	-	-
	DEBRIS MELT THRU AT 30 MINUTES	5 - 30	ABOVE GRADE	-	1	5	5	5	25	1	1	-	0.2	3000	-	-
	1900 K DEBRIS WATER	30 - 120	BELOW GRADE	-	-	-	-	-	50	20	13	-	1.0		-	-
		30 - 180	BELOW GRADE	-	-	-	-	-	-	-	-	-	-	-	147	100
2 LOSHA DUE TO BASIN DRAIN																
	VESSEL BOIL OFF PERIOD	5 - 80	ABOVE GRADE	-	-	-	-	-	-	-	-	-	-	45000	-	-
		80 - 85	ABOVE GRADE	100	-	-	-	-	-	-	-	-	-	-	-	-
	DEBRIS MELT THRU AT 110 MINUTES	80 - 110	ABOVE GRADE	-	50	25	50	50	25	10	6	-	2.0	300	-	-
		110 - 250	ABOVE GRADE	-	-	25	25	10	-	-	-	-	1.0	-	-	-
		110 - 250	BELOW GRADE	-	-	-	-	-	50	20	13	-	1.0	-	170	100
3 PRIMARY SYSTEM BREAK WITH DEGRADED ECS																
		5 - 10	ABOVE GRADE	100	-	-	-	-	-	-	-	-	-	3000	-	-
	DEBRIS MELT THRU AT 30 MINUTES	5 - 30	ABOVE GRADE	-	25	13	25	13	13	5	3	-	1.0	10000	-	-
	1900 K DEBRIS WATER	30 - 120	BELOW GRADE	-	-	-	-	-	50	20	25	-	1.0	-	151	100
4 LOSS OF COOLANT ACCIDENT																
		5 - 10	ABOVE GRADE	-	-	-	-	-	-	-	-	-	-	3000	-	-
		10 - 15	ABOVE GRADE	100	-	-	-	-	-	-	-	-	-	-	-	-
	DEBRIS MELT THRU AT 30 MINUTES	10 - 30	ABOVE GRADE	-	50	25	50	50	25	10	6	-	20	300	-	-
	1900 K DEBRIS WATER	30 - 120	BELOW GRADE	-	-	25	25	10	-	-	-	-	-	-	-	-
		30 - 120	BELOW GRADE	-	-	-	-	-	50	20	13	-	1.0	-	147	100

a Above Grade = Above Grade Release to the Process Room, Below Grade = Below Grade Release to the Pump Room
b Percentage of Fission Product Groups Released Relative to Shutdown Condition :
c Hydrogen Released from MCCI
d Log Normal Size Distribution

GROUPS	ELEMENTS
Noble Gases	Xe, Kr
Iodine	I, Br
Cesium	Cs, Rb
Te	Te, Sb, Se
Sr	Sr
Ru	Ru, Rh, Pd Mo, Tc
La	La, Zr, Nd, Nb, Pm, Pr, Y
Eu	Eu, Sm
Ce	Ce Pu Np

Table 4.3-8, Sheet 2

Release Category Fission Product and Material Sources to the Confinement

PRIMARY RELEASE CATEGORY	COMMENTS	TIME INTERVAL (minutes)	RELEASE LOCATION[a]	FISSION PRODUCT SOURCES[b] (% OF CORE INVENTORY) Noble Gases	Molec. Iodine	I as CsI	Cs as CsOH	Te as TeO	Sr as SrO	Eu/Sm as Oxides	Ba as BaO	Ru/La/Ce as Oxides	Aerosol Size Mean[d] (μm)	SATURATED STEAM (kg)	HYDROGEN[c] (kg)	MCCI AEROSOLS (kg)
5 SEISMIC LOSHA DUE TO EXTERNAL COOLING LINE BREAK	STACK BUILDING FAILURE															
	VESSEL BOIL OFF PERIOD	5 - 80	ABOVE GRADE	.	.	.	.	.	.	.	.	.	.	.	.	.
		80 - 85	ABOVE GRADE	100	.	.	.	.	.	.	.	.	.	45000	.	.
	DEBRIS MELT THRU AT 110 MINUTES	80 - 110	ABOVE GRADE	.	50	25	50	50	25	10	6	.	2.0	300	.	.
		110 - 250	ABOVE GRADE	.	.	25	25	10	.	.	.	.	1.0	.	.	.
		110 - 250	BELOW GRADE	.	.	.	.	.	50	20	13	.	1.0	.	170	100
6 SEISMIC LOSHA DUE TO EXTERNAL COOLING LINE BREAK	FAN/FILTER ISOLATION															
	VESSEL BOIL OFF PERIOD	5 - 80	ABOVE GRADE	.	.	.	.	.	.	.	.	.	.	.	.	.
		80 - 85	ABOVE GRADE	100	.	.	.	.	.	.	.	.	.	45000	.	.
	DEBRIS MELT THRU AT 110 MINUTES	80 - 110	ABOVE GRADE	.	50	25	50	50	25	10	6	.	2.0	300	.	.
		110 - 250	ABOVE GRADE	.	.	25	25	10	.	.	.	.	1.0	.	.	.
		110 - 250	BELOW GRADE	.	.	.	.	.	50	20	13	.	1.0	.	170	100
7 SEISMIC LOPA DUE TO INTERNAL COOLING LINE BREAK	STACK BUILDING FAILURE															
		5 - 10	ABOVE GRADE	100	.	.	.	.	.	.	.	.				
	DEBRIS MELT THRU AT 30 MINUTES	5 - 30	ABOVE GRADE	.	1	5	5	5	25	1	1	.	0.2	3000	.	.
	1900 K DEBRIS WATER	30 - 120	BELOW GRADE	.	.	.	.	.	50	20	13	.	1.0		.	.
		30 - 180	BELOW GRADE	.	.	.	.	.	.	.	.	.	.	.	147	100
8 SEISMIC LOPA DUE TO INTERNAL COOLING LINE BREAK	FAN/FILTER ISOLATION															
		5 - 10	ABOVE GRADE	100	.	.	.	.	.	.	.	.				
	DEBRIS MELT THRU AT 30 MINUTES	5 - 30	ABOVE GRADE	.	1	5	5	5	25	1	1	.	0.2	3000	.	.
	1900 K DEBRIS WATER	30 - 120	BELOW GRADE	.	.	.	.	.	50	20	13	.	1.0		.	.
		30 - 180	BELOW GRADE	.	.	.	.	.	.	.	.	.	.	.	147	100

a. [illegible] heat [illegible] to coolant temperature over 5 seconds.
b. [illegible] next 5 minutes
c. [illegible]
d. [illegible] as generated from steam explosion

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Table 4.3–8, Sheet 3

Release Category Fission Product and Material Sources to the Confinement

PRIMARY RELEASE CATEGORY	COMMENTS	TIME INTERVAL (minutes)	RELEASE LOCATION[a]	FISSION PRODUCT SOURCES[b] (% OF CORE INVENTORY) Noble Gases	Molec. Iodine	I as CsI	Cs as CsOH	Te as TeO	Sr as SrO	Eu/Sm as Oxides	Ba as BaO	Ru/La/Ce as Oxides	Aerosol Size Mean[d] (μm)	SATURATED STEAM (kg)	HYDROGEN[c] (kg)	MCCI AEROSOLS (kg)
9 LOPA DUE TO SECONDARY LOCA	STEAM EXPLOSION															
		5 - 10	ABOVE GRADE	100	-	-	-	-	-	-	-	-	-	-	-	-
	DEBRIS MELT THRU AT 29.9 MINUTES	5 - 30	ABOVE GRADE	-	1	5	5	5	25	1	1	-	0.2	3000	-	-
	1900 K DEBRIS WATER	29.9 - 30	BELOW GRADE	-	-	1	1	1	15	1	1	1	5.0	e	269[f]	g
		30 - 120	BELOW GRADE	-	-	-	-	-	50	20	13	-	1.0	-	-	-
		30 - 180	BELOW GRADE	-	-	-	-	-	-	-	-	-	-	-	147	100
10 LOSHA DUE TO BASIN DRAIN	STEAM EXPLOSION															
	VESSEL BOIL OFF PERIOD	5 - 80	ABOVE GRADE	-	-	-	-	-	-	-	-	-	-	-	-	-
		80 - 85	ABOVE GRADE	100	-	-	-	-	-	-	-	-	-	45000	-	-
	DEBRIS MELT THRU AT 110 MINUTES	80 - 110	ABOVE GRADE	-	50	25	50	50	25	10	6	-	2.0	300	-	-
		29.9 - 30	BELOW GRADE	-	-	1	1	1	15	1	1	1	5.0	e	269[f]	g
		110 - 250	ABOVE GRADE	-	-	25	25	10	-	-	-	-	1.0	-	-	-
		110 - 250	BELOW GRADE	-	-	-	-	-	50	20	13	-	1.0	-	170	1000
11 LOSS OF COOLANT ACCIDENT	STEAM EXPLOSION															
		5 - 10	ABOVE GRADE	-	-	-	-	-	-	-	-	-	-	-	-	-
		10 - 15	ABOVE GRADE	100	-	-	-	-	-	-	-	-	-	3000	-	-
	DEBRIS MELT THRU AT 30 MINUTES	10 - 30	ABOVE GRADE	-	50	25	50	50	25	10	6	-	20	300	-	-
	1900 K DEBRIS WATER	29.9 - 30	BELOW GRADE	-	-	1	1	1	15	1	1	1	5.0	e	269[f]	g
		30 - 120	BELOW GRADE	-	-	25	25	10	-	-	-	-	-	-	-	-
		30 - 120	BELOW GRADE	-	-	-	-	-	50	20	13	-	1.0	-	147	100

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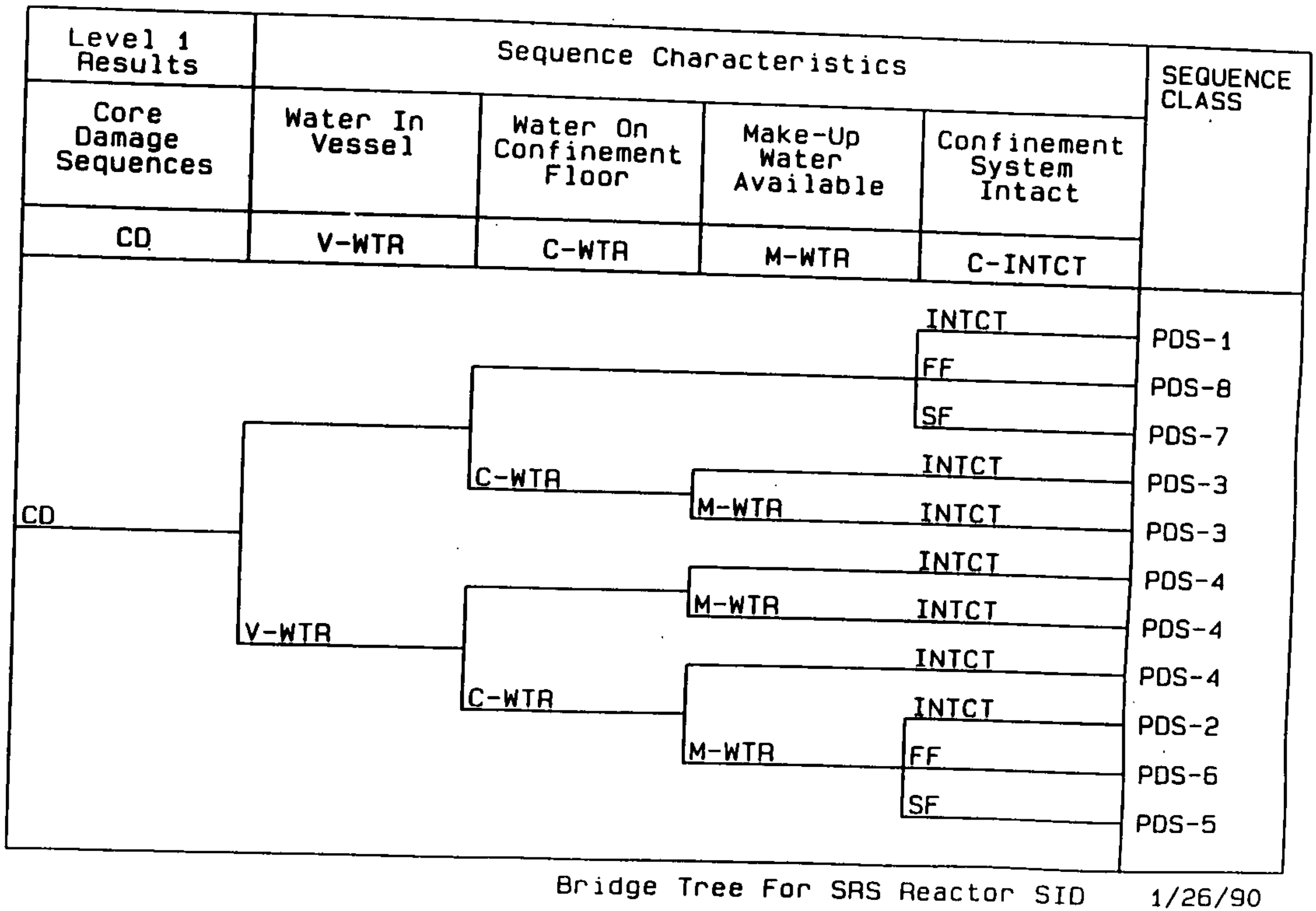
Figure 4.3-1
Bridge Tree for Plant Damage State Binning



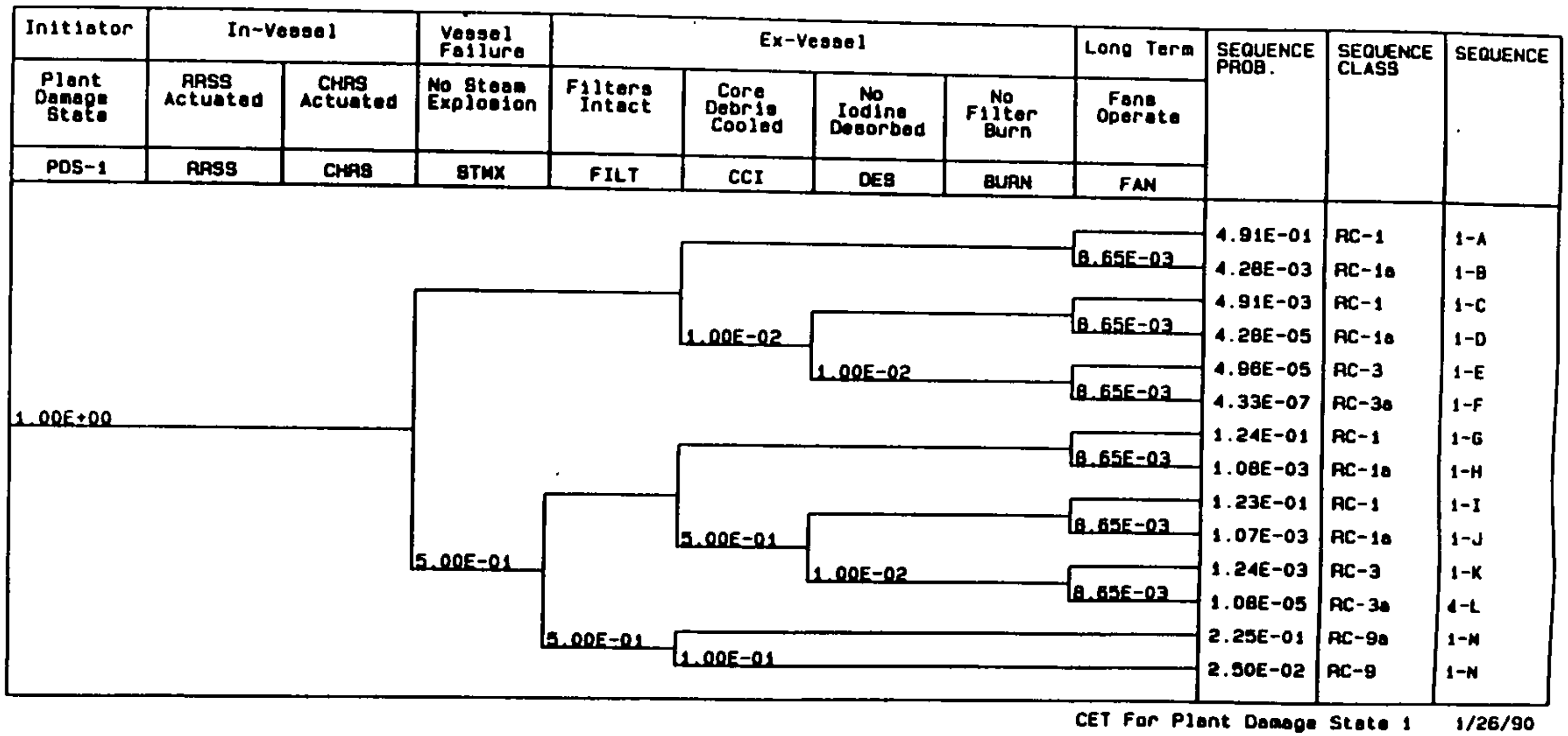


Figure 4.3-2
Confinement Event Tree for PDS 1

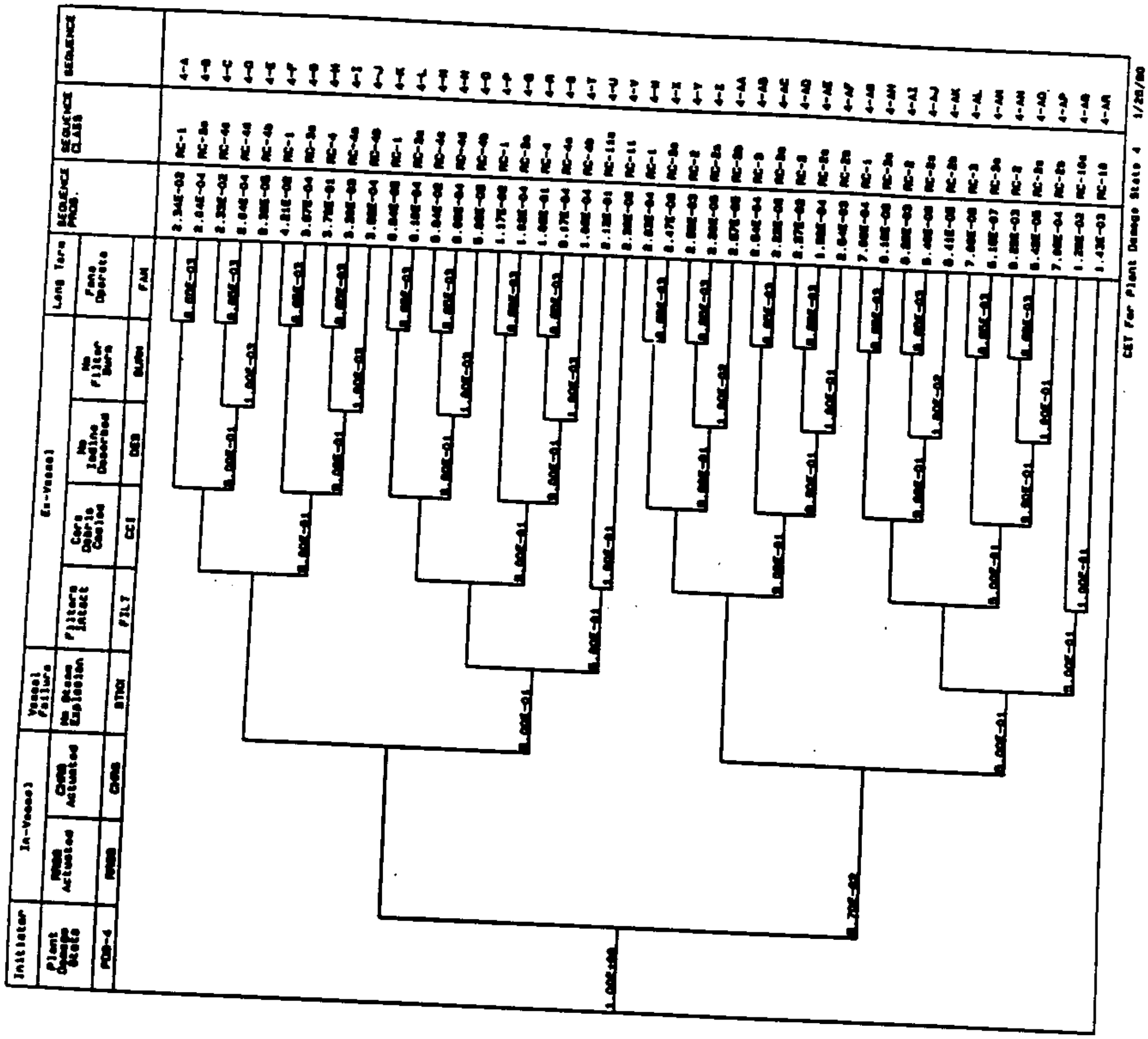


Figure 4.3-3
Confinement Event Tree for PDS 2

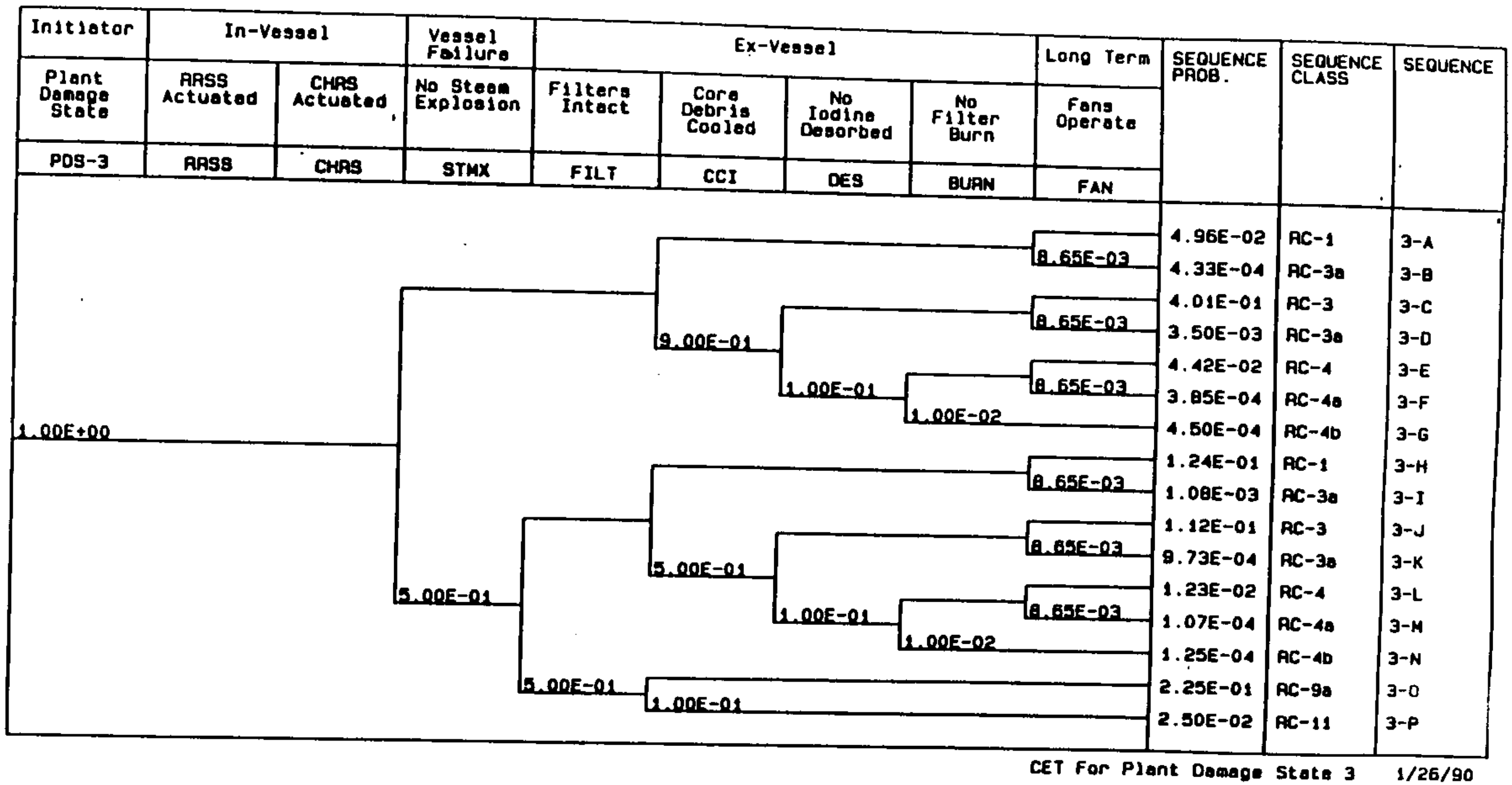


Figure 4.3–4
Confinement Event Tree for PDS 3

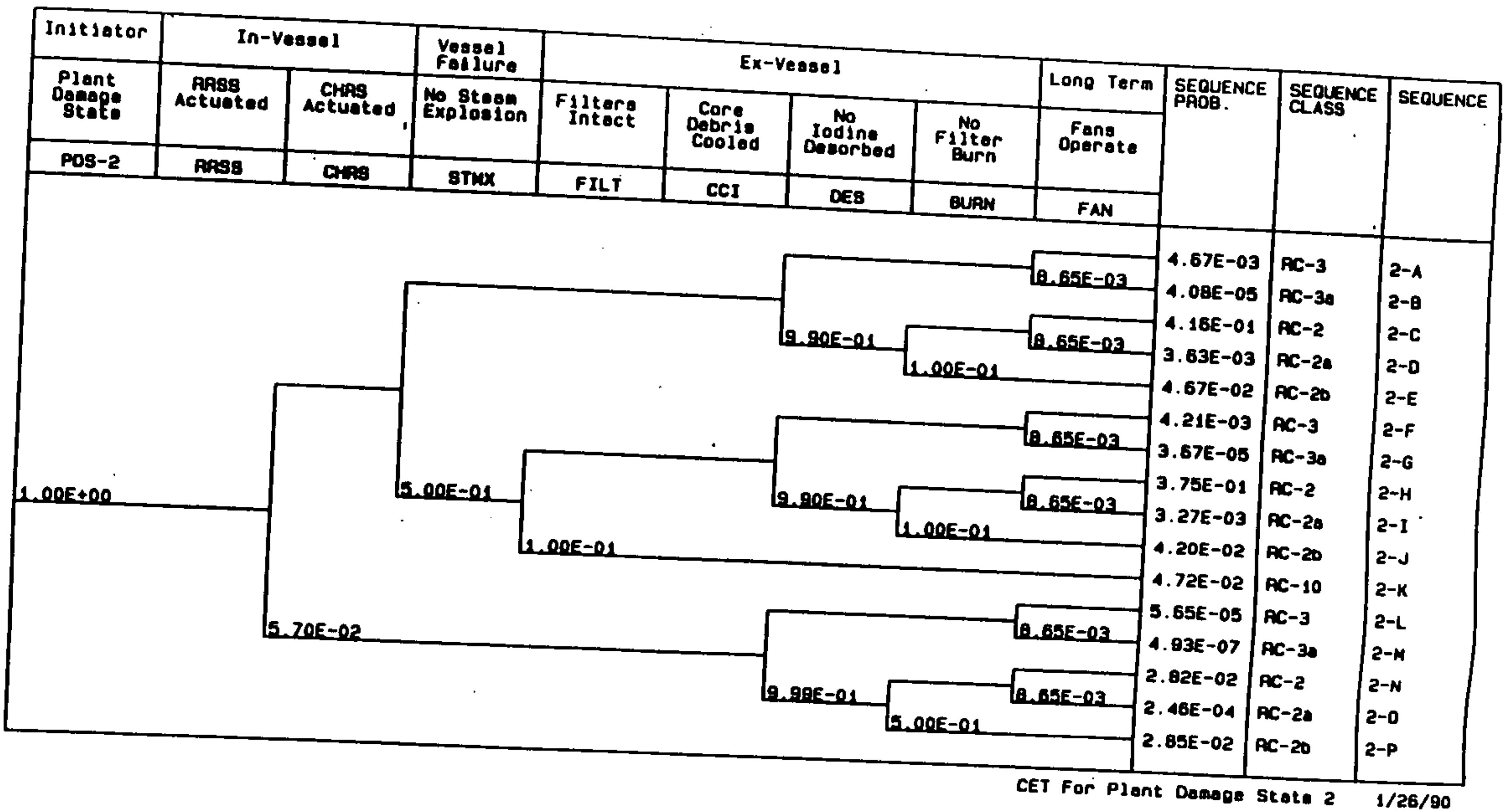


Figure 4.3-5
Confinement Event Tree for PDS 4

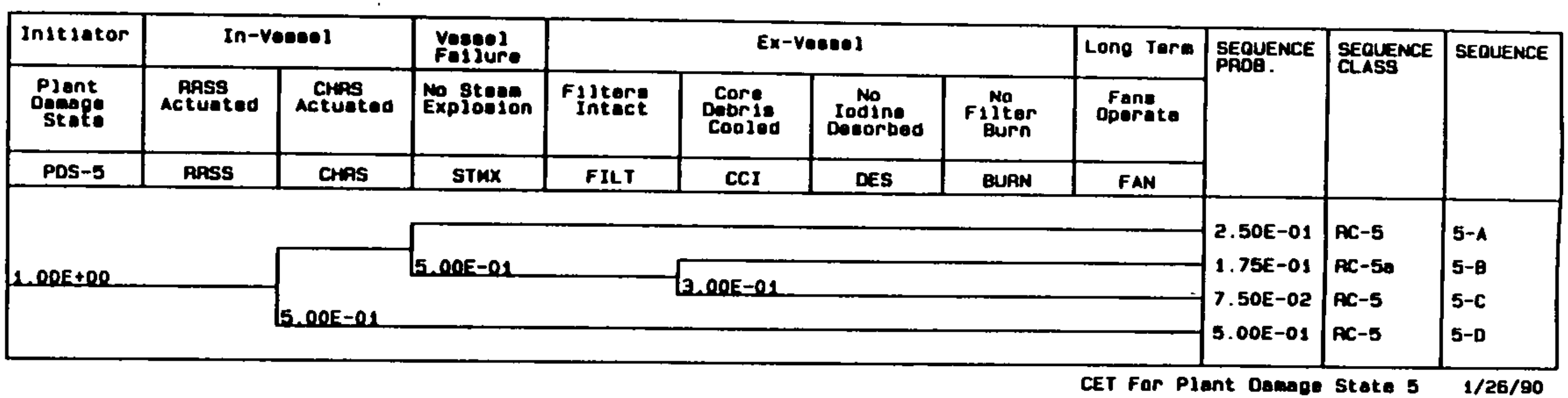


Figure 4.3–6
Confinement Event Tree for PDS 5

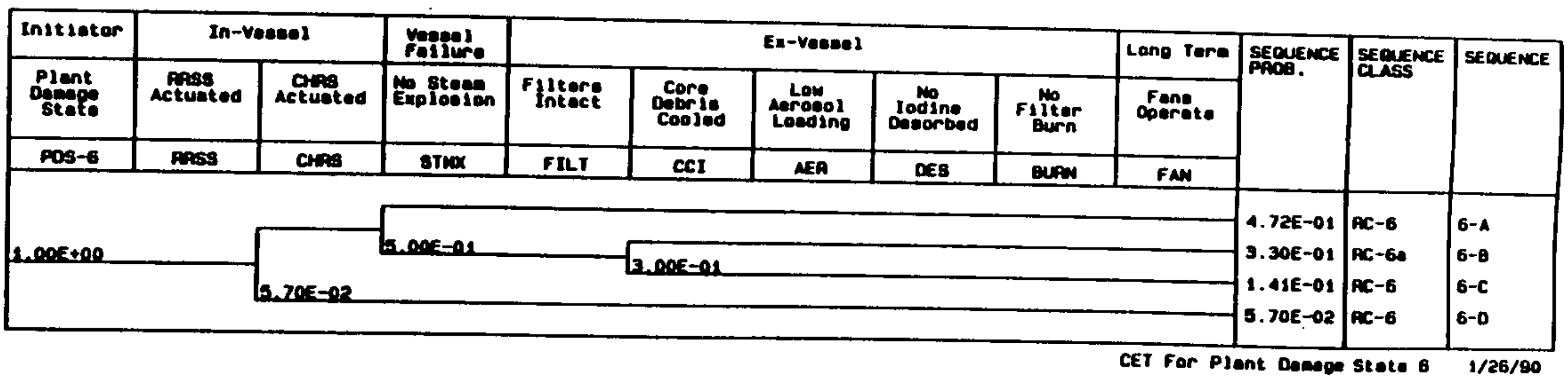


Figure 4.3-7
Confinement Event Tree for PDS 6

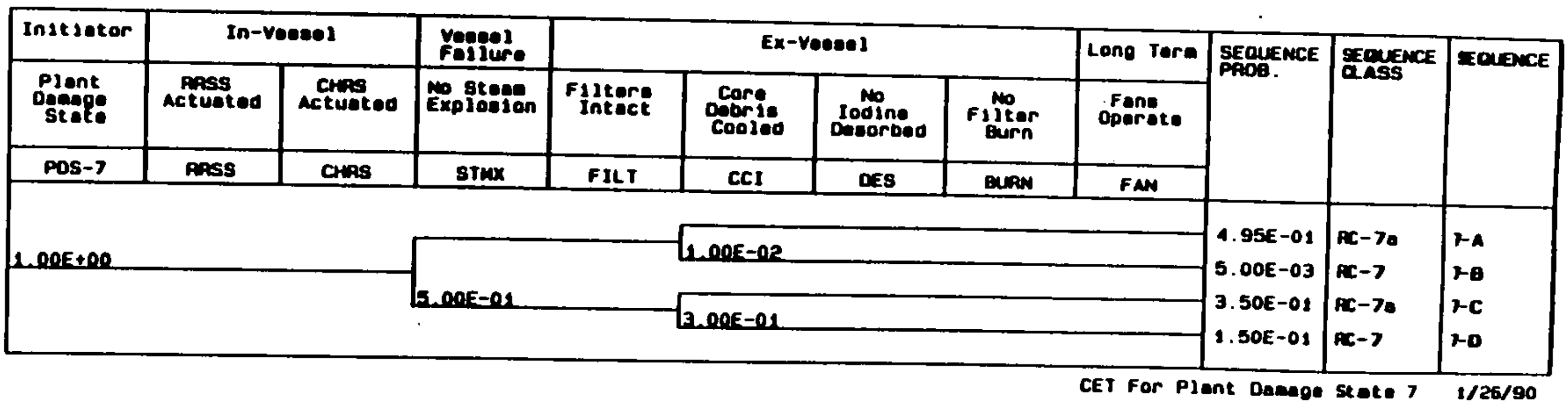


Figure 4.3–8
Confinement Event Tree for PDS 7

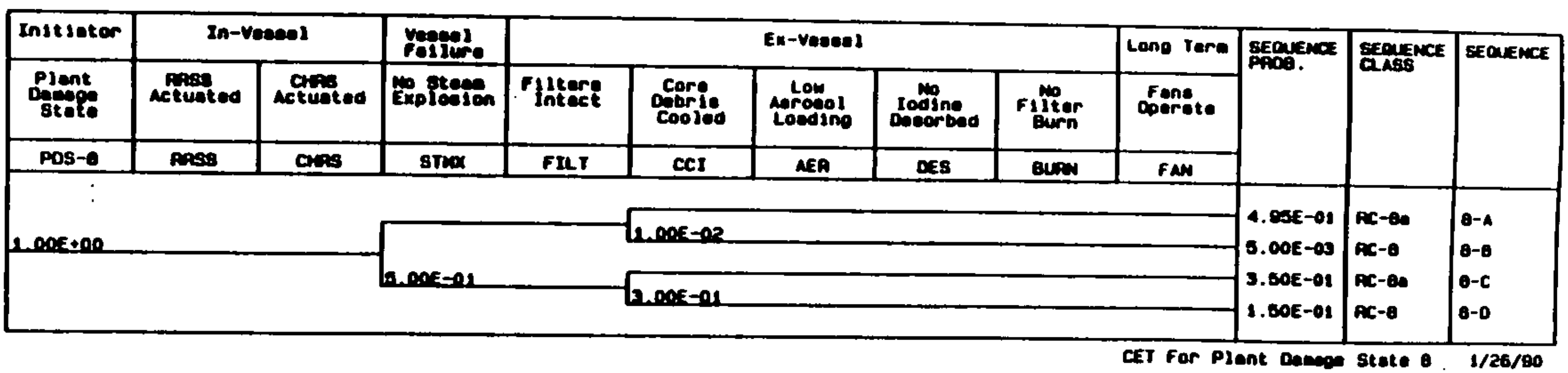


Figure 4.3–9
Confinement Event Tree for PDS 8

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4.4 Consequence Analysis

Consequences of hypothetical severe accidents are presented in this section as health effects incurred by the populations on the Savannah River Site and in the surrounding area. Individual calculations predict the dispersal of radioactive material away from the source, account for the deposition of radionuclides released, and estimate the health effects for each release category. The health effects are expressed as the probabilities of prompt and latent fatalities, and the total dose to the population within 500 miles. The safety goals currently proposed by DOE are discussed, and the risk of reactor operation is compared to those safety goals.

As described earlier, source terms for each release category were calculated based on mechanistic computer models and engineering judgement. The source terms are processed using a probabilistic consequence analysis code as described in this section. For these analyses, the MELCOR Accident Consequence Code System (MACCS) is the principal best-estimate code available for consequence modeling. MACCS Version 1.4, released in 1987 by Sandia National Laboratories (SNL), was modified by SRL to incorporate tritium in the dose conversion file. This change permits calculation of effects due to inhalation and water ingestion from the initial plume passage and due to resuspended tritium. Additional analyses permit assessment of long-term tritium uptake in the food chain.

Health effects calculated in this analysis include short-term (radiation exposure-induced fatalities and injuries) and long-term (latent cancer fatalities and injuries) categories. Models interpreting the doses to individual target organs for various health effects are discussed in "Health Effects Models for Nuclear Plant Accident Consequence Analysis," NUREG/CR-4214. These models are based on the 1980 BEIR-III study, The Effects on Populations of Exposure to Low Levels of Ionizing Radiation. The cancer estimates provided in this section are summed over all types for all critical organs. A quadratic model is used to address cancer risk for organ doses below a threshold of 1.5 Sv (150 rem). Above that level, a linear fit is employed.

4.4.1 Assumptions And Input Data

The MACCS source term input is arranged into ten isotope release groups shown in Table 4.4-1. The release from a given group is specified by start time and duration for each released plume, and release timing and environmental transport is the same for all isotopes within a given group. This allows different characterization of transport characteristics for radionuclides in the plume. Table 4.4-2 lists the end-of-cycle fission product inventory for a tritium producing charge and indicates the MACCS release group assignment.

Each source term considered may be differentiated into early and late release timing. Fission product release from the fuel in the reactor tank to the AACS as the core melts is treated as an early release. Following failure of the primary coolant system, two other large contributors to fission product release may occur. Release from the MCCI and possible release from MFCI is treated as a late release. Table 4.4-3 specifies the release fraction and location (stack, roof, or ground) by release category and isotope release group.

An offsite population database developed from the 1980 census data was utilized for the offsite analyses for a 500-mile radius from K reactor. Consequences for 1990 populations may be estimated by scaling the reported levels of consequence by 1.14. However, no adjustment was used in this study because of nonuniform growth patterns in the Central Savannah River Area (CSRA) and elsewhere.

The population database for the onsite cases reflect a day-averaged condition in the period from April to August, 1989. This was developed from conservatively weighted occupancy assumptions for peak daytime conditions and evening-early morning conditions. This weighting assumes the peak onsite workforce is present for eleven hours during the day to account for "rush-hour" traffic and transit time. The off-normal workforce is assumed for the remaining thirteen hours. The weighted peak and off-normal populations are summed by location to yield the onsite population dataset used in the MACCS calculations.

MACCS meteorological data files for K Area for the years 1982 to 1986, inclusive, were developed by SRL in 1987 and consist of hourly data derived from K Area meteorological tower and Augusta (Bush Field) Airport readings. For this study, 1986 weather is used because it gives mid-range or high results relative to the other four years of hourly data. The offsite cases use the seasonal afternoon mixing layer heights, and the onsite cases use an average of the peak afternoon height and early morning heights. The mixing layer heights are shown in Table 4.4-4.

Land usage factors are used in the calculation of offsite, chronic health effects because the effects are primarily due to food pathway uptake of radionuclides. The land usage factors for this study are taken from the 1987 Statistical Abstract of the United States, 107th edition (1987) published by the U. S. Department of Commerce. Onsite calculations do not use this data because only the early-phase, non-chronic portion of the accident is considered.

The above data have been assigned to three MACCS grids for calculation of consequences: (l) Onsite, consisting of one-mile radial rings from 0 to 10 miles from K Reactor, and two-mile intervals from ten miles to twenty miles; (2) Offsite/near-field, consisting of one-mile rings from 0 to 22 miles from K Reactor; and, (3) Offsite/far-field, consisting of 0 to 500 miles offsite from K Reactor. The onsite grid is used to compute seven measures of consequence. The near-field offsite grid is used to compute six measures of consequence. The far-field offsite grid is used to compute two quantities: (1) the total latent cancers (0 to 500 miles); and (2) the population dose to red bone marrow (0 to 500 miles). Under some meteorological conditions, the plume may move beyond a radius of 500 miles. A boundary weather condition is used to deplete the plume and to deposit remaining activity. This procedure allows nearly complete accounting of the source term impact.

4.4.2 Determination of Consequences

Offsite consequences are computed for three phases of each assumed reactor accident. The first phase considered is an emergency phase. This phase begins with the radiological release and lasts for seven consecutive days from the arrival of the first plume segment in a given grid element. If the average red bone marrow dose from groundshine to an individual is projected to be greater than 0.25 Sv (25 rem) in the seven day period, the entire population in that grid element is relocated at twenty-four hours from the arrival of the plume in the grid element for the remainder of the emergency phase.

The second, or intermediate phase, begins at seven days and lasts for thirty days from the accident initiation time. If a direct dose to the lungs of more than 0.05 Sv (5 rem) is projected for an individual through groundshine and resuspension during the thirty day period, then the population in that grid element is relocated for the remainder of the intermediate phase.

The final, long-term phase lasts for thirty years following the end of the intermediate phase. If a lung dose of more than 0.25 Sv (25 rem) from groundshine and resuspension is projected to be exceeded over the 30-year period, then decontamination and/or interdiction actions are employed to reduce or limit the critical organ dose. Calculation of offsite consequences for this third phase includes both the dose to resident populations on a grid element and the incremental dose incurred through decontamination efforts.

The onsite consequence analysis considers only the first, or emergency phase of the accident. Relocation parameters are set to prevent return of evacuated and sheltered workers within the emergency phase of the accident. Also, for the onsite calculations, interdiction/decontamination activities are removed from the overall dose and health determination.

4.4.3 Consequence Mitigation Assumptions

The evacuation and/or sheltering plan associated with each source term is summarized by Tables 4.4-5 and 4.4-6 for offsite and onsite populations respectively. Dose reduction factors used for evacuation and sheltering strategies are noted at the bottom of each table. The factors account for the reduction in dose received from cloudshine, inhalation, skin contamination, and groundshine during evacuation, sheltering, and normal activity in the shelter/evacuation zone. Factors of unity imply that the [illegible] dose is received from the dose pathway in question. In this analysis, it is assumed that dose reduction factors for the onsite and offsite cases are the same, so that onsite sheltering affords no greater relative protection than offsite.

For the general public, an emergency response model that is approximately the same as the model employed in the Draft Environmental Impact Statement for Plant Vogtle, NUREG-1087, was developed. The offsite evacuation model assumes that all populations within twenty miles of K Reactor begin evacuation at two hours after the release of the first plume segment. All evacuees move radially outward at a speed of 2.5 miles per hour (mph). Once reaching the twenty-mile radius, evacuees are not subject to any additional dose within the emergency phase of the accident. The offsite model beyond the twenty mile ring does not actively evacuate anyone, but instead may relocate or interdict depending on projected dose criteria.

Major seismic events could present difficulties to the evacuating public because roadways and traffic control systems could be disrupted and any mitigative activity could be delayed. Therefore, for the major seismic release categories it is assumed the offsite population within twenty miles of K Reactor does not evacuate until four hours after the time of the first released plume. It is further assumed the evacuees can only move at the speed of 1.25 mph. Once these evacuees reach the twenty-mile radius, they are assumed to receive no additional dose.

The onsite workforce is assumed to be informed of the impending radiological release by the SRS communication network. In the event of a reactor accident, directives to workers would be coordinated by the onsite Technical Support Center. Workers in the first five miles of the K Reactor are assumed to begin evacuation at 0.5 hours after reactor scram and travel radially outward at a speed of 8.5 mph. The evacuees in this case include K Area employees. Once the evacuees reach a radius of five miles, they are assumed to be sheltered and are no longer subject to any influence of the radioactive plume or to resuspended radioactivity for the balance of the calculation.

For the major seismic release categories, it is assumed the start time for the onsite evacuation action is delayed to one hour after shutdown. Furthermore, it is assumed the onsite evacuees move at one-half of the nonseismic speed, or at 4.25 mph.

4.4.4 Discussion of Results

Three indices of consequence are presented for this analysis, viz., early or prompt fatalities, latent or cancer fatalities, and population dose to red bone marrow. The results are presented in the complementary cumulative distribution frequency (CCDF) format, whereby the conditional probability of exceeding a level of consequence is plotted as a function of the consequences. A typical curve shows the conditional exceedance probability versus magnitude of consequences, and ranges from a high conditional probability – low consequence portion to a low probability – high consequence portion.

4.4.4.1 Offsite Dose and Health Effects

Within ten miles of the Savannah River Site boundary, the mean offsite population dose to red bone marrow per reactor-year from severe accidents is calculated as 2.1 person-Sv/reactor-year (210 person-rem/reactor-year). The equivalent mean dose per reactor-year from severe accidents to an individual in this offsite region is 0.059 mSv/reactor-year (5.9 mrem/reactor-year). The CCDF plot for the ten-mile population dose is shown in Figure 4.4-1.

Mean values only are reported for the population dose within 500 miles. The mean value for the total offsite population dose to red bone marrow, weighted by the frequencies of contributing release categories is estimated to be 9.0 person-Sv/reactor-year (900 person-rem/reactor-year). The mean bone marrow dose to an individual living within a 500-mile radius of K Reactor is 1.8×10^{-7} Sv/reactor-year (0.018 mrem/reactor-year). The analyses performed to support these estimates indicate that 62% of population dose per year is based on external (seismic) events, while the remainder is due to internal events.

The frequency of exceeding various levels of prompt fatalities in the offsite population is given as the lower curve in Figure 4.4-2. The prompt fatality magnitude is a function of release category magnitude and frequency, the wind-rose and other meteorological conditions within the local environs, the near-field population distribution and the emergency response effectiveness. The prompt fatalities tend to be almost completely due to the first phase of the assumed accident. These are based principally on the source term size and timing of the first plume segment. The offsite prompt fatalities are dominated (> 90%) by the dry tank - dry below-grade release categories associated with the stack building failure mode assumed for the major seismic event (RC-5 and RC-5A). Other than several internal events coupled to steam explosions (RC-9, -10, -10A, -11, and -11A), no other release categories contribute to offsite prompt fatalities.

The frequency of exceeding various levels of cancer fatalities is shown as the uppermost curve in Figure 4.4-2. Cancer fatality differences among various source terms are predominately due to the source term size and frequency, and secondarily due to the long-term protective measures invoked. Evacuation and sheltering response activities have minimal impact on the overall cancer fatality CCDF.

Internal events (RC-9A, -11, and -11A) with steam explosions comprise 42% of the expected value of 0.09 offsite cancer fatalities/reactor-year from severe accidents, within 500 miles of the site boundary. Two seismic release categories, viz., RC-5 and RC-7, contribute an additional 23%.

Figure 4.4-2 indicates offsite prompt fatalities are three to four orders-of-magnitude lower than the latent cancer magnitudes across all levels of consequence. The difference is due primarily to the substantial distance from K Reactor to the site boundary. The reactor-to-site-boundary distance ranges from 5.9 miles to 14.4 miles. Prompt fatalities are modeled as threshold effects with the health effect models in this consequence analysis. The parameters set for these effects presume negligible lethal dose risk for doses below 2 Sv (200 rem). Because of K-Reactor siting within the SRS DOE reservation borders, there are few source terms capable of delivering the requisite doses before deposition. Meteorological patterns deplete the plume sufficiently to preclude the occurrence of early deaths. Additionally, close-in offsite regions (within ten to fifteen miles of K Reactor) are sparsely populated in most directions. The latent effect magnitudes are due largely to long-term factors several tens to several hundred miles distant from the reactor release point. For the latent effects, introduction of released radioactivity into the food pathway and decontamination effort-incurred dose adds significantly to the overall population dose. The total population susceptible to these effects is much greater than for the near-field effects, so that the overall result is one of relatively few offsite early fatalities compared to larger predicted latent fatalities.

Table 4.4-7 summarizes the near-field offsite expectation or mean estimate for the following measures of consequence: (1) the number of prompt fatalities per reactor-year within one mile of the site boundary; (2) the number of prompt fatalities per reactor-year within ten miles of the site boundary; (3) the total number of latent cancer fatalities per reactor-year within ten miles of the site boundary; (4), the individual early fatality risk within one mile of the site boundary; (5) the individual early fatality risk within ten miles of the site boundary; and (6) individual latent risk for the ten-mile offsite boundary. The estimates are the summed means of the various CCDF distributions weighted by the release category frequency. Two sub-totals are provided, viz. the sum over internal initiators and the sum of all external (seismic) initiators (RC-5, -5A, -6, -6A, -7, -7A, - 8, and -8A). The largest component of all consequences shown is that of the seismic event contributors. The steam explosion–enhanced release categories are the significant source terms controlling the internal event contributors. The dominant internal and external cases share the common characteristic of relatively large releases from iodine, tellurium, cesium, and strontium fission product groups occurring in at least one plume segment.

4.4.4.2 Onsite Dose and Health Effects

The mean value of the dose to red bone marrow for the onsite population weighted by the frequencies of contributing release categories, is estimated to be 2.5×10^{-2} person–Sv/reactor–year (2.5 person–rem/reactor–year). Based on the day–averaged site population used in this analysis, the mean red marrow dose from reactor accidents to an individual working at the Savannah River Site is 0.0031 mSv/reactor–year (0.31 mrem/reactor–year). Approximately 55% of this dose comes from internal events. Of these, the steam explosion sequences are the largest contributors. However, release categories RC–1, RC–3, and RC–4 also constitute a significant dose component.

The overall dose to red bone marrow for the onsite population is shown as the lower curve in Figure 4.4–3. The offsite dose is shown as the upper curve. The low probability – high population dose portion of the onsite CCDF is due to low–dispersive meteorological conditions providing minimal plume dilution before reaching the portion of the onsite population assumed not to evacuate in this analysis (those located at five miles radius from K Reactor and beyond).

The exceedance frequency for total early fatalities and total latent cancer deaths to the onsite population is plotted in Figure 4.4–4. The low frequency, high consequence portion of the prompt fatality CCDF (the portion of the CCDF that "crosses over" the latent fatality CCDF) results from highly unlikely meteorological conditions that provide minimal plume dilution before reaching the portion of the onsite population assumed not to evacuate by this study. This latter assumption is highly conservative. An alternative emergency response plan with both evacuation and sheltering is shown in the sensitivity analysis section. Over 88% of the prompt fatality risk at the mean level of consequence is based on the steam explosion–enhanced source terms associated with RC–9A (65%), RC–11A (15%), and RC–11 (8%). Slightly more than 4% is derived from the RC–1 source term with early release timing characteristics coupled to an appreciable release category frequency. The mean frequency for prompt fatalities is 8.44×10^{-5} prompt fatalities/reactor-year.

The onsite mean latent fatality frequency is 1.10×10^{-3} latent cancer fatalities/reactor-year. Of this total, 68% is from the internal events. This component of risk is mostly due to RC-9A, RC-11, and RC-11A. The majority of the external (seismic) contribution is due to the dry tank - dry reactor building floor categories, RC-5 and RC-6A.

Table 4.4-8 summarizes the onsite mean estimate for the following measures of consequence: (1) the number of prompt fatalities per reactor-year within one mile of K Reactor; (2) the number of prompt fatalities per reactor-year within ten miles of K Reactor; (3) the number of latent cancer fatalities per reactor-year within ten miles of K Reactor; (4) the number of latent cancer cases per reactor-year for the full site; and (5) the total red bone marrow dose per reactor-year for the full site. The sixth and seventh columns correspond to the individual prompt fatality risk and the latent cancer death risk within one mile and within ten miles of K Reactor respectively. As in the case of the offsite calculations, the estimates are the summed means of the various CCDF distributions weighted by the release category frequency. Two sub-totals are provided, viz., the sum over internal initiators and the sum of all external initiators (RC-5, -5A, -6, -6A, -7, -7A, -8, and -8A). The internal events dominate for all measures of consequence except for the total onsite dose per reactor-year.

Two features of Table 4.4-8 should be elaborated. First, the onsite grid centered at K Reactor covers the location of virtually the entire site population by the ten-mile radius. Hence, there are negligible increments to the ten-mile totals by extending to the entire site. The second feature of importance from the table is the prompt fatalities and the prompt fatality risk within one mile of K Reactor. There are no co-located facilities with onsite workers stationed within one mile of K Area. It is only due to the somewhat artificial feature of the evacuation model employed that K-Area workers are included in the evacuating 0- to 5-mile zone considered. Exclusive of the K-Area workers, both measures would be identically zero over all levels of consequence.

4.4.5 Severe Accident Risks

The risk from postulated severe accidents associated with operation of the SRS reactors is the summed product of the frequency of a reactor accident sequence, the conditional probability of radioactive release to the environment for that sequence, and the consequence given that the release has occurred. Both offsite and onsite risks have been determined in this study.

The offsite risks were given earlier, in Table 4.4–7. The results of this analysis show that the mean risk of an accidental death (prompt fatality) resulting from an accident at one of the Savannah River reactors, for an offsite individual residing within 1 mile of the site boundary, is 5.88×10^{-11} per reactor-year of operation. The mean risk of cancer (latent fatality) resulting from an accident at one of the Savannah River reactors, for an offsite individual residing within 10 miles of the site boundary, is 3.4×10^{-8} per reactor-year of operation. When compared to the naturally occurring risks to the offsite population for accidental death and cancer, the risks associated with accidents at a Savannah River reactor represent a contribution which is considerably less than 0.1%.

The mean onsite risks were tabulated earlier in Table 4.4–8. These results show that the mean risk of an accidental death (prompt fatality) resulting from an accident at one of the Savannah River reactors, for an average individual on the Savannah River Site is 1.6×10^{-7} per reactor-year of operation. The mean latent fatality risk resulting from an accident at one of the Savannah River reactors, for an average individual on the Savannah River Site work force, is 1.4×10^{-7} per reactor year of operation. When compared to the other occupational risk of accidental death to the work force, the average risk to the work force population of prompt fatalities associated with hypothetical accidents at the Savannah River reactors represents a contribution which is approximately 0.2%. The latent fatality risks represent a 0.007% contribution to the cancer risks that are faced by an average individual in the work force.

4.4.6 Comparison of Risk to Draft DOE Safety Goals

The Department of Energy published draft safety goals, in May 1989, for the severe accident risks to the offsite population and the onsite work force at DOE facilities. The primary Safety Goal is to assure that there is no significant increase in risk to individuals resulting from the operation of DOE facilities. Specifically, the quantitative Safety Goals for the offsite population are:

(1) The risk, to an average individual in the vicinity of a DOE facility, of early fatalities that might result from accidents should not exceed one-tenth of one percent of the sum of early fatality risks resulting from other accidents to which members of the U.S. population are generally exposed. The accidental fatality rate in the U.S. is given as 4.0×10^{-4} per year per individual. The vicinity is interpreted as extending 1 mile from the DOE reservation boundary.

(2) The risks to the population in the area near a DOE facility of cancer fatalities that might result from radiological exposure originating

within the facility should not exceed one-tenth of one percent of the sum of cancer risks to which members of the U.S. population are generally exposed. The average cancer fatality rate in the U.S. is given as 2.0×10^{-3} per year per individual. The area near a facility is interpreted to extend 10 miles from the DOE reservation boundary.

(3) The risk to an average individual worker in the vicinity of a DOE nuclear facility, of prompt fatalities that might result from radiological accidents should not exceed one percent of the sum of prompt fatality risks typical of average occupational death rates in the U.S. The occupational death rate in the U.S. is given as 1.0×10^{-4} per year per individual. The vicinity of a DOE facility refers to the zone from the security fence, to one mile beyond.

(4) The risk to workers in the area near a DOE nuclear facility of cancer fatalities that might result from radiological exposure originating with accidents at the facility should not exceed one-tenth of one percent of the sum of cancer risks to which members of the U.S. population are generally exposed. The average cancer fatality rate in the U.S. is given as 2.0×10^{-3} per year per individual. The area near a DOE nuclear facility extends from the facility control perimeter for a distance of ten miles beyond.

A comparison of the individual risks to the draft DOE Safety Goals found from these scoping assessments is presented in Table 4.4–9.

4.4.7 Sensitivity Analyses

4.4.7.1 K Reactor, P Reactor, and K Reactor Siting

The P Reactor location has also been evaluated for offsite risk to the individual due to hypothetical reactor accident releases. A near-field equivalent of the second grid discussed above was established centered at P Area. Dose and related health effects within the first twenty miles offsite for the P Reactor location are calculated with the consequence analysis procedures described earlier. For the same total radiological release, the P and K Reactor sites will have indistinguishable health effects at distances greater than one hundred miles. Thus, far-field calculations (extending to a radius of five hundred miles from K Reactor) were not repeated. Furthermore, due to the L-Reactor location, assessments for K and P Reactor severe accident releases will bound L Reactor severe accident impacts to the near-field offsite region.

Figure 4.4–5 shows the exceedance frequencies for individual [illegible] of cancer death within ten miles of the SRS boundary [illegible] differences across five orders-of-magnitude of frequency [illegible]

10^{-2}) are negligible for P- and K-Reactor sites. Similarly, the individual prompt fatality risk for the one-mile offsite population is plotted for the two reactors in Figure 4.4-6. The P-Reactor CCDF is nearly an order of magnitude higher than K for individual risk levels above 10^{-5}. The P-Reactor facility has a mean that is 1.4 times greater than that of K (8×10^{-11} compared to 5.9×10^{-11}). However, both results are substantially below the draft DOE Safety Policy goals.

4.4.7.2 Onsite Emergency Response

Sensitivity of onsite consequences to alternative evacuation/sheltering scenarios was evaluated for two release categories. Results are shown in Table 4.4-10 for three scenarios: evacuation within five miles of the reactor and no evacuation beyond five miles (i.e., the base case for this study); evacuation within five miles of the reactor and a nominal sheltering scenario beyond five miles (i.e., sheltering beginning at one hour for non-seismic cases and two hours for seismic cases); and no onsite evacuation/sheltering. Several measures of onsite consequences are tabulated in Table 4.4-10 for release categories RC-2b and RC-5.

Figure 4.4-7 shows the complete CCDFs for both prompt and latent fatalities for the second scenario (involving onsite evacuation and sheltering) outlined above. These curves also account for the full onsite workforce, rather than the day-averaged totals described earlier as the base case. The full workforce is approximately twice the day-averaged level and is conservative since all shifts are assumed present. The comparable averaged workforce CCDFs are found in Figure 4.4-4.

The comparisons indicate that even limited emergency response measures provide significant onsite dose and health effect reductions, and that significant additional benefits come from implementation of onsite sheltering strategies.

4.4.7.3 Health Effects

The health effects model used in MACCS is based on interpretation of the 1980 BEIR-III report by the Harvard School of Public Health and Sandia National Laboratories. In mid-December of 1989, a BEIR-V report was issued. The new report on the biological effects of ionizing radiation, "Health Effects of Exposure to Low Levels of Ionizing Radiation," is based on new risk models, revised bomb survivor dose estimates, and additional epidemiological data. This report suggests that the risk of developing cancer following exposure to low levels of

X-rays and gamma-rays may be three to four times higher than previously thought.

Such findings are typically subject to several years of review before implementation into any regulatory guidelines or health effect models. Nevertheless, the effect of these findings on the risk estimates in this study were estimated. The cancer risk per unit dose was adjusted upward by a factor of three for all but the leukemia cancer type where it was adjusted by a factor of four.

The expectation value for the total offsite latent cancer fatalities for the RC-10 source term was found to be 1.04×10^{-4}, or a factor of 3.1 higher than the earlier calculations. An approximately threefold increase in latent health effect is, therefore, the adjustment to this study advised to estimate the sensitivity to the BEIR-V results.

4.4.7.4 Offsite Prompt Fatalities

The irregular shape of the Savannah River Site adds considerable complexity to the assessment of certain indices of risk within a given distance of the site boundary. In particular, within the limitations of a two-dimensional radial-azimuthal grid, a constant radius sweep to assess the one-mile offsite prompt fatalities will account for more than one mile of offsite population in some directions and less in others. The estimates presented earlier of offsite prompt fatalities within one mile of the site are conservative since they account for more than the one-mile population in the directions that yield the most early fatalities.

An alternative more accurate MACCS model was constructed for the near-field offsite consequence assessment of the individual risk of early fatality within one mile of the boundary. The model consisted of limiting the offsite population to only the first mile offsite in each of the sixteen compass directions. Beyond the one-mile ring in a given direction, the population was set identically to zero. A twenty-five mile radius grid was necessary to fully encompass all directions around the reactors. As the offsite evacuation model is run in the consequence calculation, the evacuees are followed (and are considered for lethal dose assessment) until reaching a distance of twenty miles from K and P Reactors. The recalculated mean individual prompt fatality risks for P and K Reactors are 9.2×10^{-11} and 5.9×10^{-11} per reactor year, respectively. Thus, the base case estimates are confirmed to be conservative by approximately one order of magnitude.

4.4.7.5 Contribution of Tritium to Offsite Dose

A full tritium inventory release has been evaluated for offsite dose and subsequent health effects. The analysis indicates that the whole-body dose is less than one percent of that due to fission products releases in severe accident scenarios. Furthermore, the population dose and related latent effects from the tritium component in the initial plume passage are negligible when compared with the doses incurred during the intermediate and long-term phases of the release.

Inhalation, skin absorption, and food ingestion/water uptake pathways contribute to the total tritium dose during the latter phases of the radiological release. The inhalation/skin absorption is that of resuspended tritium. Tritium in contaminated soil and crops will exchange with atmospheric hydrogen and thus be subject to uptake by humans. Second, a large contributor to long term tritium doses is through ingestion of food that is contaminated, but below any prescribed interdiction levels. The individual dose in this case is small; however, the integrated population dose tends to be relatively large. A full tritium inventory atmospheric release yields an offsite whole-body population dose of 1.5×10^3 person-Sv (1.5×10^5 person-rem) when the long-term dose pathways are integrated over a fifty year period.

4.4.7.6 Multiple Reactor Operation

The analysis of the likelihood, progression, and consequences of severe accidents for the Savannah River reactors presented in the previous sections of this document are based on consideration of only one reactor. The Savannah River Site has three reactors available for operation; the K Reactor, the L Reactor, and the P Reactor. These reactors can be, and have been, operated with an availability of 80%. This value accounts for reactor downtime for refueling, routine maintenance, repairs, and safety upgrades. Thus, the three reactors can be operated such that an average of 2.4 reactors would be in operation at any one time.

The risk associated with simultaneous operation of three reactors can thus be estimated to be, on the average, about 2.4 times the risk for a single reactor reported in this study. This estimate could be affected by the following factors:

The possibility of simultaneous failure among reactors due to common cause mechanisms, in response to an initiating event that effects two or more reactors;

The non-linearity in health effects exposure threshold affecting prompt fatality estimates; and

The sensitivity of offsite prompt fatality estimates to differences in siting.

Notwithstanding these factors, the operational limitations of the reactors are expected to limit the site risk from three reactor operation to no more than three times the single reactor risk as estimated by this study.

4.4.7.7 Reduced Power Operation

The radiological inventory present inside the reactor core is directly proportional to the reactor power level and the length of reactor operating cycle. Furthermore, substantial changes in the core damage frequency as a function of reactor power level are not expected to occur. Therefore, the risk of operating at reduced reactor power is expected to be bounded by the risk associated with operation at 100% power level.

4.4.7.8 Alternate Production Modes

The primary product produced in the Savannah River reactors is tritium. Future requirements for the SRS may include the production of plutonium-239 (Pu-239) and a variety of small quantities of special radioisotopes such as plutonium-238 (Pu-238).

The production of Pu-239 requires the use of reactor fuel and target assemblies different from those used in producing tritium. As a result, additional accident initiating events must be considered, in particular, misloading accidents where a target assembly is inserted in a fuel position, and dropped target slugs during reloading operations. The misloading accident is very unlikely because of a large number of procedural and physical limitations designed to prevent such an event. Additionally, this postulated accident would result in no more than localized fuel melting, so a significant quantity of radionuclides would not be released. The dropped target slug would result in melting of a single assembly only so a significant quantity of radionuclide would not be released. Thus the probability of a severe accident during production of Pu-239 is not appreciably greater than that during tritium production, and the difference in consequences is not significant in the context of severe accident consequences.

Comparisons of end-of-cycle fission product inventories for the Pu-239 and the tritium production charges show very small differences in the core inventories. The largest differences are for tritium and Pu-239. A Pu-239 production charge has a very small tritium inventory, and a tritium producing charge has essentially no plutonium-239. Plutonium, which has a boiling point of 3505 K, is not a volatile element under the conditions which would occur in a severe accident in the Savannah River reactors, so it would not contribute greatly to the health effects associated with a severe accident, and the effects of tritium are discussed in Section 4.4.7.5. All of the other differences in inventories are insignificant in terms of their potential effect on the onsite and offsite health consequence predictions.

The production of plutonium-238 requires the use of target assemblies containing neptunium-237, which displace reactor fuel or target assemblies. The presence of neptunium targets in a reactor loading will not yield any differences in the probabilities of severe accidents or a significant change in the fission product inventory. Neither plutonium-238, with a boiling point of 3505 K, nor neptunium-237, with a boiling point of 4175 K, are volatile elements under the conditions which would occur in a severe accident in the Savannah River reactors, and thus would have essentially no affect on health effects which would result from such an accident.

4.5 Summary and Conclusions

The risk from hypothetical severe accidents for the Savannah River reactors has been assessed using probabilistic risk assessment methodologies. The assessment has been performed to represent the restart condition of the reactors and characteristics of the site and its environs.

The severe accident frequency is predicted to be 1.3×10^{-4} per reactor year of operation from internally initiated events (e.g. pipe breaks, plant transients, etc.). The severe accident frequency due to seismically initiated events is predicted to be 6.0×10^{-5} per reactor year of operation. The severe accident frequency predicted for accidents initiated by fire is predicted to be 1.7×10^{-7} per reactor year of operation. Thus, the total core melt frequency is predicted to be 2.0×10^{-4} per reactor year of operation.

The risk to offsite individuals from hypothetical severe accidents at the Savannah River reactors has been calculated to be less than 0.1% of the background risk to which the population in the area of the Savannah River Site is exposed. Specifically, the mean risk of an accidental fatality to an average individual within one mile of the Savannah River Site boundary, as a result of a severe accident, is 5.88×10^{-11} per year. The risk of a cancer fatality to an average individual within 10 miles of the Savannah River Site, as a result of a severe accident, is 3.4×10^{-8} per year. Both of these values are well below the draft safety goals proposed by the Department of Energy for DOE reactor facilities.

The risk to individuals in the work force population at the Savannah River Site has also been found to be less than 0.1% of the background risks to which they are normally exposed. Specifically, the risk to an average individual in the worker population of fatality from a severe accident involving the release of fission products from the facility has been found to be 1.6×10^{-7} per year. The mean risk of cancer fatality due to a severe accident has been found to be 1.4×10^{-7} per year. Both of these values are a factor of six or more below the safety goals proposed by DOE for reactor facilities such as the Savannah River Site.

The conclusion is that hypothetical severe accidents the Savannah River reactors do not represent a significant detrimental risk to the population working at or living in the area of the Savannah River Site.

Table 4.4-1

Fission Product Groupings

Group No.	Group Label	MACCS Source Term Groups	CONTAIN Source Term Groups
1	NG	Xe, Kr	Noble Gas
2	I	I	Halogens [I, Br]
3	Cs	Cs, Rb	Cs, Rb
4	Te	Te, Sb	Te, Sb, Se
5	Sr	Sr	Sr
6	Ru	Ru, Co, Mo, Tc, Rh	Ru, Rh, Pd, Mo, Tc
7	La	La, Y, Zr, Nb, Pr, Nd, Am, Cm	La, Zr, Nd, Nb, Pm, Pr, Y
8	Ce	Ce, Np, Pu	Ce, Np, Pu
9	Ba	Ba	Ba + Eu, Sm
10	T	^{3}H	Noble Gas

Table 4.4-2
Reactor Radionuclide Inventory for Severe Accident Consequence Analysis

Radionuclide	Inventory, Bq	Release Group	Parent
Co-58	8.2×10^{18}	6	n*
Co-60	9.8×10^{18}	6	n
Kr-85	8.1×10^{15}	1	n
Kr-85m	1.2×10^{18}	1	n
Kr-87	2.4×10^{18}	1	n
Kr-88	3.3×10^{18}	1	n
Rb-86	2.6×10^{15}	3	n
Sr-89	4.1×10^{18}	5	n
Sr-90	6.6×10^{16}	5	n
Sr-91	5.6×10^{18}	5	n
Sr-92	5.7×10^{18}	5	n
Y-90	6.9×10^{16}	7	Sr-90
Y-91	5.0×10^{18}	7	Sr-91
Y-92	5.7×10^{18}	7	Sr-92
Y-93	6.0×10^{18}	7	n
Zr-95	5.2×10^{18}	7	n
Zr-97	5.7×10^{18}	7	n
Nb-95	4.3×10^{18}	7	Zr-95
Mo-99	5.9×10^{18}	6	n
Tc-99m	5.1×10^{18}	6	Mo-99
Ru-103	2.9×10^{18}	6	n
Ru-105	1.0×10^{18}	6	n
Ru-106	1.2×10^{17}	6	n
Rh-105	8.9×10^{17}	6	Ru-105
Sb-127	1.1×10^{17}	4	n
Sb-129	6.2×10^{17}	4	n
Te-127	1.0×10^{17}	4	Sb-127
Te-127m	1.1×10^{16}	4	n
Te-129	5.9×10^{17}	4	Sb-129
Te-129m	1.6×10^{17}	4	n
Te-131m	3.5×10^{17}	4	n
Te-132	4.1×10^{18}	4	n
I-131	2.8×10^{18}	2	Te-131m
I-132	4.1×10^{18}	2	Te-132
I-133	6.5×10^{18}	2	n
I-134	7.4×10^{18}	2	n
I-135	6.1×10^{18}	2	n
Xe-133	6.5×10^{18}	1	I-133
Xe-135	6.0×10^{17}	1	I-135
Cs-134	6.9×10^{16}	3	n

Table 4.4-2 continued

Radionuclide	Inventory, Bq	Release Group	Parent
Cs-136	2.8×10^{16}	3	n
Cs-137	6.5×10^{16}	3	n
Ba-139	6.2×10^{18}	9	n
Ba-140	6.0×10^{18}	9	n
La-140	6.1×10^{18}	7	Ba-140
La-141	5.7×10^{18}	7	n
La-142	5.7×10^{18}	7	n
Ce-141	5.5×10^{18}	8	La-141
Ce-143	5.7×10^{18}	8	n
Ce-144	1.8×10^{18}	8	n
Pr-143	5.6×10^{18}	7	Ce-143
Nd-147	2.1×10^{18}	7	n
Np-239	1.8×10^{18}	8	n
Pu-238	1.2×10^{15}	8	Cm-242
Pu-239	1.2×10^{13}	8	Np-239
Pu-240	8.5×10^{12}	8	Cm-244
Pu-241	2.5×10^{15}	8	n
Am-241	4.3×10^{11}	7	Pu-241
Cm-242	1.1×10^{14}	7	n
Cm-244	3.5×10^{12}	7	n
H-3	2.6×10^{18}	10	n

* None

Table 4.4–3
Radionuclide Releases to Environment, Fraction of Core Inventory

Release Category	Time Interval (minutes)	Release Location	Energy Release Rate (MW)	Noble Gas	Iodine	Cs	Te	Sr	Ba	Ru, La, Ce	Tritium
1	5–10	Stack	0.10	0.15	3.4E–07	3.4E–07	3.1E–07	1.8E–07	1.1E–07		0.15
	10–30	Stack	0.62	0.79	8.3E–06	8.3E–06	8.4E–06	4.2E–06	3.4E–06		0.79
	30–53	Stack	0.42	0.06	4.3E–06	4.3E–06	4.3E–06	6.2E–06	4.3E–06		0.06
	53–180	Stack	0.33		9.8E–08	3.9E–08		9.4E–05	6.2E–05		
1A	5–10	Stack	0.10	0.15	3.4E–07	3.4E–07	3.1E–07	1.8E–07	1.1E–07		0.15
	10–30	Stack	0.62	0.79	8.3E–06	8.3E–06	8.4E–06	4.2E–06	3.4E–06		0.79
	30–53	Stack	0.42	0.06	4.3E–06	4.3E–06	4.3E–06	6.2E–06	4.3E–06		0.06
	72–192	Stack	0.33		3.4E–02	3.9E–08		9.4E–05	6.2E–05		
2	80–110	Stack	1.40	0.95	1.5E–04	1.7E–04	1.7E–04	8.5E–05	5.5E–05		0.95
	110–250	Stack	3.40	0.05	5.0E–02	1.9E–04	1.2E–04	2.7E–04	1.8E–04		0.05
	250–509	Stack	3.30		3.0E–02	8.5E–06	3.4E–06	1.2E–05	7.6E–06		
2A	80–110	Stack	1.40	0.95	1.5E–04	1.7E–04	1.7E–04	8.5E–05	5.5E–05		0.95
	110–250	Stack	3.40	0.05	5.0E–02	1.9E–04	1.2E–04	2.7E–04	1.8E–904		0.05
	250–310	Stack	3.30		1.6E–02	8.5E–06	3.4E–06	1.2E–05	7.6E–06		
	720–840	Stack	1.20		9.2E–01						
2B	80–110	Stack	1.40	0.95	1.5E–04	1.7E–04	1.7E–04	8.5E–05	5.5E–05		0.95
	110–250	Stack	3.40	0.05	5.0E–02	1.9E–04	1.2E–04	2.7E–04	1.8E–04		0.05
	250–310	Stack	3.30		1.6E–02	8.5E–06	3.4E–06	1.2E–05	7.6E–06		
	310–910	Stack	3.50		9.2E–01	7.1E–01	2.9E–01				
3	5–20	Stack	0.87	0.66	1.2E–05	2.3E–05	1.2E–05	1.2E–05	7.5E–06		0.66
	20–38	Stack	1.50	0.29	3.6E–05	6.0E–05	3.1E–05	2.2E–05	2.3E–05		0.29
	38–125	Stack	1.70	0.05	1.4E–04	9.7E–06	5.0E–06	1.6E–04	9.9E–95		0.05
	125–217	Stack	1.50		4.0E–05			4.6E–06	3.0E–06		

Table 4.4-3 (continued)

Release Category	Time Interval (minutes)	Release Location	Energy Release Rate (MW)	Noble Gas	Iodine	Cs	Te	Sr	Ba	Ru, La, Ce	Tritium
3A	10–30	Stack	0.87	0.66	1.2E–05	2.3E–05	1.2E–05	1.2E–05	7.5E–06		0.66
	20–38	Stack	1.50	0.29	3.6E–05	6.0E–05	3.1E–05	2.2E–05	2.3E–05		0.29
	38–217	Stack	1.70	0.05	1.4E–04	9.7E–06	5.0E–06	1.7E–04	1.0E–04		0.05
	720–840	Roof	0.36		2.8E–01						
4	10–30	Stack	1.20	0.79	4.8E–05	8.4E–05	8.3E–05	4.2E–05	1.0E–05	1.7E–05	0.79
	30–44	Stack	3.20	0.16	1.2E–03	5.6E–05	5.2E–05	4.6E–05	1.1E–05	1.8E–05	0.16
	44–120	Stack	3.10	0.05	2.0E–02	7.6E–05	3.8E–05	2.0E–04	5.2E–05	8.0E–05	0.05
	120–180	Stack	2.80		1.2E–02	9.0E–06	3.7E–06	1.9E–05	4.9E–06	7.7E–06	
4A	10–30	Stack	1.20	0.79	4.8E–05	8.4E–05	8.3E–05	4.2E–05	1.0E–05	1.7E–05	0.79
	30–44	Stack	3.20	0.16	1.1E–03	5.6E–05	5.2E–05	4.6E–05	1.1E–05	1.8E–05	0.16
	44–180	Stack	3.20	0.05	3.2E–02	8.5E–05	4.2E–05	2.2E–04	5.7E–05	8.8E–05	0.05
	720–840	Stack	3.00		6.3E–01						
4B	10–30	Stack	1.20	0.79	4.8E–05	8.4E–05	8.3E–05	4.2E–05	1.0E–05	1.7E–05	0.79
	30–44	Stack	3.20	0.16	1.2E–03	5.6E–05	5.2E–05	4.6E–05	1.1E–05	1.8E–05	0.16
	44–120	Stack	3.10	0.05	2.0E–02	7.6E–05	3.8E–05	2.0E–04	5.2E–05	8.0E–05	0.05
	120–720	Stack	3.50		6.4E–01	4.3E–01	1.7E–01	1.9E–05	4.9E–06	7.7E–06	
4C	10–30	Stack	1.20	0.79	4.8E–05	8.4E–05	8.3E–05	4.2E–05	1.0E–05	1.7E–05	0.79
	30–44	Stack	3.20	0.16	1.2E–03	5.6E–05	5.2E–05	2.4E–05	5.8E–06	9.3E–06	0.16
	44–120	Stack	3.00	0.05	2.0E–02	7.6E–05	3.8E–05	4.1E–06	1.1E–06	1.6E–06	0.05
	120–180	Stack	2.80		1.2E–02	9.0E–06	3.7E–06		4.9E–06		
4D	10–30	Stack	1.20	0.79	4.8E–05	8.4E–05	8.3E–05	4.2E–05	1.0E–05	1.7E–05	0.79
	30–44	Stack	3.20	0.16	1.2E–03	5.6E–05	5.2E–05	2.4E–05	5.8E–06	9.3E–06	0.16
	44–180	Stack	3.00	0.05	3.2E–02	8.5E–05	4.2E–05	4.1E–06	1.1E–06	1.6E–06	0.05
	720–840	Stack	3.00		6.3E–01						

Table 4.4-3 (continued)

Release Category	Time Interval (minutes)	Release Location	Energy Release Rate (MW)	Noble Gas	Iodine	Cs	Te	Sr	Ba	Ru, La, Ce	Tritium
5	80–110	Roof	0.68	0.71	4.9E–01	2.3E–01	2.3E–01	1.1E–01	7.3E–02		0.71
	110–162	Roof	0.52	0.24	2.7E–01	2.4E–01	2.0E–01	1.7E–01	1.1E–01		0.24
	162–250	Roof	0.11	0.05	1.3E–01	1.4E–01	7.5E–02	2.9E–01	1.9E–01		0.05
	250–285	Roof	0.12		5.0E–02	5.1E–02	3.1E–02	1.1E–01	4.1E–02		
5A	80–110	Roof	0.68	0.71	4.9E–01	2.3E–01	2.3E–01	1.1E–01	7.3E–02		0.71
	110–115	Roof	4.00	0.24	2.7E–01	2.4E–01	2.0E–01	1.8E–01	1.2E–01	1.0E–02	0.24
	762–850	Roof	0.11	0.05	1.3E–01	1.4E–01	7.5E–02	2.9E–01	1.9E–01		0.05
	850–885	Roof	0.12		5.0E–02	5.1E–02	3.1E–02	1.1E–01	4.1E–02		
6	113–187	Ground	0.00	0.00	1.1E–04	2.2E–04	2.2E–04	2.0E–03	1.3E–03		0.00
	187–188	Ground	8.80	0.01	2.4E–03	1.8E–03	1.6E–03	2.6E–02	1.7E–02		0.01
	234–235	Ground	6.20	0.01	7.0E–04	8.9E–04	5.2E–03	2.6E–02	1.7E–02		0.01
	235–835	Ground	0.01	0.98	5.0E–01	2.6E–03	1.6E–03	1.3E–02	3.5E–02		0.98
6A	113–187	Ground	0.00	0.00	1.1E–04	2.2E–04	2.2E–04	2.0E–03	1.3E–03		0.00
	187–188	Ground	8.80	0.01	2.4E–03	1.8E–03	1.6E–03	2.6E–02	1.7E–02		0.01
	234–235	Ground	6.20	0.01	7.0E–04	8.9E–04	5.2E–03	2.6E–02	1.7E–02		0.01
	235–835	Ground	0.01	0.98	5.0E–01	2.6E–03	1.6E–03	1.3E–02	3.5E–02		0.98
7	5–27	Roof	0.06	0.24	2.5E–03	2.3E–03	2.2E–03	1.1E–03	9.0E–04		0.24
	35–240	Roof	0.03	0.28	2.3E–03	2.3E–03	2.3E–03	2.3E–01	1.5E–01		0.28
	240–465	Roof	0.00	0.48	1.3E–05	1.2E–03	1.1E–03	8.4E–03	2.4E–03		0.48
7A	5–27	Roof	0.06	0.24	2.5E–03	2.3E–03	2.2E–03	1.1E–03	9.0E–04		0.24
	35–240	Roof	0.03	0.28	2.3E–03	2.3E–03	2.3E–03	1.2E–03	1.0E–03		0.28
	240–465	Roof	0.00	0.48	1.3E–05	1.2E–03	1.1E–03		2.0E–04		0.48
8	5–30	Ground	0.05	0.13	1.7E–03	1.7E–03	1.7E–03	8.4E–04	6.8E–04		0.13
	30–101	Roof	0.00	0.00	8.9E–06	6.8E–06	7.6E–06	6.8E–04	4.5E–04		0.00
	101–120	Roof	1.20	0.02	1.6E–04	1.5E–04	1.4E–04	6.4E–02	4.3E–02		0.03
	120–720	Roof	0.10	0.85	5.9E–03	5.9E–03	5.9E–03	1.7E–01	1.1E–01		0.84

Table 4.4-3 (continued)

Release Category	Time Interval (minutes)	Release Location	Energy Release Rate (MW)	Noble Gas	Iodine	Cs	Te	Sr	Ba	Ru, La, Ce	Tritium
8A	5–30	Ground	0.05	0.13	1.7E–03	1.7E–03	1.7E–03	8.4E–04	6.8E–04		0.13
	30–101	Roof	0.00	0.00	8.9E–06	6.8E–06	7.6E–06	3.3E–06	2.8E–06		0.00
	101–120	Roof	1.20	0.02	1.6E–04	1.5E–04	1.4E–04	7.2E–05	6.0E–05		0.03
	120–720	Roof	0.10	0.85	5.9E–03	5.9E–03	5.9E–03	2.9E–03	2.4E–03		0.84
9	5–10	Stack	0.10	0.15	3.4E–07	3.4E–07	3.1E–07	1.7E–07	1.1E–07		0.15
	10–15	Ground	29.00	0.80	1.7E–03	1.1E–03	1.1E–03	1.3E–03	2.2E–03	7.9E–03	0.80
	15–80	Roof	3.00	0.05	2.5E–02	9.6E–03	9.6E–04	5.7E–02	3.1E–02	9.1E–04	0.05
	80–150	Roof	2.80		8.5E–03	6.5E–03	6.4E–03	7.1E–02	4.9E–02	1.6E–05	
9A	5–10	Stack	0.10	0.15	3.4E–07	3.4E–07	3.1E–07	1.7E–07	1.1E–07		0.15
	10–15	Ground	29.00	0.80	1.7E–03	1.1E–03	1.1E–03	1.3E–03	2.2E–03	7.9E–03	0.80
	15–80	Roof	3.00	0.05	2.5E–02	9.6E–03	9.6E–04	0.0E+00		9.1E–04	0.05
	80–150	Roof	2.80		8.5E–03	6.5E–03	6.4E–03	0.0E+00		1.6E–05	
10	80–110	Stack	1.70	0.95	2.2E–04	2.2E–04	2.2E–04	1.0E–04	2.6E–05		0.95
	110–115	Ground	29.00	0.01	2.1E–02	9.1E–03	9.1E–03	1.2E–02	1.6E–02	7.9E–03	0.01
	115–180	Roof	3.00	0.04	6.2E–01	3.3E–01	1.4E–01	2.1E–01	1.1E–01	9.1E–04	0.04
	180–250	Roof	2.50		1.0E–01	1.1E–01	5.2E–02	2.6E–01	1.8E–01	1.6E–05	
10A	80–110	Stack	1.70	0.95	2.2E–04	2.2E–04	2.2E–04	1.0E–04	2.6E–05		0.95
	110–115	Ground	29.00	0.01	2.1E–02	9.1E–03	9.1E–03	1.2E–02	1.6E–02	7.9E–03	0.01
	115–180	Roof	3.00	0.04	6.2E–01	3.3E–01	1.4E–01	3.4E–03	1.8E–03	9.1E–04	0.04
	180–250	Roof	2.50		1.0E–01	1.1E–01	5.2E–02	1.7E–03	1.2E–03	1.6E–05	
11	10–30	Stack	1.20	0.79	4.8E–05	8.4E–05	8.3E–05	4.2E–05	2.6E–05		0.79
	30–35	Ground	29.00	0.17	2.2E–02	9.2E–03	9.2E–03	1.2E–02	1.6E–02	7.9E–03	0.17
	35–100	Roof	3.00	0.04	5.2E–01	2.4E–01	1.3E–01	1.7E–01	9.0E–02	9.1E–04	0.04
	100–170	Roof	2.80		1.0E–01	1.1E–01	5.2E–02	2.1E–01	1.4E–01	1.6E–05	

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Table 4.4-3 (continued)

Release Category	Time Interval (minutes)	Release Location	Energy Release Rate (MW)	Noble Gas	Iodine	Cs	Te	Sr	Ba	Ru, La, Ce	Tritium
11A	10-30	Stack	1.20	0.79	4.8E-05	8.4E-05	8.3E-05	4.2E-05	2.6E-05		0.79
	30-35	Ground	29.00	0.17	2.2E-02	9.2E-03	9.2E-03	1.2E-02	1.6E-02	7.9E-03	0.17
	35-100	Roof	3.00	0.04	5.2E-01	2.4E-01	1.3E-01			9.1E-04	0.04
	100-170	Roof	2.80		1.0E-01	1.1E-01	5.2E-02			1.6E-05	

Table 4.4–4

Mixing Layer Height Data Used In The Consequence Analysis

Grid	Seasonal Mixing Layer – height above ground, m			
	Winter	Spring	Summer	Fall
Offsite (afternoon):	1020	1700	1800	1400
Onsite (average of morning and afternoon data)	710	1050	1100	850

Table 4.4–5
Offsite Consequence Mitigation Plan by Release Category

Release Category	(Miles from K Reactor)	(Hours after release)	mph
1	0 – 20	2	2.5
1A	0 – 20	2	2.5
2	0 – 20	2	2.5
2A	0 – 20	2	2.5
2B	0 – 20	2	2.5
3	0 – 20	2	2.5
3A	0 – 20	2	2.5
4	0 – 20	2	2.5
4A–4D	0 – 20	2	2.5
5	0 – 20	4	1.25
5A	0 – 20	4	1.25
6	0 – 20	2	2.5
6A	0 – 20	2	2.5
7	0 – 20	4	1.25
7A	0 – 20	4	1.25
8	0 – 20	2	2.5
8A	0 – 20	2	2.5
9	0 – 20	2	2.5
9A	0 – 20	2	2.5
10	0 – 20	2	2.5
10A	0 – 20	2	2.5
11	0 – 20	2	2.5
11A	0 – 20	2	2.5

Shielding Factors Applicable During Evacuation:	Cloudshine Protection Factor	1.00
	Inhalation Protection Factor	1.00
	Skin Protection Factor	1.00
	Groundshine Protection Factor	0.50
Shielding Factors Applicable During Sheltering:	Cloudshine Protection Factor	0.50
	Inhalation Protection Factor	0.50
	Skin Protection Factor	0.50
	Groundshine Protection Factor	0.33
Shielding Factors Applicable During Normal: Activity in Sheltering and Evacuation Zone	Cloudshine Protection Factor	0.75
	Inhalation Protection Factor	0.75
	Skin Protection Factor	0.75
	Groundshine Protection Factor	0.33

Table 4.4–6

Onsite Consequence Mitigation Plan by Release Category

Release Category	Onsite Evacuation & Sheltering Plan: Radius of Applicability (Miles from K Reactor)	Start Time (Hours after release)	Radial Evac. Velocity (mph)
1,1A	0 - 5	0.5	8.5
	5 - 20	Not Sheltered	
2,2A	0 - 5	0.5	8.5
	5 - 20	Not Sheltered	
2B	0 - 5	0.5	8.5
	5 - 20	Not Sheltered	
2B (Sensitivity Study)	0 - 5	0.5	8.5
	5 - 20	1.0	Sheltered for 12 hrs.
3,3A	0 - 5	0.5	8.5
	5 - 20	Not Sheltered	
4–4D	0 - 5	0.5	8.5
	5 - 20	Not Sheltered	
5	0 - 5	1.0	4.25
	5 - 20	Not Sheltered	
5 (Sensitivity Study)	0 - 5	1.0	4.25
	5 - 20	2.0	Sheltered for 12 hrs.
5A	0 - 5	1.0	4.25
	5 - 20	Not Sheltered	
6,6A	0 - 5	0.5	8.5
	5 - 20	Not Sheltered	
7,7A	0 - 5	1.0	4.25
	5 - 20	Not Sheltered	
8,8A	0 - 5	0.5	8.5
	5 - 20	Not Sheltered	
9,9A	0 - 5	0.5	8.5
	5 - 20	Not Sheltered	
10,10A	0 - 5	0.5	8.5
	5 - 20	Not Sheltered	
11,11A	0 - 5	0.5	8.5
	5 - 20	Not Sheltered	

Shielding Factors Applicable During Evacuation:	Cloudshine Protection Factor	1.00
	Inhalation Protection Factor	1.00
	Skin Protection Factor	1.00
	Groundshine Protection Factor	0.50
Shielding Factors Applicable During Sheltering:	Cloudshine Protection Factor	0.50
	Inhalation Protection Factor	0.50
	Skin Protection Factor	0.50
	Groundshine Protection Factor	0.33
Shielding Factors Applicable During Normal: Activity in Sheltering and Evacuation Zone:	Cloudshine Protection Factor	0.75
	Inhalation Protection Factor	0.75
	Skin Protection Factor	0.75
	Groundshine Protection Factor	0.33

Table 4.4-7

Mean Value of "Near Field" Offsite Consequences

	Number of Prompt Fatalities within 1 mile of the plant boundary (per year)	Number of Prompt Fatalities within 10 miles of the plant boundary (per year)	Number of Latent (Cancer) Fatalities with 10 miles of the plant boundary (per year)	Risk of Prompt Fatalities within 1 mile of the plant boundary (per year)	Risk of Prompt Fatalities with in 10 miles of the plant boundary (per year)	Risk of Latent (Cancer) Fatalities within 10 miles of the plant boundary (per year)
Internal Events	5.12E-08	7.02E-08	1.09E-03	1.29E-11	2.47E-12	1.98E-08
External (seismic) Events	5.20E-07	7.56E-07	1.23E-03	4.59E-11	2.65E-11	1.43E-08
Total	5.71E-07	8.26E-07	2.32E-03	5.88E-11	2.90E-11	3.42E-08

Table 4.4-8

Mean Values of Onsite Consequences

	Number of Prompt Fatalities within 1 mile of K-Reactor (per year)	Number of Prompt Fatalities for the Full SRS Population	Number of Latent (Cancer) Fatalities with 10 miles of K-Reactor	Number of Latent (Cancer) Fatalities for the Full SRS Population	Total Dose for Full SRS Population	Risk of Prompt Fatalities within 1 mile of K-Reactor	Risk of Latent (Cancer) Fatalities within 10 miles of K-Reactor
Internal Events	8.28E-05	2.15E-04	7.44E-04	7.44E-04	1.38E-02	1.53E-07	9.29E-08
External (seismic) Events	1.67E-06	1.83E-04	3.59E-04	3.59E-04	1.11E-02	3.08E-09	4.48E-08
Total	8.44E-05	3.98E-04	1.10E-03	1.10E-03	2.49E-02	1.56E-07	1.38E-07

Table 4.4-9

Comparison to DOE Safety Goals

Safety Goal Description	Safety Goal Value	SR Reactor Value
Prompt Fatality Risk to an Offsite Individual within One Mile of the Site Boundary	4.0×10^{-7}	5.8×10^{-11}
Cancer Fatality Risk to an Offsite Individual within 10 Miles of the Site Boundary	2.0×10^{-6}	3.4×10^{-8}
Prompt Fatality Risk to an Onsite Worker within One Mile of K Reactor	1.0×10^{-6}	1.6×10^{-7}
Cancer Fatality Risk to an Onsite Worker within 10 Miles of K Reactor	2.0×10^{-6}	1.4×10^{-7}

Table 4.4-10
Onsite Conditional Consequences for Several Emergency Response Plans

RC-2B LOHSA with Filter Burn

	Prompt Fatalities (0-1 mile)	Total Site Prompt Fatalities	Total Latent Cancer Death	Total Population Dose, (Sv)
No Evac./No Sheltering	55	103	46	1570
Base Case	—	5.9	18	413
w/Evac. and Sheltering	—	0	3	36

RC-5 Major Seismic Event w/LOHSA

	Prompt Fatalities (0-1 mile)	Total Site Prompt Fatalities	Total Latent Cancer Death	Total Population Dose, (Sv)
No Evac./No Sheltering	52	203	52	7.3E+3
Base Case	0.26	59	29	1.1E+3
w/Evac. and Sheltering	0.26	1.8	14	3.5E+2

Figure 4.4-1

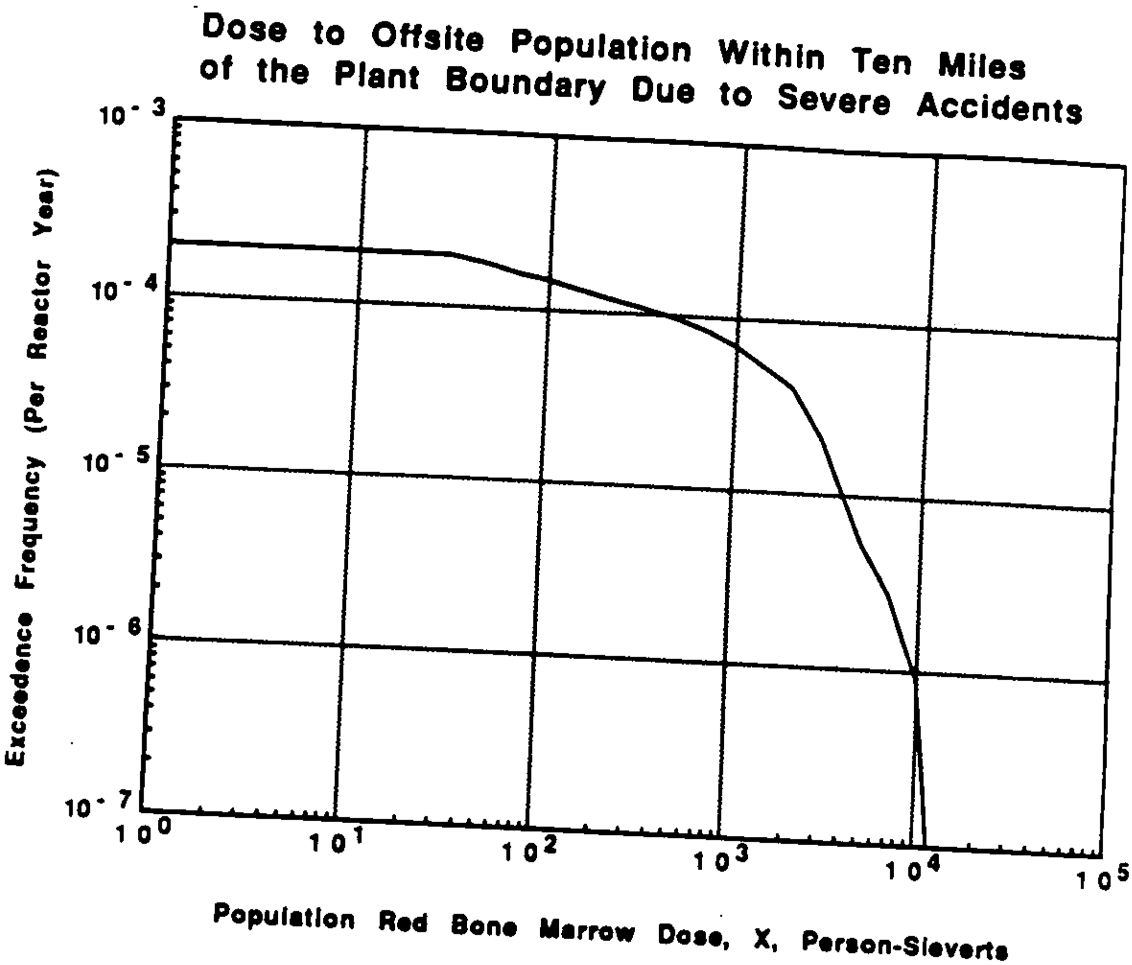

Dose to Offsite Population Within Ten Miles of the Plant Boundary Due to Severe Accidents
10^{-3}
10^{-4}
10^{-5}
10^{-6}
10^{-7}
Exceedence Frequency (Per Reactor Year)
10^{0}
10^{1}
10^{2}
10^{3}
10^{4}
10^{5}
Population Red Bone Marrow Dose, X, Person-Sieverts

Figure 4.4-2

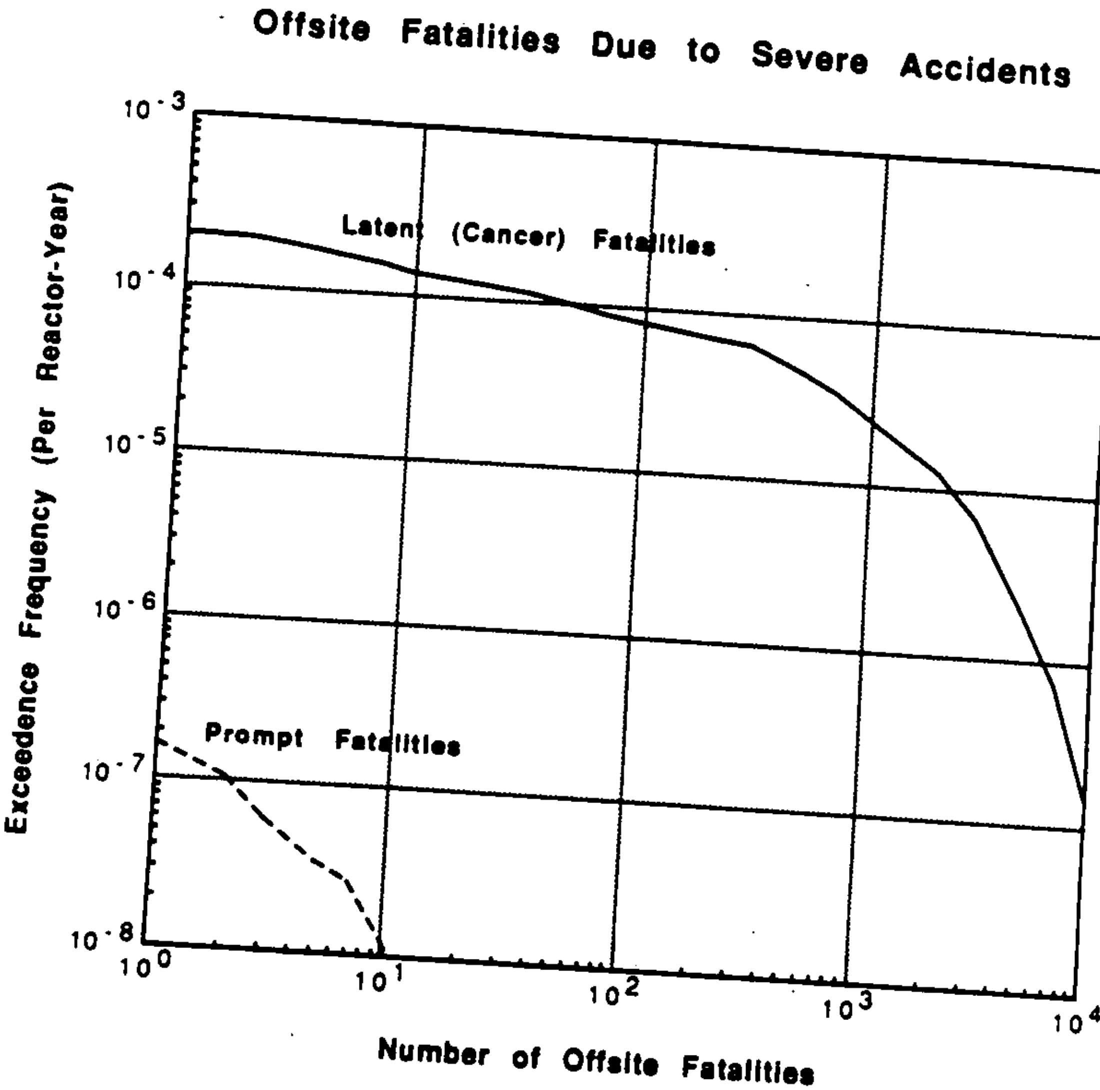
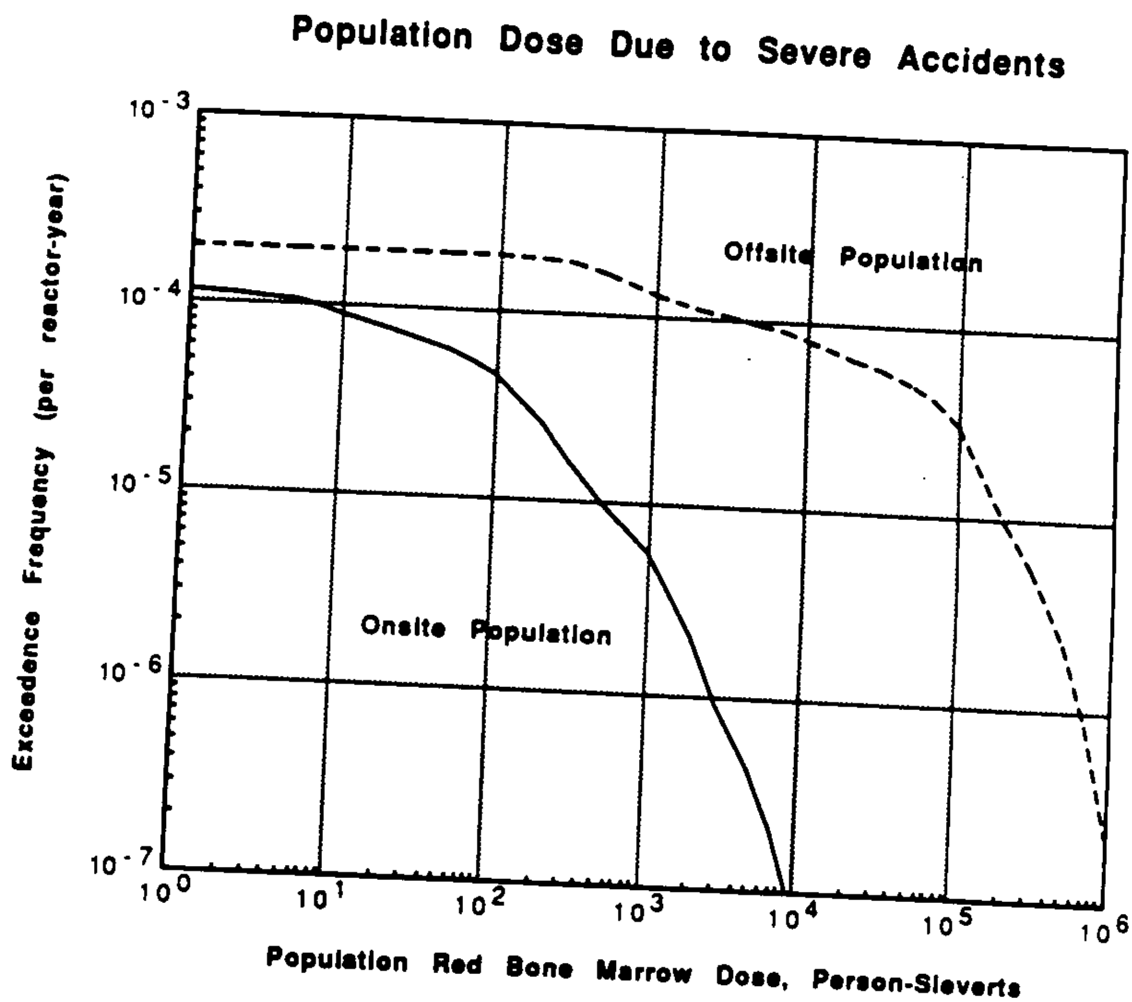

Offsite Fatalities Due to Severe Accidents
Latent (Cancer) Fatalities
Prompt Fatalities
Exceedence Frequency (Per Reactor-Year)
Number of Offsite Fatalities
10^{-3}
10^{-4}
10^{-5}
10^{-6}
10^{-7}
10^{-8}
10^{0}
10^{1}
10^{2}
10^{3}
10^{4}

Figure 4.4-3



Population Dose Due to Severe Accidents
Offsite Population
Onsite Population
Exceedence Frequency (per reactor-year)
Population Red Bone Marrow Dose, Person-Sieverts

Figure 4.4-4

Onsite Fatalities Due to Severe Accidents
Base Case Evacuation Scenario

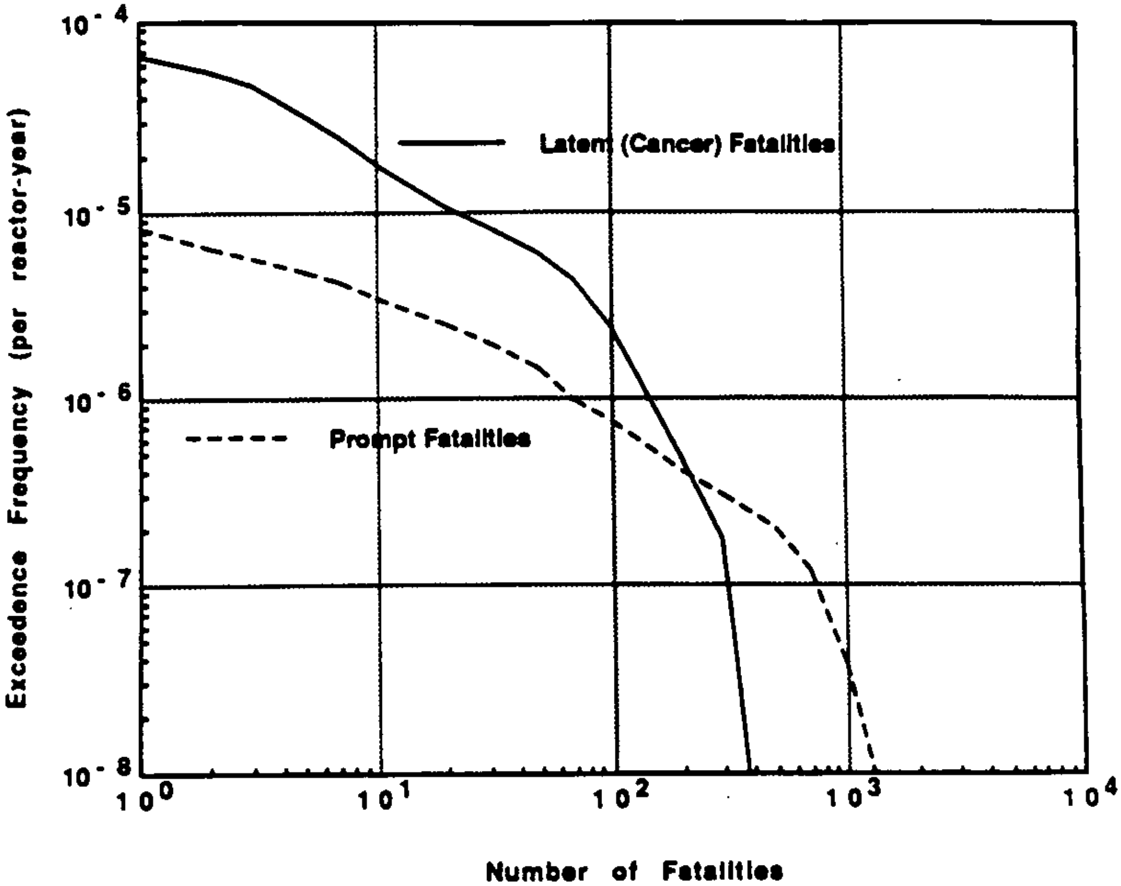


Figure 4.4-4

The onsite population outside the 5 mile radius is not evacuated or sheltered. The cross-over in the number of prompt and latent fatalities is due to low probability meteorological conditions delivering lethal doses to non-evacuees. RC-5 and RC-9 shown signs of this cross-over at low probability conditions.

Figure 4.4-5

Comparison of Latent Fatality Risk
P-Reactor vs. K-Reactor

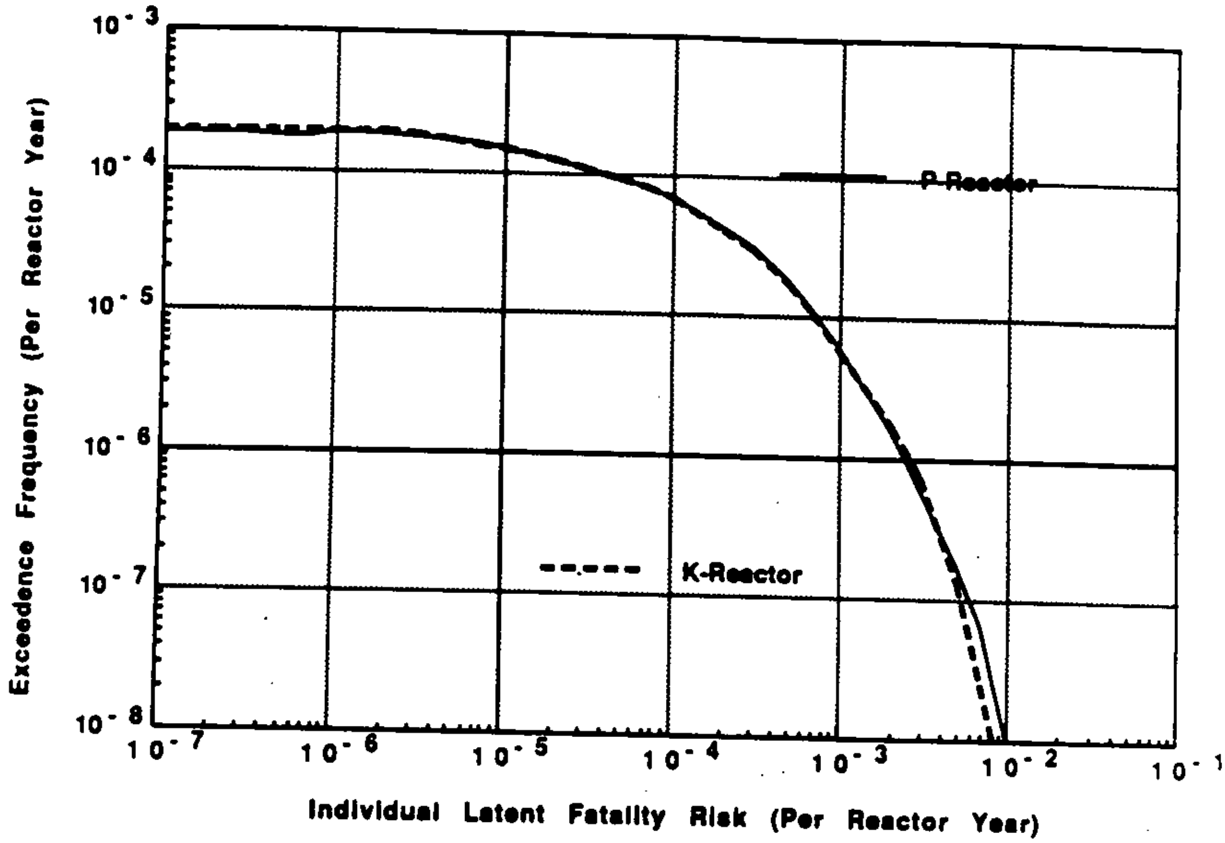


Figure 4.4-6

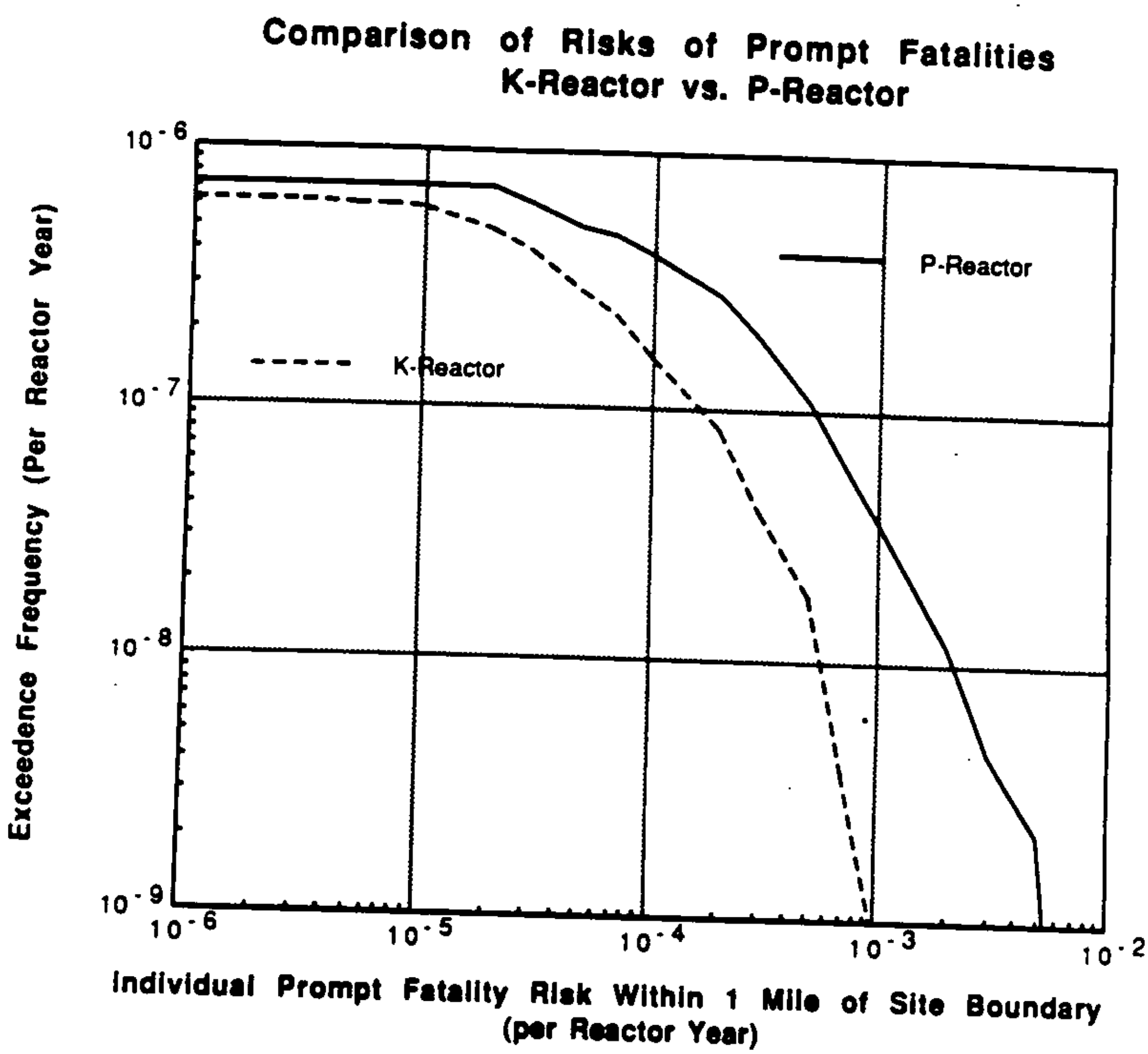
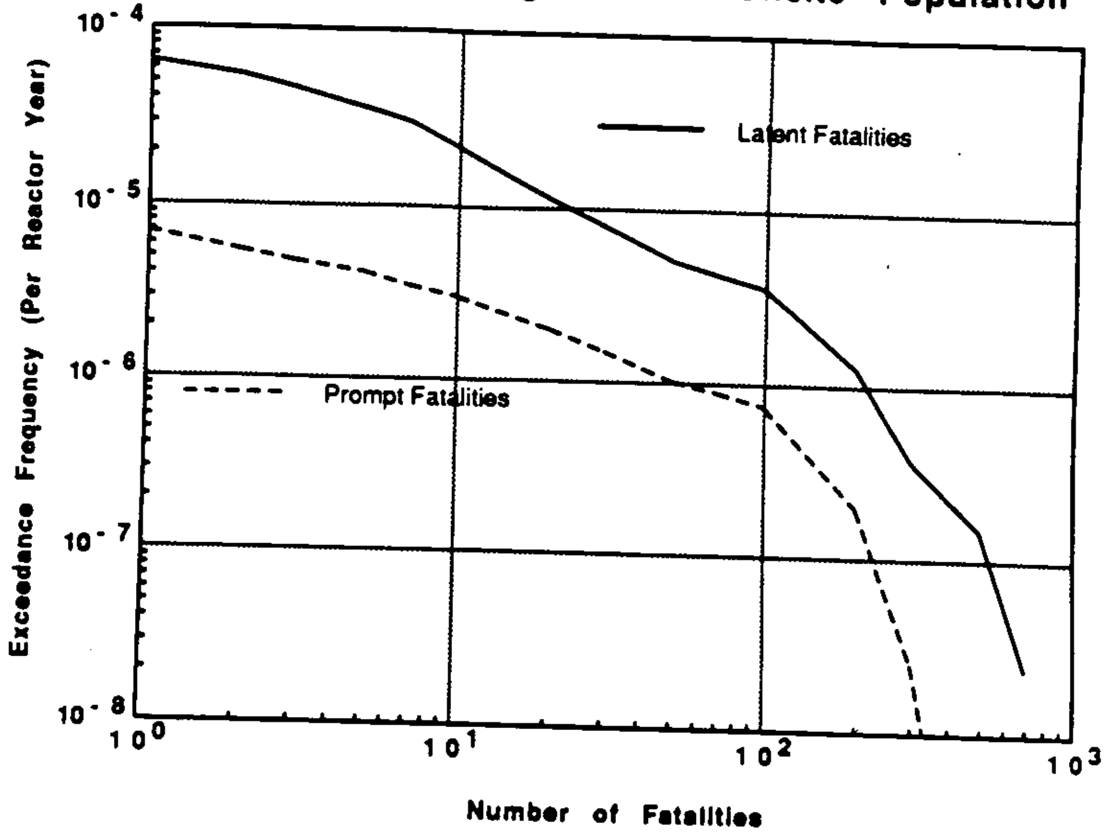

Comparison of Risks of Prompt Fatalities
K-Reactor vs. P-Reactor
P-Reactor
K-Reactor
Exceedence Frequency (Per Reactor Year)
10^{-6}
10^{-7}
10^{-8}
10^{-9}
10^{-6}
10^{-5}
10^{-4}
10^{-3}
10^{-2}
Individual Prompt Fatality Risk Within 1 Mile of Site Boundary
(per Reactor Year)

Figure 4.4-7

Onsite Fatalities Due To Severe Accidents
Evacuation/Sheltering of Full Onsite Population



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CHAPTER 5.0
OPERATING CONTRACTOR ORGANIZATION AND EMERGENCY PLANNING
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5.1 OVERVIEW OF OPERATIONS

DOE operates SRS through the services of an operating contractor, typically a large industrial company with demonstrated capability to properly manage complex process facilities. The current operating contractor is Westinghouse Savannah River Company (WSRC), a subsidiary of Westinghouse Electric Corporation.

DOE places firm requirements on the contractor's organization to ensure effective operation of SRS facilities. For example, DOE requires a strong reactor operation division with line responsibility for operating the plant safely.

DOE also requires that the contractor maintain independent organizations for oversight in the safety and quality assurance functions. Further, the contractor works with an advisory group of outside experts, called the Reactor Safety Advisory Committee.

The methods and requirements for managing the reactors safety are prescribed in the contractor's formal administrative procedures. The following discussion of current WSRC procedural requirements illustrates the general scope and approach of administrative control. The details of the administrative controls are, of course, subject to change by WSRC, and/or changed dramatically, if WSRC is replaced by another operating contractor.

5.2 MANAGEMENT POLICIES

The administrative procedures for the SRS reactors are prescribed to ensure that basic and important decisions are made after review throughout the organization and that decisions that could significantly affect safety or operability are reviewed by a competent technical organization other than the one charged with direct responsibility for the operation.

Management policies are prescribed in Corporate level procedures. The following policies describe the bases for the current WSRC system of controls.

- Decisions that have significant safety or production implications shall be reviewed by several levels in the organizations, including those not directly responsible for operation, before consideration and final approval by management. Safety-related operations shall be carried out by detailed written procedures approved in advance.
- The overall activity shall be considered as a system involving the interaction of process, people, facilities, and procedures.
- Activities that have reactor safety significance shall have the review and input of the operating organization and the advisory organizations that are independent of reactor operation.
- Hierarchies of administrative control shall correspond reasonably with organizational structures, i.e., detailed procedures originate at the operational level, and concepts and principles originate at managerial levels. Authority shall be delegated throughout the organization to match functional needs, but higher management shall retain full responsibility. Achievement of this objective requires efficient communications vertically and horizontally within the organization.
- After safe boundaries for operation are carefully defined, they shall be approved by management with full recognition and acceptance of residual risks. This process

shall be well documented, and the acceptable safe boundaries communicated to affected parties.

- The principles of defense-in-depth and redundancy shall be applied as necessary to the operation including people, process, and equipment. The intent is not only to prevent accidents but to make provisions to limit consequences should accidents occur.

5.3 WESTINGHOUSE SAVANNAH RIVER COMPANY ORGANIZATION AND RESPONSIBILITIES

The Westinghouse Savannah River Company (WSRC) is a full-scope, self-sufficient, site management and operations organization designed to accomplish the DOE-SR production mission efficiently and cost-effectively while ensuring safe, secure, and environmentally sound operations.

A single, onsite executive, the WSRC President, has the responsibility and accountability for managing WSRC.

The major functions of the SRS are assigned to divisions, each under the direction of a Vice President. The functions of each division which is involved in reactor programs are described in the following sections.

5.3.1 Reactor Restart Division

The Reactor Restart Division has the responsibility to restart and operate the SRS reactors in a way that is safe, timely, and responsive to the requirements of the Department of Energy, the long-term interests of national security, and the health and well-being of site employees, the environment, and the general public.

5.3.1.1 Reactor Operations

Reactor Operations has the principal line management responsibility for the operation of the reactors in accordance with Technical Specifications. It controls physical changes in plant configuration and coordinates the activities of all work groups in the reactor areas. Additional description of reactor operating staffs are given in Section 5.2.7.

5.3.1.2 Reactor Outage and Maintenance

Reactor Outage and Maintenance Department controls outage maintenance, preventive and corrective maintenance, and coordinates restart outage work.

5.3.1.3 Reactor Engineering

Reactor Engineering Department provides direct engineering and technical support to reactor operations. It also ensures operating limits are within safety analysis, develops basic data, and provides liaison support for reactor projects and configuration management

5.3.1.4 Planning/Scheduling and Cost Control

This department consolidates all existing scheduling activities, provides integrated planning tools, and provides tracking for manpower-related costs.

5.3.1.5 Reactor Training and Procedures

This department provides operation and maintenance training, writes operation and maintenance procedures, and conducts reactor simulator training.

5.3.1.6 Technical Department

The Technical Department is responsible for the Issue Management Program and for managing-to-closure the significant technical and policy issues that affect the reactor.

5.3.1.7 Reactor Assessments

Reactor Assessments is responsible for providing line-organization-directed quality assurance, quality control, environmental protection coordination, critical self-assessment, and operational readiness reviews.

5.3.1.8 Reactor Strategic Programs

Reactor Strategic Programs is responsible for those forward-looking, longer-term efforts needed for improved operations in the reactor areas. This includes, but is not limited to, the strategic planning process and programs such as the Reactor Safety Improvement Program (RSIP) and the Model Utilities Program.

5.3.2 Savannah River Laboratory

The Savannah River Laboratory (SRL) has the mission of providing expert technical support to Reactor Restart Division to ensure that safe, high-quality, cost-effective, and environmentally sound technologies are used during present and future operation of the SRS. SRL has two reactor programs-related departments, Nuclear Reactor Technology and Scientific Computations (NRT&SC) and Safety Analysis and Risk Management (SA&RM). Their functions are described below.

5.3.2.1 Nuclear Reactor Technology and Scientific Computations

The Nuclear Reactor Technology and Scientific Computations (NRT&SC) Department of the SRL supports reactor and nuclear engineering technology, develops new process technology for reactor target and fuel fabrication processes, and develops scientific computational methods. NRT&SC provides technical support in areas of reactor charge design, reactor physics, heat transfer and hydraulics, advanced reactor design, criticality safety, and analytical computer code development. NRT&SC also reviews charge designs. For reactor charges that are significantly different from previous operation. NRT&SC also monitors reactor startup and initial reactor operation.

5.3.2.2 Safety Analysis and Risk Management

The Safety Analysis and Risk Management (SA&RM) Department conducts safety analyses for reactor facilities. SA&RM identifies those areas where the safety of reactor operations can be improved. The SA&RM Department is responsible for the development and administration of safety analysis reports and other safety control documents. Review of technical manuals, systems analyses, Test Authorizations, and Reactor Startup Authorizations is also provided. Safety analyses are conducted using both probabilistic and deterministic methods.

5.3.3 Engineering and Project Division

The mission of the Engineering and Project (E&P) Division is to design and construct new facilities and modification projects in a safe, secure, cost-effective, environmentally sound, and quality manner. The division provides engineering and project management service and is also responsible for technical direction of the principal subcontractor, Bechtel Savannah River, Inc., and any other design, engineering, and construction services subcontractors which are retained by WSRC.

5.3.4 Environment, Safety, Health, and Quality Assurance Division

The Environment, Safety, Health, and Quality Assurance (ESH&QA) Division provides radiation protection resources, emergency planning and emergency operations, environmental support, fire protection services, security, and independent safety and quality assurance oversight. In addition, ESH&QA is responsible for developing performance objectives and acceptance criteria based on technical assessments of DOE policies and orders. ESH&QA has five reactor-related departments: Quality Assurance (QA) Department; Safety Department; Health Protection Department; the Environmental Department; the Safeguards and Security Department. Their functions are described below.

5.3.4.1 Quality Assurance Department

The Quality Assurance Department (QAD) establishes the overall requirement of the Quality Assurance Program applicable to SRS Reactors and support activities through development of the SRS QA Plan and QA Manual. QAD is responsible for verifying that an adequate QA Program is properly implemented through the performance of audits and documentation review. Additionally, QAD provides training in quality practices and procedures, training and certification of independent inspection personnel, supplier QA capability, and the performance of source surveillance and receiving inspection.

5.3.4.2 Safety Department

The Safety Department provides independent oversight of reactor activities and independent oversight of the programs for managing these activities with the goal of assuring and enhancing reactor safety and reliability. The Safety Department assesses DOE policies and orders and develops corresponding performance objectives and acceptance criteria.

5.3.4.3 Health Protection

The Health Protection Department provides organizational assistance and training in health protection matters. It also provides employee radiation protection as well as assuring radiation safety of new or modified equipment, processes, or facilities as needed.

5.3.4.4 Environmental

The Environmental Department provides organizational assistance and training in environmental matters. It also provides environmental assessments of new or modified equipment, processes, or facilities as needed. It coordinates environmental studies as related to the SRS and maintains liaison for environmental matters with DOE-SR, other DOE-SR contractors, other government agencies, and other envirionmental study groups.

5.3.4.5 Safeguards and Security

The Safeguards and Security Department is responsible for physical security of the site. The department reviews accountability records for nuclear material and controls for nuclear material. In addition to maintaining physical security systems and planning new security systems, this department prepares plans for emergency situations, such as evacuation plans. This department ensures proper classification and control of documents on the site.

5.3.5 Reactor Operating Staff

The Reactor Operations Department (ROD) operates the reactors and associated equipment in accordance with approved Technical Specifications, Mechanical Standards, and operating procedures under the direction of the department manager who is responsible for department-wide operations.

Area Managers have the overall responsibility for day-to-day reactor operation in their respective areas (P, L, or K).

The operating area organizations are divided into two groups. One area group is the operation shift crew. Each shift crew is headed by a certified Shift Manager. This group is also responsible for operations, environmental protection, and coordination of auxiliary systems.

The second area group is responsible for coordination of area personnel, training, preparation of job plans for maintenance work, planning shutdown work, and scheduling preventive maintenance.

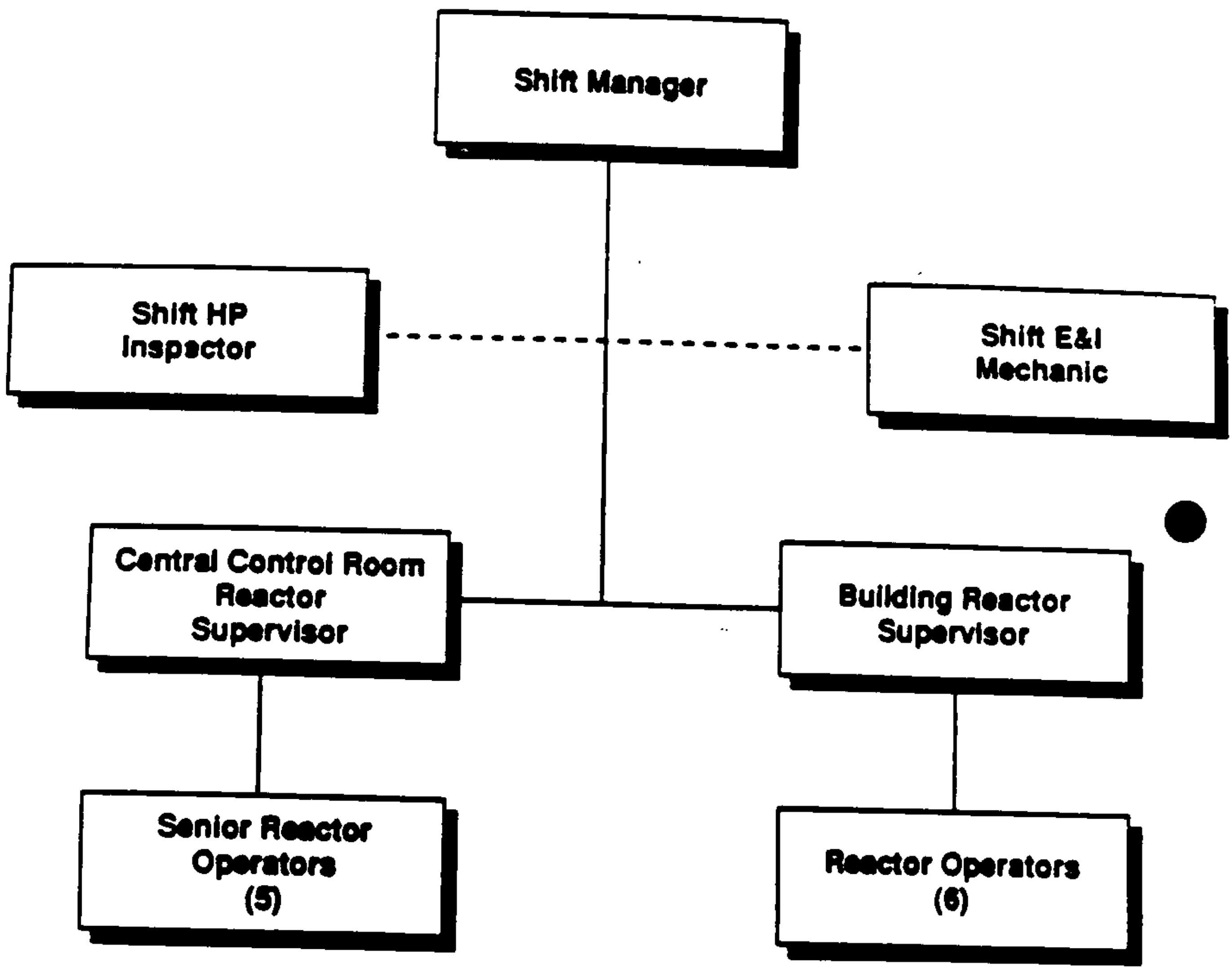

Figure 5-1

Reactor Shift Organization

5.4 PROCEDURAL SYSTEM OF CONTROL DOCUMENTS

5.4.1 System Approach

A formalized system is employed at Westinghouse Savannah River Company (WSRC) to ensure the reactors are operated and maintained as prescribed both by the Technical Specifications and WSRC management policies.

The control point in the system is a set of Technical Standards which prescribe limits on requirements for the basic variables within which the process must be operated. These Technical Standards constitute the bases for Categories 1–4 Technical Specifications. Operating manuals and procedures of varying degrees of detail are designed to ensure that limits in the Technical Standards are not exceeded. These manuals and procedures are usually written to allow some margin of safety between the operating limits and the limits specified in the Technical Standards. The necessary flexibility of the system is obtained by means of Test Authorizations intended to set the basic controls for tests of process changes or other operations. Technical Standards specify the limits for control of incidents which may be associated with the equipment or process as discussed in the Safety Analysis Reports (SARs) or Technical Manuals.

5.4.2 Relationship of Controls

The SRS reactor control documents are those which prescribe nuclear safety control in the reactor areas. The relationship is not strictly hierarchal but, in general, the ordering is in accordance with the control significance of the documents and levels of approval.

The SAR, Technical Specifications, and Test Authorizations are approved by DOE as well as WSRC. Some reactor startups must also be approved by DOE. Modifications to these documents also must receive DOE approval.

Control documents are prepared, reviewed, and approved by personnel in Reactor Restart Division, SRL, E&P, and ESH&QA. The preparation is usually on a lower level, but the review of a control document usually spans many levels within a division of WSRC. All four divisions review and concur on a control document, in accord with Management Procedures and Requirements.

5.5 EVENT INVESTIGATIONS AND REPORTS

All events in the reactor areas which fall under the definitions of DOE requirements are investigated and reported as required by the DOE.

Events are reported as Reactor Event Reports or as a result of the Issues Management Program. The Reactor Restart Division, in consultation with others, as necessary, determines when operation outside Technical Standards or Technical Specifications has occurred. Such determinations are made on the basis of evidence that implies the limits or requirements of Technical Standards or Technical Specifications were not met.

5.6 AUDITING, INSPECTION, AND REVIEW

The Reactor Restart Division maintains a system of inspection, auditing, and surveillance. Operating and Maintenance Procedures are audited and the facilities are surveyed for adherence to procedures and adequacy of procedures and for indications of abnormal behavior in the process, equipment, and personnel performance. A system is maintained to provide internal review of safety considerations by competent technical personnel other than those having direct responsibility for the subject being reviewed.

An integral part of the internal review system includes (1) a semiannual review of reactor operations by WSRC staff, and (2) an independent annual review of the adequacy of the training program for Certified Reactor Supervisors/Managers and Senior Reactor Operators by the Reactor Restart Division. Furthermore, ESH&QA Division provides at least annually an independent overview of the safety of operation of the Savannah River Site production reactors. The internal review system is evaluated every three years for adequacy of performance by WSRC Staff Management. Auditing, inspection, and review records are available for examination by DOE on request.

5.7 TRAINING

Initial training, continuous training, and retraining of Certified Reactor Supervisors/Managers and Senior Reactor Operators are carried out by formal classroom instruction, simulator training, computer assisted instruction, and on-the-job experience. Initial certification of Reactor Supervisors and Senior Reactor Operators is made by higher supervision based on a demonstrated acceptable level of competence and performance. Initial certification depends on satisfactory completion of comprehensive written, oral, and operating examinations; satisfactory physical condition; general health; and higher supervision's judgment of general qualifications. In addition, those certified for control room duty are evaluated by a Certification Board composed of members from the ESH&QA Division and Reactor Restart Division.

Performance-based training, as defined in the Department of Energy Training Accreditation Program (TAP) documents, will be used for the design and implementation of all reactor training. Continuous training and reexamination on procedures for handling abnormal reactor conditions is done annually, and biennially for other procedures important to the safe operation of the reactors. Recertification is biennial and depends in part on the satisfactory completion of written, oral, and operating examinations on both normal and abnormal operating conditions. The bases for both initial certification and recertification are documented. This documentation includes a copy of the most recent test results and test grades. Documentation on testing is available to DOE on request. An independent review of the adequacy of the training program is made annually.

The policy and overall content of the training program for personnel involved in the operation of Savannah River Site reactors are reviewed by WSRC Staff. A summary of the content of the training program is distributed to DOE for approval prior to implementation.

5.8 PLANT OPERATING RECORDS

Records retention practices are in accordance with DOE requirements and the Savannah River Site Quality Assurance Plan.

5.9 QUALITY ASSURANCE

The Quality Assurance (QA) program for Savannah River Site activities satisfies DOE requirements. Application of this policy is implemented through written procedures and instructions contained in the Savannah River Site Quality Assurance Plan and in derivative documents.

The Savannah River Site QA plan is applicable to operation, research, development, and design. The QA plan requires that sufficient records be maintained to preserve design basis test and experimental results including computational data used to derive safety limits, limiting safety system settings, limiting conditions for operation, and design features for operation.

The Quality Assurance Policy and Manual and changes thereto are approved by WSRC Staff Management and DOE prior to implementation.

5.10 EMERGENCY PLANNING

This section provides an overview of the Savannah River Site emergency preparedness program.

5.10.1 Description of Activities

SRS was constructed during the early 1950s to produce materials for the nation's nuclear weapons program. SRS facilities include three operating nuclear materials production reactors (P, L, and K).

Major support facilities include the main administration area, Savannah River Laboratory (SRL), two coal-fired power plants, a heavy water rework operation (TNX), and the Savannah River Ecology Laboratory (SREL).

5.10.2 Description of SRS and Nearby Areas

SRS is a government-owned, contractor-operated reservation, which occupies a geographical area of approximately 300 square miles (192,000 acres) in western South Carolina, 25 miles southeast of Augusta, Georgia. The site encompasses portions of three South Carolina counties (Aiken, Barnwell, and Allendale), and borders the Savannah River for approximately 17 miles. Public access to the site is restricted except for controlled through-traffic on South Carolina Highway 125, a CSX Distribution Services rail right-of-way, and seasonal organized hunting. Across the Savannah River is Georgia Power Company's Vogtle Electric Generating Plant (VEGP), which consists of two Westinghouse pressurized water reactors.

5.10.3 Emergency Planning Zones

Offsite emergency planning zones (EPZs) have been established to facilitate emergency planning, preparedness, response, and dissemination of emergency information to the public. There are two EPZs defined for SRS; Plume Exposure Pathway EPZ and Ingestion Exposure Pathway EPZ.

The plume exposure pathway EPZ is defined as the area where the principal sources of exposure would be total body exposure from the gamma radiation emitted by a plume and the radioactive materials which it may deposit on the ground, and exposure from inhaling radioactive gases and/or materials in the plume. The plume exposure pathway EPZ defines the area where provisions for prompt protective actions (i.e., sheltering or evacuation) may be required.

The ingestion exposure pathway EPZ is defined as the area where the principal source of exposure would be from ingesting contaminated water or foods such as milk, fresh vegetables or fruits, or aquatic organisms, and from animals who ingested contaminated feed.

Onsite emergency planning, preparedness, and response has been established for each facility/area and is not based on specific EPZs.

5.10.4 Types of Emergencies

5.10.4.1 Description of Postulated Emergencies

The types of credible occurrences which could cause the declaration of an emergency at an SRS reactor include:

- Industrial accidents;
- Accidental releases of radioactive or toxic materials to the environment;
- Damage or potential damage to facilities by extreme external phenomena such as earthquakes, tornadoes, floods, high winds, or forest fires; and
- Security-related events (e.g., bomb/terrorist threats, sabotage, unauthorized entry onto SRS property, extortion, hostage situations, loss or theft of Special Nuclear Materials [SNM])

Chapter 2 of this document provides a detailed analysis of credible emergencies for SRS reactors.

5.10.4.2 Detection of Emergency Conditions

Each SRS reactor is equipped with a variety of means to warn the operating staff of abnormal conditions so that appropriate action can be quickly initiated. Emergency procedures provide reactor operators with guidance for rapid recognition and classification of emergency conditions.

5.10.5 Emergency Classification and Notification

5.10.5.1 Emergency Classification System

The SRS emergency classification system uses four different event classifications which indicate an increasing level of severity. The four classifications, in order of increasing severity, are:

- Unusual Event
- Alert
- Site Area Emergency
- General Emergency

This classification system is the same as that used in the commercial nuclear industry.

5.10.5.2 Notification and Coordination

The decision to declare an emergency is made by on-shift supervision in the affected area(s) or by the Emergency Duty Officer in the Technical Support Center, depending on the nature and/or location of the incident. The communications staff in the Technical Support Center performs the required offsite notifications to State and local authorities and the DOE Headquarters (HQ) Emergency Operations Center (EOC), and activates the SRS emergency response organization (ERO), as appropriate.

5.10.5.3 Information to be Communicated

A standard notification form is used for initial and followup notifications to State and local authorities, VEGP, and DOE HQ. This form was developed jointly with representatives of the emergency management agencies from the States of Georgia, South Carolina, and North Carolina, and VEGP, and contains all information critical to offsite authorities for the protective action decision making process. This form is completed by the TSC staff and transmitted initially by voice and later by facsimile.

Information concerning an emergency at VEGP is transmitted to SRS using the same form. TSC personnel receive these notifications and determine appropriate actions.

5.10.6 Responsibilities

5.10.6.1 On-Shift Response Capability

Due to the nature of SRS operations, a considerable amount of on-shift capability exists for response to an SRS emergency. The SRS operating and security contractors each maintain a large on-shift staff which contains virtually all of the expertise necessary to deal with any SRS emergency.

5.10.6.2 Emergency Response Organization

Upon declaration of an SRS emergency, additional response personnel are activated to provide supplemental resources and logistical and strategic planning support to the on-shift staff.

For an efficient, integrated response to emergencies, the SRS emergency response organization is organized along five functional lines.

(1) Operations

(2) Health, Safety & Environmental

(3) Safeguards & Security

(4) Offsite Interactions

(5) Management Oversight

The SRS emergency response organization consists of personnel from DOE and the operating and security contractors. The specific makeup of the ERO varies depending upon the type and/or location of a given emergency. ERO personnel are selected according to their specific background, experience and qualifications and receive function specific training on their emergency responsibilities.

There is one primary individual and two alternate individuals assigned to each ERO position.

5.10.6.3 Local Offsite Assistance to SRS

Memorandums of Understanding are maintained with local offsite agencies for onsite and offsite emergency medical services, fire fighting, and law enforcement support.

Site specific training is offered annually to offsite agencies and personnel who may respond to SRS emergencies. Also, these offsite agencies routinely participate in SRS emergency drills and exercises.

5.10.6.4 Coordination with Participating Government and Private Agencies

There are also several offsite organizations which are participants in the SRS Emergency Plan. Each agency has specific responsibilities for response to various SRS emergencies. These organizations are:

- Federal Bureau of Investigation
- Department of Energy Headquarters
- Federal Emergency Management Agency
- South Carolina Department of Health and Environmental Control
- South Carolina Emergency Preparedness Division
- Georgia Department of Natural Resources, Environmental Protection Division
- Georgia Emergency Management Agency
- Georgia Power Company (Vogtle Electric Generating Plant

- Aiken County (South Carolina) Disaster Preparedness Agency
- Allendale County (South Carolina) Disaster Preparedness Agency
- Barnwell County (South Carolina) Disaster Preparedness Agency
- Burke County (Georgia) Emergency Management Agency

5.10.7 Emergency Response Measures

5.10.7.1 Activation of the ERO

The TSC is designated as the point of contact for onsite and offsite emergency notifications. Each area or facility and/or Vogtle Electric Generating Plant will immediately notify the TSC of any event(s) that cause declaration of an emergency.

For emergency declarations of Alert or higher, the SRS emergency response organization is activated. For incidents falling below the Alert classification, the DOE duty officer is notified and will determine the appropriate level of response.

The primary method of notifying members of the ERO of an emergency is by pager. ERO members have pagers which can be activated individually or in groups by the TSC. When notified, each ERO member will call the TSC for further information and instructions. The TSC communications staff monitors the return call list to verify that key personnel have been notified. In the event that personnel assigned to a key position have not responded to the page notification, one of the alternate individuals assigned to the position will be notified.

5.10.7.2 Assessment Actions

During an SRS emergency which involves an operating area or process, unique and specific indicators (e.g., system pressures, flow rates, temperatures, radiation and contamination levels, release rates, etc.) are used to assess the extent of the problem and determine necessary corrective or mitigative actions. These indicators are monitored continuously throughout an emergency so that changing or degrading conditions may be indicated early enough to allow preventive measures to be implemented.

5.10.7.3 Corrective Actions

Each SRS facility has established procedures for responding to emergencies. These procedures address every aspect of emergency response specific to the facility for which they are written. Subjects addressed in these procedures include:

- Recognition of abnormal events
- Emergency classification
- Steps for safe, orderly shutdown of operations
- Steps to mitigate or terminate releases of radioactive or hazardous materials to the environment
- Damage control
- Emergency command and control

5.10.7.4 Onsite Protective Measures

Each SRS facility has established procedures for protecting site personnel during emergencies. Subjects addressed in these procedures include:

- Personnel evacuation or sheltering
- Personnel accountability
- Use of protective equipment and supplies
- Contamination control measures
- Emergency exposure authorization and control
- Decontamination

5.10.7.5 Medical Transportation

SRS maintains ambulances for transporting ill or injured individuals. Ambulances are available on a 24–hour basis. If necessary, the helicopters maintained by the SRS security contractor may be utilized for transporting seriously ill or injured individuals.

5.10.7.6 Medical Treatment

Rescue team members in each area are trained in rendering first aid to ill or injured individuals. Additionally, first aid stations maintained in each site operating area and each SRS ambulance is staffed by qualified Emergency Medical Technicians. Where appropriate, area first aid stations are staffed on a 24–hour basis.

During normal working hours the SRS medical facility in the 3/700 area is staffed by a full complement of nurses and physicians. This facility has the capability for emergency surgery and treatment of radiologically contaminated or irradiated individuals.

5.10.8 Emergency Response Facilities and Equipment

5.10.8.1 SRS Emergency Response Facilities

Several facilities are maintained at SRS for response to emergency.

Each operating facility at SRS has an Operations Support Center (OSC) which is activated at the Alert level. The purpose of this facility is to serve as a central point for assembly and coordination of essential facility personnel in supporting the facility operations staff during an emergency.

The Technical Support Center (TSC) is staffed on a 24-hour basis by an Emergency Duty Officer and a communications staff. This staff is augmented by technical and management personnel upon declaration of an Alert level or higher. The purpose of the TSC is to relieve the affected facility operations personnel of peripheral duties, such as off-site communications and dose assessment, so that they can focus their efforts on immediate emergency response needs.

The Emergency Operations Facility (EOF) is activated at the Alert level or higher to relieve the TSC of all functions not directly related to onsite emergency response activities or emergency notifications to State and local officials, DOE headquarters, or the Vogtle Electric Generating Plant. The primary purpose of the EOF is to focus on the long-term ramifications of the emergency and recovery/reentry planning. A backup EOF, located in Aiken, South Carolina, is available for key emergency management personnel in the event that the EOF is not accessible during an emergency.

The Joint Information Center (JIC) is activated at the Alert level or higher to provide information about the emergency to the public.

5.10.8.2 Offsite Agency Emergency Facilities

Offsite Federal, State, county, and private organizations which could be impacted by an SRS emergency have established facilities from which their emergency operations are conducted.

5.10.8.3 Communications Equipment

Numerous methods of communication are available at SRS for use during emergencies. Currently, SRS emergency communications capabilities include:

- Emergency Notification Network (ENN)
- Direct line Selective Signalling (SS-1) telephones located in the TSC, each facility control room, and each area Central Alarm Station
- Direct lines between the TSC and EOF
- STU-III secure telephones
- Extensive radio communication equipment
- Site telephone system

5.10.8.4 Onsite Medical Facilities

First aid stations are maintained in each site operating area and have the facilities, supplies, equipment, and personnel to treat most types of injuries or illnesses.

The STS Medical Department maintains specialized onsite facilities for treating injured individuals who have radiological complications.

5.10.8.5 Emergency Monitoring Equipment

During an emergency, Health Protection personnel are assigned to provide evacuee monitoring, and inarea and inplant monitoring.

Each SRS facility maintains equipment for monitoring all effluents. The effluent monitoring program utilizes continuous air monitors and thermoluminescent dosimeters at the site perimeter and at a distance of 40 kilometers from the site. Additionally, a program is in place for monitoring water from area streams and rivers, ground water, soil, vegetation, drinking water, and plant and animal foodstuffs.

Meterological data is available from a system of eight towers located on SRS and from a 1000–foot television tower located 15 kilometers from the site boundary. Real time data from this system is available in the Weather Central Analysis Laboratory and the Technical Support Center. Additionally, data gathered by the National Weather Service is readily available during an emergency.

If a release of radioactive materials to the environment occurs, monitoring teams can be dispatched to onsite and offsite locations. These teams are equipped with instruments capable of monitoring for all types of radioactive materials which can originate from SRS or the Vogtle Electric Generating Plant.

The SRL Tracking Radioactive Atmospheric Contaminants mobile laboratory (TRAC van) is available for use during an emergency. The TRAC van is capable of measuring low levels of specific airborne radionuclides as they move beyond the site boundary.

The Weather Information Display System (WINDS) is available in each facility control room and the Technical Support Center for performing hazard assessments in the event of a release of radioactive or hazardous materials to the environment.

Mobile and fixed laboratories are maintained at SRS which have the capability to perform radiological analysis of environmental samples, thermoluminescent dosimeters, and bioassay analyses.

Through DOE headquarters, SRS can access the sophisticated resources available throughout DOE to assist in an emergency. These resources include:

- Aerial monitoring aircraft
- Equipped field monitoring teams
- Mobile laboratories
- Radio, microwave, and satellite communications systems

- Power generators
- Data analysis vans

Other Federal agencies such as the Environmental Protection Agency can provide extensive field monitoring and laboratory assistance.

The Radiological Assistance Plan (RAP) van is maintained to provide radiological assistance to facilities within DOE Region 3, as part of a multi-agency Federal response to requests for assistance. The van may be used for additional radiological monitoring and assessment capabilities.

5.10.9 Maintenance of Emergency Response Capability

5.10.9.1 Emergency Plans and Implementing Procedures

Emergency plans and implementing procedures are maintained for each SRS facility. Controlled distribution is maintained for all emergency plans and implementing procedures. Controlled sets of these documents are maintained at each emergency response facility.

The approval of DOE is required for all SRS emergency plans and implementing procedures.

All SRS emergency plans and implementing procedures are reviewed annually and revised as necessary.

5.10.9.2 Training

Initial emergency preparedness training and annual refresher training is given to all persons assigned to the SRS ERO. Training courses include an overview of the site emergency preparedness program and position specific training. All personnel not assigned to the ERO are given a brief overview of their expected actions during emergencies.

The SRS emergency preparedness training program is revised periodically to incorporate programmatic improvements, changes to plans and procedures, lessons learned from drills and exercises, results of appraisals, and student feedback.

5.10.9.3 Drills and Exercises

A program of drills is an integral part of the SRS emergency preparedness training program. Drills are conducted frequently in all site areas to develop and maintain emergency response capability.

A comprehensive exercise program has been implemented at SRS to test the overall capability to respond to an emergency.

Exercises are conducted annually for each major operating SRS facility. These exercises may include participation of offsite federal, State, and local emergency response organizations.

The SRS emergency exercise program is designed to test all aspects of the site emergency response capability within each five year period.

Drill and exercise scenarios are based upon actual postulated emergencies for the involved facility and are designed to emphasize realism of emergency response activities. Simulation of response activities is kept to a minimum. Scenario information is not provided to participants prior to drills and exercises.

Each individual assigned to the SRS ERO is required to participate at least annually in a drill or exercise.

Documentation for each SRS drill and exercise includes purpose, objectives, participants and their level of participation, scope, scenario, evaluation criteria, ground rules, and logistical/administrative information.

A formal critique is conducted following each drill and exercise. Deficiencies and recommendations for improvement are documented for evaluating and upgrading the emergency preparedness program.

5.10.9.4 Review and Update of the Plan and Procedures

SRS emergency plans, implementing procedures, and memoranda of understanding are reviewed annually and updated as required.

Reviews of the emergency plans and implementing procedures are performed by persons who are not directly responsible for administering the emergency preparedness program or the response activity being reviewed.

5.10.9.5 Maintenance and Inventory of Emergency Equipment

Routine surveillance programs have been implemented for all emergency facilities, equipment, and supplies.

DOE performs periodic field surveillances to ensure that emergency facilities, equipment, and supplies are maintained in a sufficient state of readiness.

5.10.9.6 Verification of Emergency Information Subject to Frequent Change

Emergency information that is subject to frequent change (e.g., personnel assignments, telephone numbers, etc.) is verified on a quarterly basis and revised as necessary.

5.10.10 Records and Reports

5.10.10.1 Records of Incidents

All persons assigned to the SRS ERO are responsible for maintaining records, documents, log sheets, and data records pertaining to an emergency. Upon termination of an emergency, all such records are retrieved and stored in accordance with the requirements of DOE 5500.7A, Vital Records Protection Program.

5.10.10.2 Records of Preparedness Assurance

Records of readiness assurance are maintained in accordance with the requirements of DOE 5500.7A, Vital Records Protection Program. These records include:

- Emergency preparedness training
- Drills, exercises, and related critiques
- Inventory and locations of emergency equipment and supplies
- Maintenance, surveillance, calibration, and testing of emergency equipment and supplies
- Agreements with offsite support organizations
- Reviews and updates of emergency plans and implementing procedures

5.10.11 Recovery

Upon termination of an emergency, the ERO is deactivated, but some level of emergency response facility staffing is maintained to plan and monitor recovery activities. EOF and TSC management will determine an appropriate level of staffing for the recovery effort.

5.10.12 Compliance With Community Right-to-Know Act

A list of hazardous materials produced, used, or stored at SRS is provided to the State emergency management agencies of South Carolina and Georgia for distribution to local emergency planning committees.

CHAPTER 6.0
SRS REACTOR SAFETY EVOLUTION

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SRS REACTOR SAFETY EVOLUTION
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SRS REACTOR SAFETY EVOLUTION

6.1 INTRODUCTION

The SRS production reactors are complex facilities involving hazardous materials and processes, operating over an extended period of time. As with any such facility, the applicable safety-related technologies and standards evolve over time. Thus, hardware and operations change over the life of the sites. This section discusses the evolution of safety-related aspects of the SRS reactors; past, present, and future.

6.2 REASONS FOR CHANGE

There are several different reasons that changes are made to the hardware and operations. The most significant is when new information is developed indicating that a particular hazard is greater than was previously believed. This new information may be developed by the SRS operating staff, by others in DOE or its contractor organization, or by unrelated individuals. An example of this occurred in 1958, when the Atomic Energy Commission's Advisory Committee on Reactor Safeguards (ACRS) concluded that:

> "The buildings in which the SR reactors are housed do not possess any significant containment features, such as those now being provided for power reactors located in more populated areas. In the event of a serious accident that would breach the reactor tank and shield, the building shell in itself could not be expected to provide a third line of defense of any consequence on restraining the volatile fission products."

It was recommended "that the (operating contractor) explore alternative paths toward obtaining a higher degree of confinement than is now in effect."

The combination of internal and external review led to a significant increase in safety studies. Primary proposals for partial containment included building sealing and exhaust air filtration systems. Projects were initiated to provide for both.

A second cause for change is development of new technology which can be used in plant hardware. The most obvious example of this is the development of computers and advanced telecommunications which allow rapid processing of large quantities of data. This has allowed upgrading of several control, protection, and operator interface elements of the SRS reactor sites.

A third cause for change is changing standards. While the SRS reactors are not subject to regulation by the Nuclear Regulatory Commission (NRC), DOE applies NRC and commercial industry standards to the extent practicable, recognizing that there are substantial differences between commercial nuclear plant technology and SRS reactor technology. An example of this type of change is the upgrading of operator training currently underway. Over the past several years, commercial nuclear plant operators have upgraded their training and certification requirements under the guidance of the Institute for Nuclear Power Operations (INPO). DOE and its operating contractor have arranged to work with INPO, and are well into a multi-year program that will bring SRS training programs into compliance with applicable INPO guidelines.

A fourth cause for change is occurrence of a significant event at one of the SRS reactors, with subsequent root-cause analysis which conclude that changes are appropriate. Although there has never been an accident at an SRS reactor which resulted in significant radiological consequence, there have been numerous minor incidents. DOE, with its contractor(s) conduct investigations of each significant incident. If the investigation results indicate that action should be taken to prevent reoccurrence of the same or similar events, the change is implemented. A recent example is when unexpected reactivity behavior was encountered during a reactor startup. The actions taken by site operators resulted in a mild over-power event. As a consequence, changes have been made to administrative control, requiring additional DOE and contractor approvals to continue operations if reactivity anomalies are encountered.

DOE's orders require that the SRS operating contractor conducts investigations, maintains awareness of NRC standards, and considers appropriate changes to hardware and operations. Thus, the safety evolution which has occurred since 1953 will continue throughout the life of the production reactors.

6.3 SUMMARY OF CHANGES

A summary of significant changes is given below for each of the operating periods (Figures 1 through 4).

6.3.1 1953 - 1960

Increased product demand led to major improvements in heat removal systems and fuel assembly design, with resultant power gains of about five times the original design power.

A backup shutdown system was installed to permit addition of a liquid neutron absorber.

The first major external review (by the Atomic Energy Commission Advisory Committee on Reactor Safeguards) recommended an improved confinement system.

6.3.2 1960 - 1965

A confinement system was installed to provide for control of liquid and airborne contamination.

Additional redundancy was provided in the addition and source flow capability of the emergency cooling system.

Online computers were installed to compare operating temperatures with limits, and to give rod reversal action.

6.3.3 1965 - 1970

Changes were made in the emergency cooling system (ECS) to improve reliability and to protect against a light water addition accident.

The SRS reactors were compared to "70 Criteria" for licensed reactors, with favorable results.

Reactor buildings and systems were analyzed for seismic resistance: a set of seismic criteria was developed.

6.3.4 1970 – 1975

Improvements were made to the ECS and sump pumps to reduce the probability of flooding the D_2O pump motors if the ECS was activated.

An automatic incident action (AIA) system was installed to activate the ECS for major loss of cooling incidents.

An automatic backup shutdown (ABS) system was installed to inject liquid neutron absorber automatically in less than a second.

6.3.5 1975 – 1980

Safety computers were installed to provide scram protection from low flow and high temperatures in fuel assemblies.

Redundant assembly cooling sources were added to the discharge machine.

SRS operation was reviewed against TMI lessons learned.

6.3.6 1980 – 1985

An electrical load-shedding system was installed to reduce the potential for damage to onsite power generation equipment if power to the site were to be lost.

A diagnosis of multiple alarms (DMA) system was installed to aid operators in analyzing abnormal reactor conditions such as process water or cooling water leaks.

Reactor operator/supervisor training programs (e.g., the reactor training simulator) were strengthened as part of TMI follow-up activities.

6.3.7 1985 – 1990

A fourth ECS injection (top addition) system was installed.

A moderator recovery system (MRS) was installed to cope with small D_2O leaks and preclude the need for ECS activation.

The remote monitoring and control system (REMACS) was installed to provide improved remote control capability. Seismically qualified panels are being installed.

Several comprehensive appraisals were conducted by outside agencies, aimed at evaluating SRS reactor safety.

A full-scale control room simulator is used for operator training and certification.

The initial results of the PRA are available.

Evaluation of seismic capability is being upgraded. Related upgrading of hardware seismic resistance includes replacement of the Central Control Room ceiling, installation of new emergency lighting, and installation of a seismically initiated trigger (0.05g) for the secondary shutdown system (SSS).

Improved fire detection capability is being provided in the reactor building, along with installation of a standpipe to facilitate local fire fighting efforts.

A comprehensive program of tests and analysis is being conducted to establish safe power limits for the reactor.

Automatic trips for the circulation water (CW) pumps are being installed to allow more time for operator action in response to a CW pipe break.

New, more sensitive tritium monitors are being installed in the reactor building to enable earlier detection of a response to small process water leaks.

6.3.8. 1991 and Beyond

The following actions are presently planned in the near future. Most, if not all, will be completed as indicated here. However, some project may be modified, replaced, or deferred as a result of new technical or programmatic information.

Flood control pumps in the lower level of the reactor building will be replaced and/or supplemented to improve reliability and increase flow rate. This will provide additional protection against loss of reactor coolant circulation due to flooding of the reactor coolant pump motors.

A new computerized control system will be provided for charge and discharge operations. This will reduce the likelihood of a misloading error.

An integrated set of upgrade will be implemented for the reactor electrical distribution system (REDS). The upgrade includes new seismically gratified diesel generators, additional fuel and lube oil supplies, improved power distribution system, and a new load sequencer.

Other projects contributing to the evolution of reactor safety will be identified as time passes.

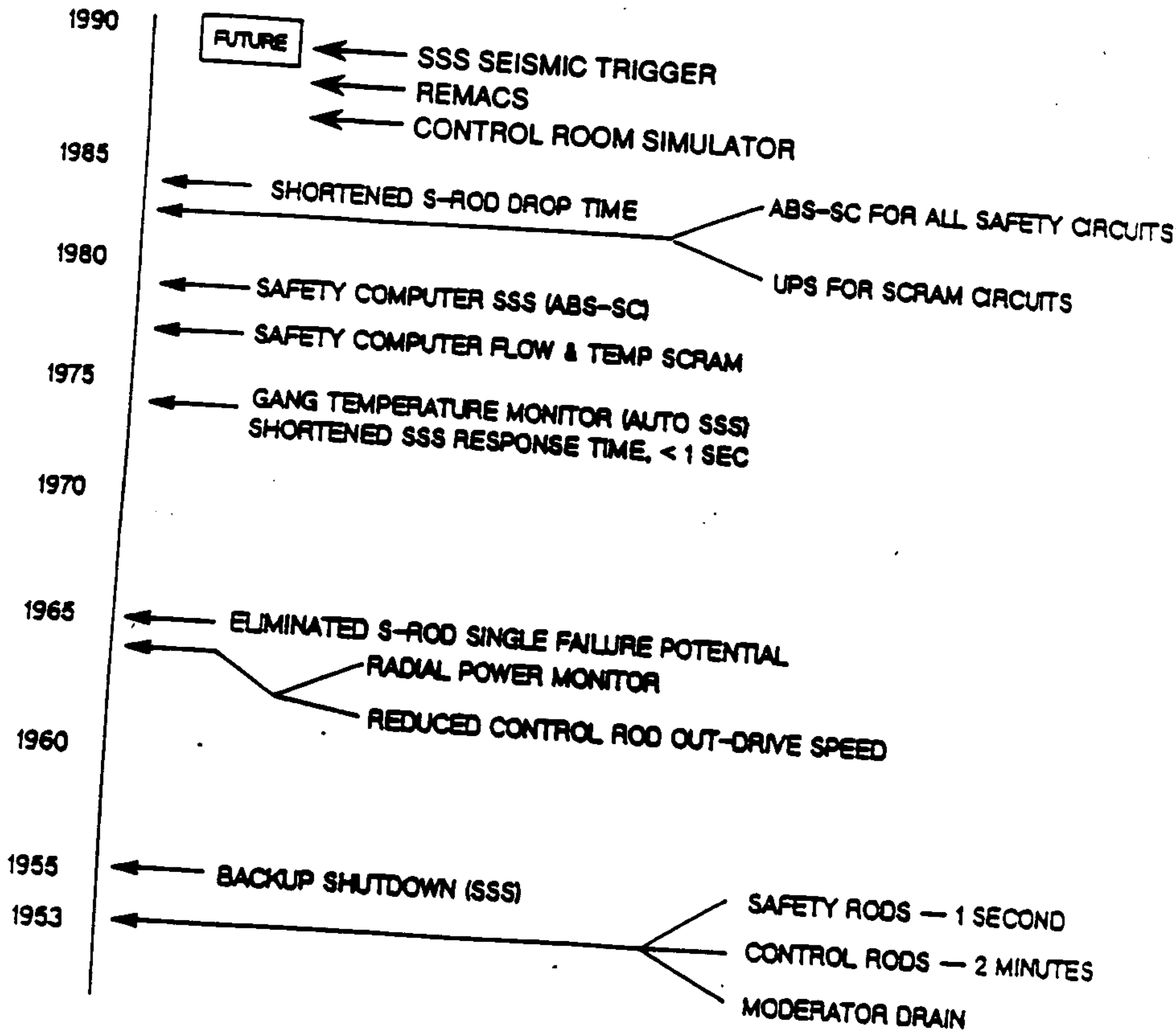

Figure 6-1
Control and Protection Systems Safety Evolution

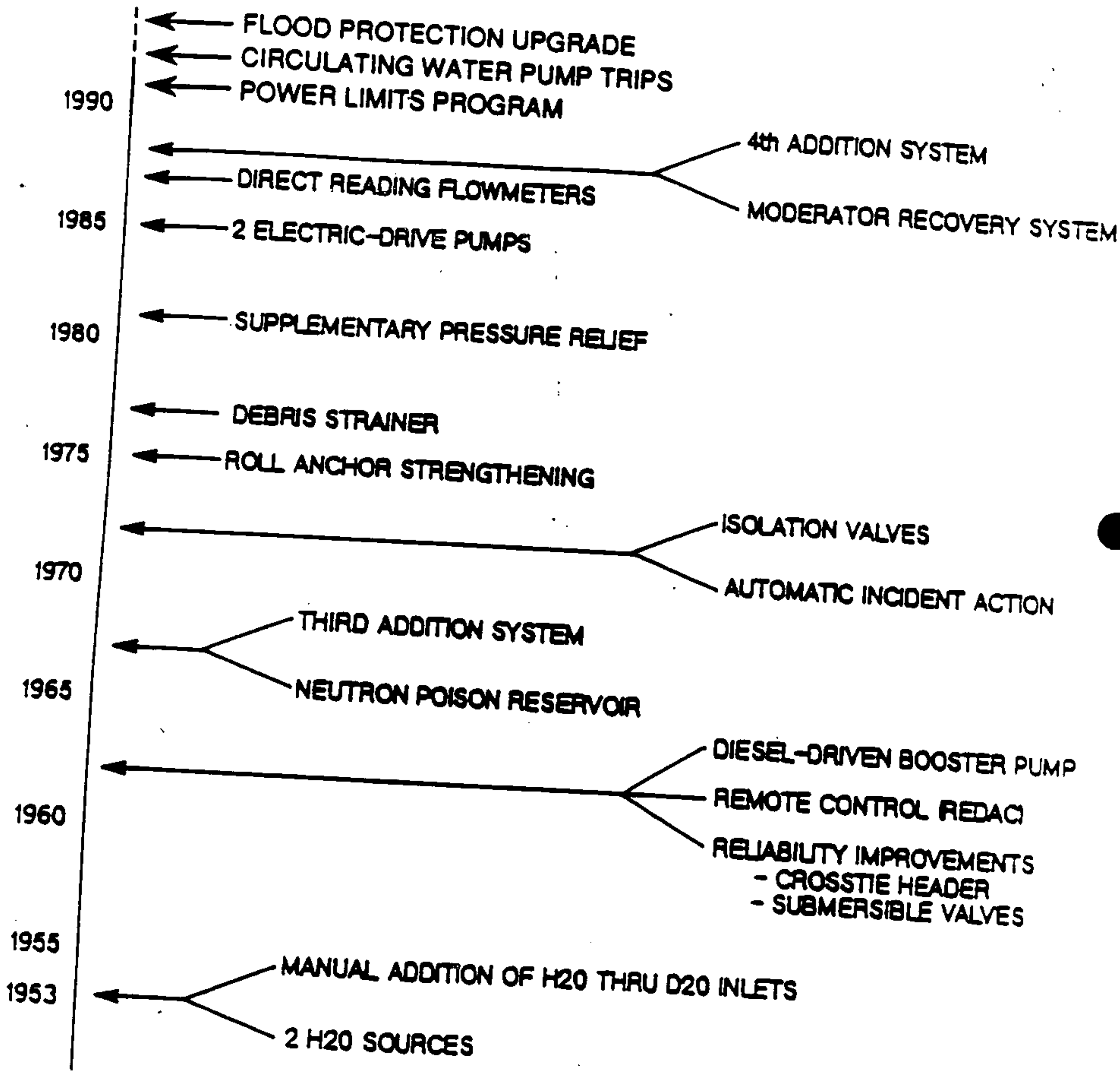

Figure 6–2
Emergency Cooling System Safety Evolution

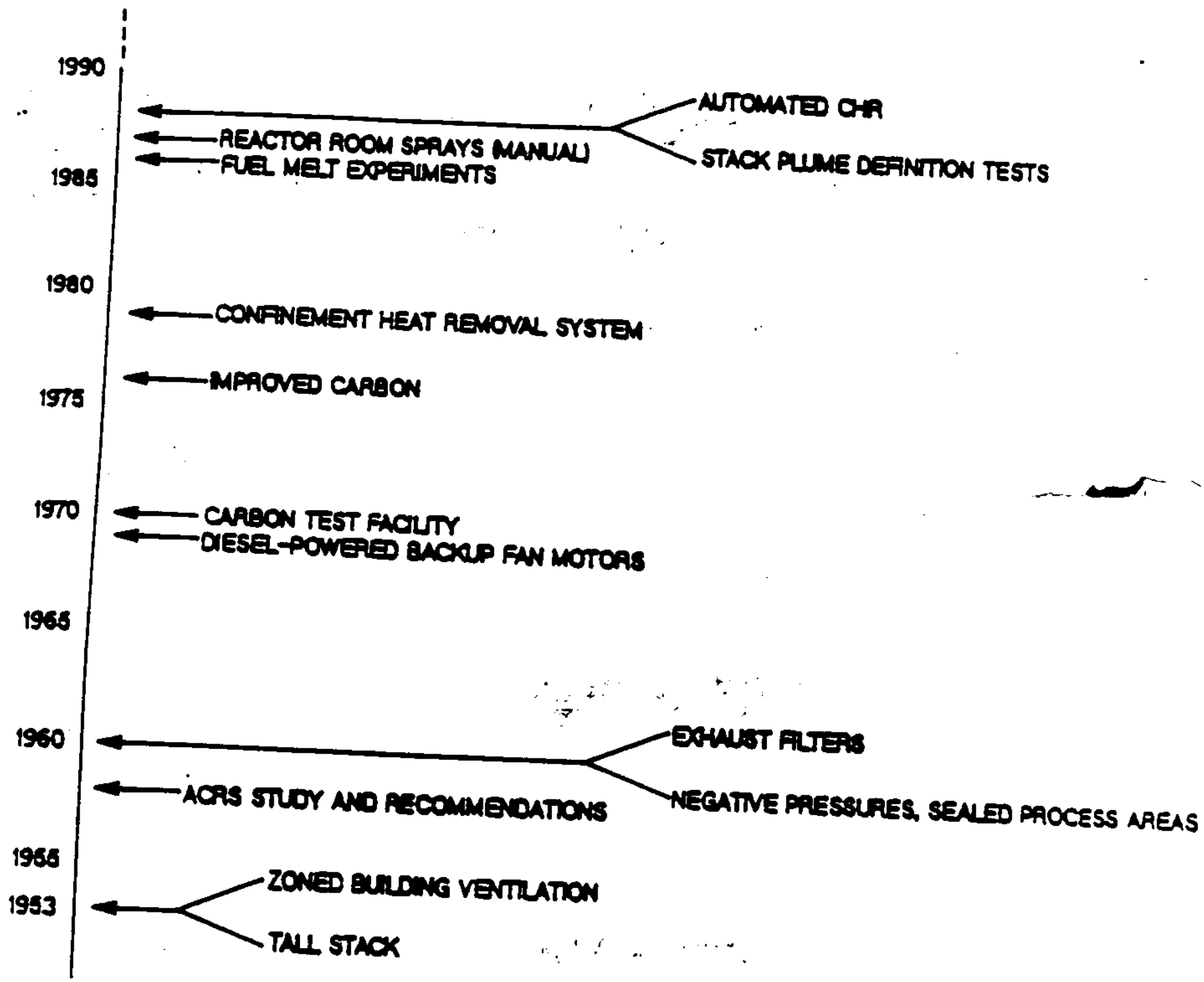

Figure 6–3
Confinement System Safety Evolution

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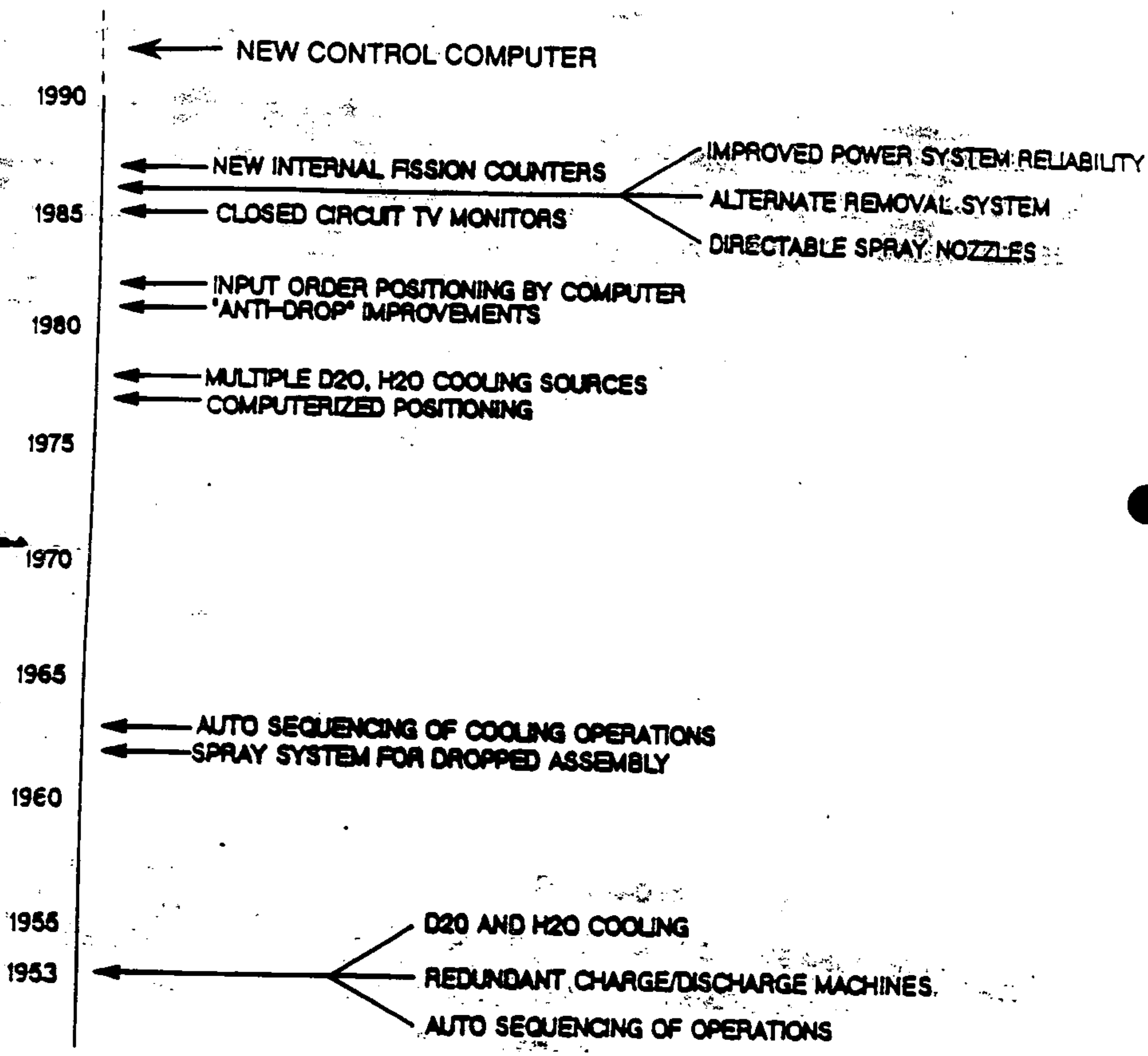

Figure 6-4
Fuel Handling Safety Evolution

Form SR-161
(Rev 9-89)
SR 5650.2A
(SR Form P&ME-1 and earlier copies of this form are obsolete)

Mary Allen

U.S. DEPARTMENT OF ENERGY
Savannah River Operations Office

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