APPENDIX J

SRP REACTOR SAFETY EVOLUTION*

INTRODUCTION

The Savannah River Plant reactors have operated for over 115 reactor years without an incident of significant consequence to on or off-site personnel. The reactor safety posture incorporates a conservative, failure-tolerant design; extensive administrative controls carried out through detailed operating and emergency written procedures; and multiple engineered safety systems backed by comprehensive safety analyses, adapting through the years as operating experience, changes in reactor operational modes, equipment modernization, and experience in the nuclear power industry suggested. Independent technical reviews and audits as well as a strong organizational structure also contribute to the defense-in-depth safety posture. A complete review of safety history would discuss all of the above contributors and the interplay of roles. This appendix, however, is limited to evolution of the engineered safety features and some of the supporting analyses.

The discussion of safety history is divided into finite periods of operating history for preservation of historical perspective and ease of understanding by the reader. Programs in progress are also included.

The accident at Three Mile Island was assessed for its safety implications to SRP operation. Resulting recommendations and their current status are discussed separately at the end of the appendix.

SUMMARY

Operation of the Savannah River reactors began in 1953 with a conservative design, automatic shutdown systems, and detailed written procedures. As reactor safety technology advanced at Savannah River and in the U.S. nuclear industry, as modernized equipment (e.g., computers) became available, and as operating experience and comparison to industry standards suggested needs for change, projects were undertaken to upgrade and supplement existing safety systems.

Confinement

Original control of airborne radioactive releases was by dispersion via a tall stack. Confinement features were added beginning in 1960 to cope with the very low probability accidents that could release radioactive materials. The features include filtration of all the ventilating air leaving the reactor building using moisture separators, particulate filters and halogen adsorbers (carbon).

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*This appendix, in its entirety, is derived from: Rankin, D.B., 1983. SRP Reactor Safety Evolution, DPST-83-718, E. I. du Pont de Nemours and Company, Savannah River Laboratory, Aiken, South Carolina.

Emergency Cooling and Liquid Activity Containment

The original design provided for emergency manual addition of light water to the reactor core with retention of the water in a dedicated tank after it leaves the building. Improved sources of light water and a common pressurized addition header were added in the mid 60s. Automatic emergency cooling was provided beginning in 1973 with many additional system improvements toward increased reliability being made over the years. Larger contaminated water removal pumps and increased storage capacity were added beginning in 1973.

Computer Monitoring and Control

Safety of reactor operation was enhanced beginning in 1964 by computer monitoring of critical process conditions and in 1970 by computer control of reactor operating and shutdown systems. Updated redundant control computers replaced the original ones in 1978 and redundant "safety" computers were installed for assembly temperature and flow scram protection.

Automatic Emergency Shutdown

The original instrumentation provided numerous monitoring circuits which could actuate the safety rods to drop if prescribed limits were exceeded. Safety rod system improvements over the years provided diverse relay logic and paths for scram signals, utilizing both AC and DC power sources. A backup shutdown system, the Supplementary Safety System, was added in 1957 to provide for manual injection of a liquid neutron poison in the event the safety rods failed to drop. The Gang temperature monitor automated this system in 1974 to be actuated upon sensing of very high temperatures. The Safety Computers were programmed beginning in 1979 to back up scrams with SSS actuation, providing protection for all postulated transients.

Seismic Protection

Seismic scram protection was provided in 1955. As a result of earthquake and building structural analyses, seismic bracing was added beginning in 1976 to protect the susceptible building structures and emergency cooling system piping from postulated credible earthquakes.

Fuel Handling

Fuel handling operations improvements in the late 70s and 80s have equipped the charge and discharge machines with computer positioning and misload protection as well as assembly identification capability. Automatic cooling for all irradiated assemblies was provided to the discharge machines.

Programs In Progress

Programs are currently in progress to assess possible improvements to the confinement system, emergency cooling system and fuel handling equipment and to provide additional seismic analyses for critical systems. In addition, Savannah River Laboratory performs continuing research and development in all areas of reactor safety. And their detailed review of plant operations and programs provides a strong independent safety overview as outlined in the Nuclear Safety Control Procedures.

In addition, an extensive program is currently in effect to incorporate lessons learned from the Three Mile Island accident into the SRP reactor safety features. Numerous changes have been incorporated into the SRP operations and others are in progress related to this program. Computerized diagnosis of multiple alarm situations and availability monitoring of critical equipment are two of the programs in progress.

Figure J-1 presents safety milestones from initial startup to present.

DISCUSSION

1953-1960

The original design concept of the SRP reactors envisioned the prevention of accidents by backup equipment, comprehensive instrumentation, detailed written procedures, well-trained personnel and strong technical backup. Additional safeguards were provided by an isolated site to protect neighboring people, by reactor buildings designed to resist pressures from external blasts, and by exhausting the building air through a high stack for increased dispersion of hazardous airborne contamination.

The earliest formal safety analysis report for SRP reactors was issued in April 1953; inherent reactor hazards and postulated accident scenarios were discussed and the initial facility design features protective of such accidents and mitigative of the consequences were presented.

Original design features directly related to reactor safety (and maintained in similar form to present) are listed below. These formed the "building blocks" for later improvements.

- Safety rod system - Sixty-six cadmium safety rods (79 in C-Reactor) are provided in 1-inch positions interstitial to fuel positions throughout the reactor. The rods are suspended just above the reactor core and will reach their full IN position and shut down the chain reaction about 1 second after a scram demand signal. Original circuits capable of initiating safety rod (scram) action are discussed later.

- Control rod system - Sixty-one control rod bundles (73 in C-Reactor), housing seven control rods each, occupy 4-inch lattice positions in the central portions of the reactor. A combination in each bundle of half-length and full-length (effective portions) rods provides for detailed axial and radial flux control, thus minimizing hot spots and the resultant challenge to assembly cladding and providing for overall power optimization. The control rods are capable of shutting down the reactor and maintaining it subcritical, independent of the safety rods.

- Cooling system - 6 D_2O coolant pumps powered by 3000 HP a.c. motors and backed up by 120 HP d.c. motors originally provided the capability of circulating about 78,000 gpm of assembly coolant. The online (via gear reducers) d.c. motors provided 24,000 gpm capability to remove shutdown decay heat. About 67,000 gpm of Savannah River water provided cooling for the primary loop (D_2O) through 6 heat exchangers.

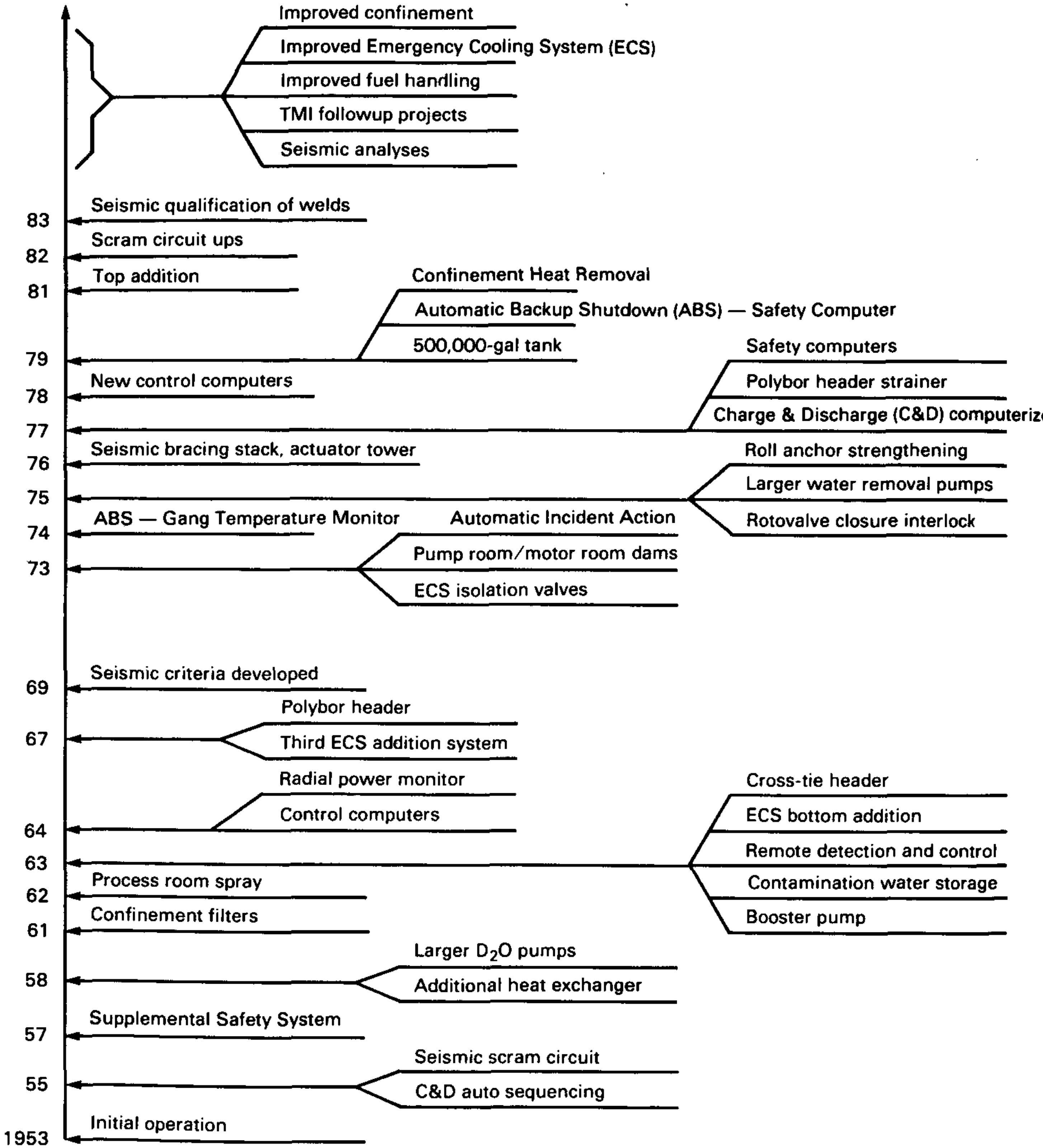

Figure J-1. Reactor safety milestones.

- Shielding - More dedicated to personnel safety than reactor safety, top, bottom and thermal circulating deionized water shields were provided for radiation shielding and removal of heat from neutron and gamma attenuation. Seven feet of concrete immediately surround the thermal shield. The walls, floors, and ceilings of the pump rooms, motor rooms, and heat exchanger rooms are constructed with a minimum of 4 ft of concrete for shielding.

- Instrumentation - Instruments primarily concerned with reactor safety are those that monitor neutron flux in the reactor and cooling system parameters such as activity, temperature and pressure at various points in the system. The original "action-oriented" instrumentation was divided into 4 modes:

  (1) Scram I - energized the safety and control systems, causing reactor shutdown within about 1 second.

  (2) Scram II - energized the scram circuit on the control rod system only, causing reactor shutdown within about 2 minutes. More rapid shutdown was not deemed necessary for these events to justify the thermal transient caused by safety rod action.

  (3) Reversal - caused control rods to be driven into the reactor in the normal sequencing pattern. This slow reduction in power allowed time for correction of less threatening problems with possible return to standard operation without shutdown.

  (4) Annunciators - provided audible and visual indication of abnormal signals. Items causing scram or reversal action as well as numerous process variables were annunciated.

  Original items monitored for scram and/or reversal action included:

    - Neutron flux level
    - Reactor period
    - Temperature of D_2O effluent from assemblies
    - D_2O plenum pressure
    - D_2O pump motor failure
    - Heat exchanger H_2O cooling water flow low
    - Shield flow low

- Fuel Handling

  A charge machine is provided to charge fuel from the assembly area presentation point to the reactor; the discharge machine removes heat-generating irradiated assemblies from the reactor and transports them to the disassembly area canal. All operations are conducted in air. The charge and discharge machines and associated equipment were provided several safety features to prevent erroneous loading and irradiated assembly overheating. Protection included:

  (1) Redundancy of charge and discharge cranes to be able to handle one another's functions.

(2) Ability to discharge housing (with assembly) quickly should sticking of an assembly occur.

(3) D_2O and H_2O coolant supplies to an assembly if discharge required longer than 2 minutes.

(4) Protection against charging to a position already filled.

(5) Anti-collision devices for the machines.

(6) Remote position indicators to operators in the crane control room.

Accidents Considered

The 1953 safety analyses hypothesized two basic accidents: 1) a "boiling accident" caused by complete stoppage of D_2O and H_2O flow by external forces (e.g., earthquake) and 2) a prompt criticality. Offsite doses were calculated to be less than the lethal range for the boiling accident with 10% fission product inventory release. At that time, no credible mechanism for a prompt criticality was determined to exist. Accident scenarios considered for dose calculation were expanded by 1956 to include fuel loading errors, the loss of D_2O circulation accident by system failures, and prompt criticality from startup accidents. Maximum offsite dose from the loss-of-circulation accident were calculated to be in the lethal range.

Administrative Controls

In addition to safety circuits and systems provided in the original facility, a set of detailed written operating instructions governed each step in the process of charging, startup, operation, shutdown and discharge of the reactor. Emergency procedures were provided to respond to conditions necessitating a reactor shutdown. Technical Standards, based on Technical Manual specifications, prescribed limits of operation. Test Authorizations were provided for any changes in operating mode.

Emergency Cooling

In response to postulated loss of D_2O flow incidents, a system was provided for manual addition of H_2O to the reactor. Cooling water lines were provided, with manual activation through remotely operable gate valves.

Seismic

A seismic scram circuit was installed in 1955 to ensure reactor shutdown in the event of an earthquake.

Backup Shutdown Systems

The original safety feature for terminating a power transient if the safety rod system failed was via a moderator dump to dedicated storage tanks. Manual actuation of H_2O addition following the moderator dump would have provided for decay heat removal. The Supplementary Safety System was installed in 1957 to provide for manual reactor shutdown via a liquid neutron poison injection should the safety rods fail to drop during a power rise transient.

The year 1958 was a significant one in both production-oriented facility changes and increased safety effort. New, larger D_2O pumps were installed to increase flow capacity to about 150,000 gpm. An additional heat exchanger was installed in each of the pumping systems in parallel with the existing exchanger. The resultant reactor power increase (to approximately 2000 MW) brought higher individual assembly powers and reactor fission product inventory and thus increased concerns on reactor safety issues. Accidents considered increased to include power surge mechanisms, the "cold-water accident" because of the negative temperature coefficient, loss of coolant, and single assembly melting during operation or discharge. Technical studies produced limits on the heat flux of individual assemblies to prevent cladding "burnout." Tests were proposed to determine the disposition of melted fuel; and reactor containment was first considered.

Fuel Handling

In 1955, primary operations of the charge and discharge machines were provided with the capability to automatically sequence through most steps in the operations without manual input after each step. This "auto cycle" feature significantly reduced the amount of operator input necessary and thus increased system reliability.

External Review

An integral part of the SRP safety philosophy is review by external technical experts. The Atomic Energy Commission Advisory Committee on Reactor Safeguards (ACRS) performed an extensive review in 1958 and concluded:

> "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 Du Pont Company explore alternative paths toward obtaining a higher degree of confinement that 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.

1960-1965

Confinement System

Containment of fuel melt releases continued to dominate the safety considerations during the early 60s. In 1960-1961, a major improvement project provided moisture separators, particulate filters, and halogen adsorbers (carbon) in the process area ventilation exhaust stream to remove airborne contamination, particularly I^{131}. A backup motor with independent power supply was added to each exhaust fan. Figure J-2 shows the ventilation/confinement system arrangement. The reactor room and process areas were sealed to minimize leakage

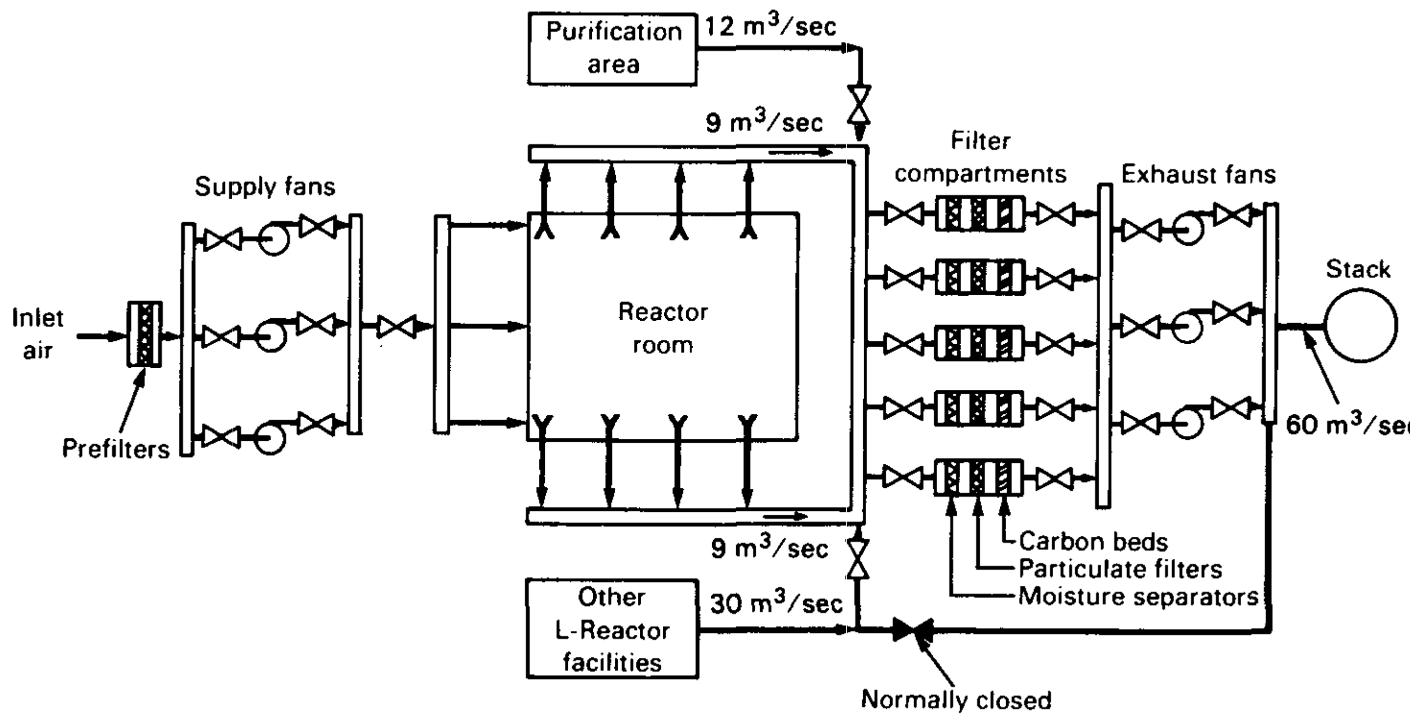

Figure J-2. Flow diagram of reactor ventilation system.

outside the confinement path. The improved system was expected to retain 99% of released particulate activity and 99.9% of the halogens.

ECS

The emergency cooling system received considerable upgrade in 1962-1963. Light water addition lines were tied together by a common "cross-tie header" to facilitate maintaining a pressurized line for monitoring system availability. Piping was redesigned to provide addition through either the top addition or the bottom addition. Submersible addition valves (89-299 series) were installed to maintain addition capability should the -20 ft, -40 ft building area flood. And the booster pump was added to increase the source flow capability and number of sources of light water.

The Remote Detection and Control (REDAC) system was provided to be able to monitor system parameters remotely and add light water if necessary after an area evacuation was required. Also, a 50-million gallon earthen basin was provided to contain contaminated water that would result from a loss of cooling or loss of circulation accident with ECS actuation.

Computers

Reactor monitoring and consequently safety was enhanced in 1964 by providing computer monitoring of critical process conditions. A GE-412 computer was installed in each control room for improved reactor temperature monitoring, calculation of the proximity to cladding burnout and better operating analysis. The computers were provided with control rod reversal capability if assembly operating temperatures increased to 1°C above prescribed thermal-hydraulic operating temperature limits. This reversal capacity prevented minor excursions from reaching the point where scram circuits were challenged.

Fuel Handling

A spray system was added to the reactor room in 1962 for cooling an irradiated assembly if dropped on the floor during discharge operations. The system consists of a header with 12 groups of fixed spray nozzles mounted on the reactor room wall. The spray pattern from the nozzles covers the area traversed by the discharge machine.

Many improvements were added to the charge and discharge machines in 1963. The ability to separately hold (chuck) both components of an assembly having separately dischargeable components ("double chucking") and a sleeve to hold down one while discharging the other were added. These features decreased the possibility of dropping components or of discharging one unintentionally with the other.

Emergency cooling for assemblies during discharge was enhanced by providing automatic sequencing to the secondary source if the primary source fails. Also, the distribution of flow to an assembly was improved by providing a better path into the top of the assembly.

An emergency shear was provided for cutting off the top portion of failed assemblies to facilitate enclosure in a failed assembly container (HARP).

Electrical Distribution System

The original supply of power to building equipment was distributed from four transformer rooms located within the reactor building. A fifth distribution station, known as the containment substation, was added in the early 60s and placed above ground level outside the building. This station will provide power to critical equipment in the event that flooding of the below-grade elevation disables the normal supply.

General

The number and type of accidents analyzed was expanded during this period to include anticipated transients without scram (ATWS). As a result, reactor operating restrictions on temperature and power were imposed in 1964. The radial power monitors (RPMs) were installed to provide scram protection for cluster (group of 6) and gang (group of clusters) temperature increases. And the maximum control rod drives speed was reduced to limit the power increase from a rod driveout accident. Explosive valves were added to the ink injection system to increase the reliability and speed of injection should the safety rods fail to drop.

Reactor power was increased several hundred MW in 1963 when operations with 5 PSIG blanket gas were established. R-Reactor operation was discontinued in 1964 due to reduced product demand.

1965-1970

ECS

In 1967, the bottom addition system was moved so that three systems now had addition capability. This increased capacity was designed to limit core damage to less than 1% for the worst-case loss of coolant accident as well as to provide additional redundancy. Also, it was recognized that the cold H_2O which the ECS would add to the reactor could produce, for current charges, a positive reactivity transient which may override the effect of safety rod insertion. So a 20,000 gallon storage header was added in line with the crosstie header and filled with a neutron poison solution (2% polyborate), sufficient to preclude a positive reactivity transient from either necessary or unwanted ECS actuation.

Seismic

The blast criteria used in the original design did not necessarily provide for the effects of earthquakes. The buildings were very resistant to external forces but their response to dynamic effects was not specifically analyzed at the time of the original design. The reactor building structures were analyzed in 1969 for their response to seismic criteria developed by Dr. George Housner of the California Institute of Technology. The structural analysis was made by John A. Blume and Associates of San Francisco. These independent consultants recommended a set of seismic criteria for the design and analysis of retrofitted SRP facilities. Reactor area buildings and associated systems and equipment were then classified according to their required seismic resistance. The classification applies to those features of each system that are essential to protect the public health and safety. The results of the studies concluded that remedial strengthening of the actuator towers and exhaust stacks was necessary to comply with the criteria. Resultant projects are discussed later in this report. Process water piping was determined to be adequately resistant to seismic stresses.

Other

Because the SRP reactors operated before formalized nuclear industry guides and standards existed, building design and construction did not necessarily conform to the later criteria. The first comprehensive, documented comparison of SRP reactors to licensed reactors was issued in 1967 after the AEC had issued a set of 70 criteria for licensees. Exact comparison with the criteria were made where system characteristics were similar; where exact comparison was not practical, an effort was made to define the intent of the criteria with respect to the SRP facilities and discuss compliance as appropriate. It was concluded "that the SRP reactors meet the overall intent of the 70 Criteria and, in most cases, meet the literally interpreted criteria."

L-Reactor operation was discontinued in 1968 due to decreased product demand.

1970-1975

Confinement System

A new carbon test facility was placed in operation for evaluating aging effects on carbon and other filter components.

ECS

Isolation valves were installed (1973 in P and K, 1974 in C) in each of the four ECW sources converging into the common polyborate storage header. This isolation limits the volume of water available to the storage header to prevent flooding of D_2O pump motors in the event of header failure. Such a failure without isolation would both produce the need for ECW addition and defeat the protection. The isolation valves are normally closed and would be opened automatically upon ECS actuation.

Also to preclude flooding of circulating pump motors from any cooling water leak or ECS actuation, four new sump pumps (2 rated at 4500 gpm; 2 at 2500 gpm) were installed in 1975 to replace the original pair of 2000-gpm pumps. The 2500-gpm pumps are submersible. The total 14,000 gpm removal capacity is capable of removing all ECS water even if all three addition systems were on-line.

A new hydro-starter was installed on the booster pump (one of the four ECW sources) after a history of minor starting problems. The original electric starter was maintained as a backup.

Remote start capability for the main cooling water pumps was provided to the central control room in 1973. These pumps could originally be shut off remotely to control a large cooling water leak. This ability to start the pumps provided greater assurance of ECW supply if light water addition were necessary during reactor shutdown while some of the pumps were off-line.

Thirty-six inch high dams were installed in the -40 ft level between the pump room and motor rooms in 1973-1974 to prevent a D_2O leak with ECS actuation from causing flooding of the D_2O pump motors. A one-way (motor room-to-pump room) gate was installed in the dam to allow flow from a postulated cooling water line break to reach the pump room sump pumps and take advantage of their removal capacity. The gate capacity was designed to match the sump pump capacity.

Automatic Incident Action

The capability for light water addition in the event of a large D_2O leak was automated beginning in 1973. The M-2 Automatic Incident Actuation (AIA) consoles contain electronic logic circuitry which utilizes two-out-of-three vote logic from in-reactor level sensors to determine the need for ECS actuation. Three types of liquid-level sensors, including absolute-pressure sensors, differential-pressure sensors, and conductivity cells, are used to provide protection from common mode or common cause sensor failure mechanisms. The system is self-bypassed for leaks of less than 1800 gpm.

System functions initiated by Incident Action originally included the following:

- Starting of the booster pump
- Opening of all ECW header isolation valves and supply valves for the two top addition systems
- Energizing of bottom addition supply valve circuitry to open on level control
- Set up of the ventilation/confinement system for optimum effectiveness

An interlock circuit was added in 1974-1975 to provide automatic closure of rotovalves (to prevent backflow of ECW through heat exchangers) in the two top addition system.

Roll Anchor Modification

An analysis of emergency cooling water hydraulic forces indicated that strengthening of the roll anchor stands for the plenum inlet lines was required in the three reactor areas. If ECS water were added to a full reactor tank through both top addition systems concurrently, the pressure generated under the top shield could have caused failure of the roll anchors. Roll anchor modifications were completed about 1975.

Computers

During 1970-1971, the online computers were modified to perform closed-loop control of the reactors in addition to the monitoring and rod reversal functions. Direct control was accomplished by adjusting control rod settings with stepping motors to control overall power level and selectively move rods for most effective power generation within the various reactor regions. The GE-412 computers remained online 99% of the time and controlled power ascension and level power as well as radial and axial flux shapes.

Backup Shutdown System

Supplementary Safety System

Piping modifications were made to reduce the SSS response time (time interval between system actuation and arrival of gadolinium nitrate "ink" in the reactor). The modifications reduced the response time from 6 seconds to 0.7 seconds. Replacement of all SSS explosive valves with an improved design increased system reliability.

Automatic Backup Shutdown

The first of the Automatic Backup Shutdown (ABS) systems was installed in 1974 and called the Gang Temperature Monitor (GTM). The GTM is completely independent of the safety rod shutdown system. It actuates the SSS to inject gadolinium nitrate into the reactor tank when the coolant effluent temperatures of selected assemblies exceed prescribed limits. Specific incidents for which the ABS-GTM can provide protection are the gang (control) rod withdrawal and the total loss of AC pumping power without scram. The ABS-GTM provides a diverse reactor shutdown channel, and therefore, increases the overall reliability of reactor shutdown protection. No mechanism has been identified that could cause a common mode failure of both the safety rod and ABS GTM systems.

Seismic

The large process heat exchangers are mounted on railroad-car-type wheels and were recognized as susceptible to movement from seismic activity. The exchangers were braced in 1974 to preclude such movement which could damage the attached cooling or process water piping.

1975-1980

Confinement System

Type GX-176 coimpregnated (potassium iodide and triethylene diamine (IEDA)) carbon was installed in all three reactor confinement systems in 1976 to replace Type 416 unimpregnated carbon. Type GX-176 carbon retains organic iodides better than Type 416.

Studies begun in the early 1960s recognized the threat of overheating of the carbon filters from airborne fission product particles in the event of extensive core melting. Such overheating would seriously reduce the iodine-retention capacity of the carbon and even cause desorption of the collected iodine. The studies and possible solutions were refined through the years and culminated in 1979 with installation of the Confinement Heat Removal (CHR) System. The CHR system is designed to flood the building -40 ft level pump room floor with water in the event of a full-core meltdown in which molten core breaches the tank bottom and is deposited on the floor. The water would maintain air temperatures low enough to prevent failure of the confinement system filters. The water is supplied from the disassembly basin through two redundant pneumatic valves. The system is manually actuated from a dedicated console in the central control room when alarms indicate (1) both a large airborne activity release and reactor tank bottom temperature greater than 232°C at one of three reactor positions where heat sensors are located or (2) temperatures greater than 232°C at any two of the three reactor positions containing heat sensors. The heat sensors are in dedicated monitor pins in three blanket (outer ring) positions in the reactor.

The CHR system design provides for both automatic and manual actuation of the pneumatic valves. Initially, only the manual mode was made operational. Future plans call for activation of the automatic mode after reliability studies are completed.

ECS

Prior to 1977, the river water supply line was considered the primary source of emergency light water cooling supply because of its independence from the 186 cooling water supply basin and pumps. However, flushing tests in 1977 identified significant debris (leaves, sticks, clams, etc.) which could become lodged in reactor assembly flow channels and reduce the cooling capability. So this line was valved off as a primary source and assigned backup status. In 1978, the maintenance pressure for this line was reduced to 20 PSIG and the source is available as a last choice if the booster pump and both cooling water inlet headers fail.

The recognition of a debris problem prompted installation of a debris strainer in the polyborate storage header common to all ECS supply sources. Two redundant bypass check valves around the strainer are designed to open if debris pluggage of the strainer causes the strainer delta P to exceed prescribed limits.

Contaminated Water Storage Facilities

A 500,000 gallon storage tank and related piping were added to the 50,000,000 gallon contaminated water earthen storage basin beginning in 1979. Following a loss of coolant or loss of circulation accident with ECS actuation the first 60,000 gallons (approximate D_2O system capacity) will now flow to Building 106 and the next 500,000 gallons to the new storage tank which is vented back through the building ventilation/confinement system. Any remaining flow is diverted to the 50,000,000 gallon basin. These facilities assure that all expected radioactive releases from credible accidents would be contained within the filtered system. Figure J-3 shows the contaminated water removal facilities.

Computers

Control Computers

The GE-412 process control computer in each area was replaced beginning in 1978 with two new Interdata-M70 computers (Reference 4). These computers, which
TE are currently in use, perform basically the same functions as the GE-412 (e.g., monitoring of process data and performing closed-loop control operations). The two machines have identical capacity, but only one performs the primary functions at a given time; the other provides secondary data processing functions while in a standby status. Either is capable of performing all required on-line functions in the event one becomes unavailable.

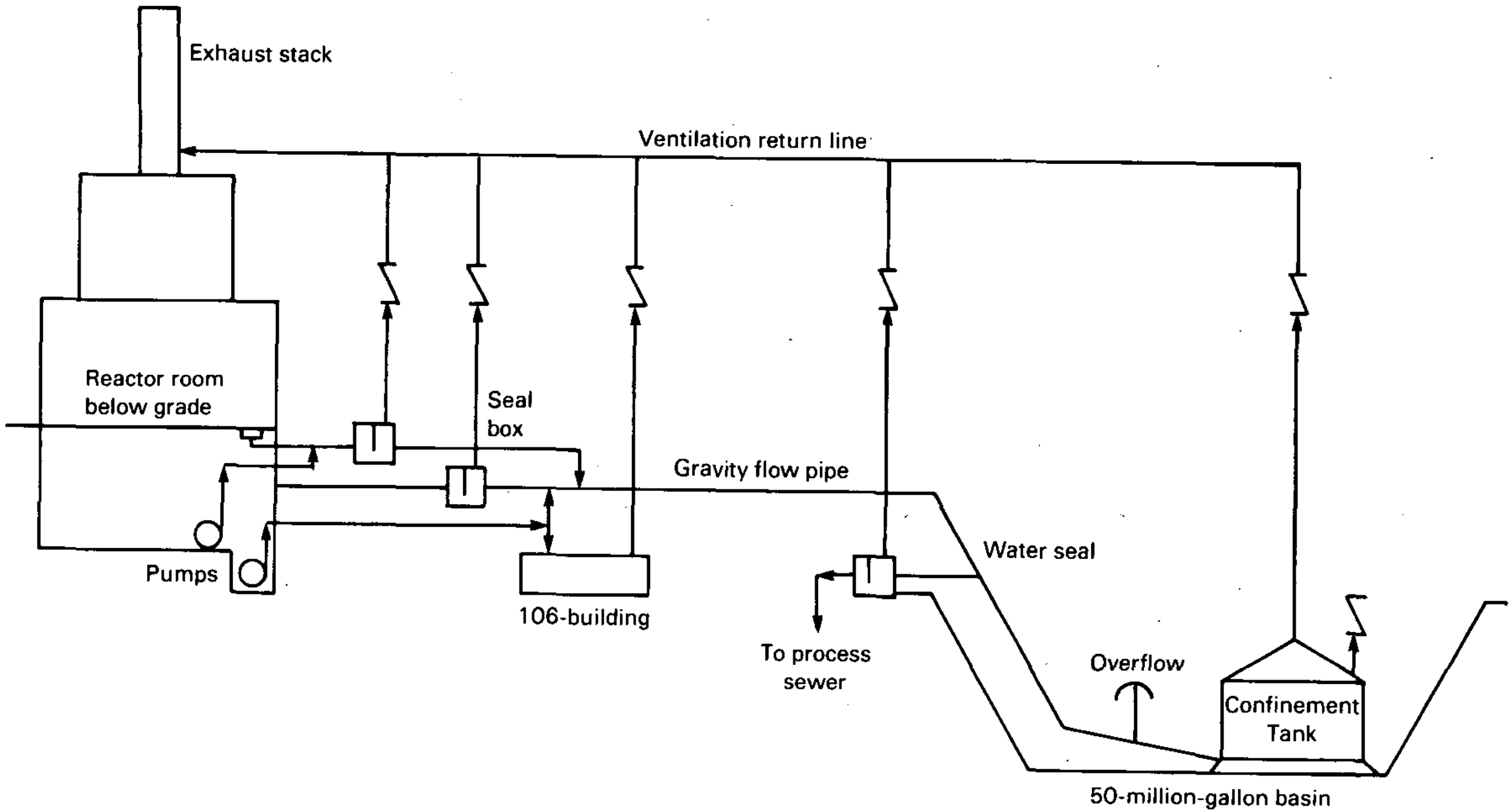

Figure J-3. Confinement tank.

TE| Improvements over the GE-412 monitoring include:

(1) Faster scan of temperature input signals (approximately 30 seconds compared to the previous five minute scan period).

(2) Monitoring continuity and continued control capability when one control system fails (only secondary data processing functions will be interrupted).

(3) Improved input-output capability and more effective communication with reactor operator through cathode ray tube (CRT) displays.

(4) Increased reliability (dual, redundant computers and improved methods for both hardware and programming fault detection).

(5) Future applications (extra memory capacity for automatic alarm analysis and monitoring during reactor shutdown).

Safety Computers

In addition to the new process control computers, dual safety computers were installed in each area beginning in 1977 to provide scram protection from low flow or high temperatures in reactor assemblies. Prior to this, scram protection for high temperature was provided by the radial power monitor (RPM, installed 1964), which monitored the average temperature of groups of six assemblies versus prescribed scram setpoints. Flow protection had been provided by the Flow Monitor, a pressure switch safety circuit which shut down the reactor if any assembly monitor pin delta P dropped below prescribed setpoints. The safety computers offered the following major improvements over the RPM and Flow Monitor:

(1) Overtemperature scram protection for all individual reactor positions rather than only clusters in the central 60%.

(2) Rapid scan time. All temperatures are scanned each 0.36 second. Flows are scanned each 0.15 second.

(3) Reduced dependence on manual operations such as setpoint adjustments and instrument bypass.

(4) Ability to reject spurious signals.

(5) Increased reliability through dual redundant computers, frequent internal operability testing, etc.

(6) Replacement of obsolete flow sensors and instrumentation.

(7) Future applications - capability for different levels of protective action, such as liquid reactor poison injection in addition to safety rod scram (ABS-SC, discussed below).

Automatic Backup Shutdown Using Safety Computers (ABS-SC)

Although the ABS-GTM provides diverse protection for certain anticipated transients without safety rod scram action, it was recognized that faster time response and more comprehensive protection was desired. The capability was added to the safety computers beginning in 1979 to back up the safety computer scram relays and supplement the safety rod scram action with automatic injection of the SSS poison (ink) if certain adverse conditions were detected. If reactor assembly effluent temperatures continue to increase or do not decrease sufficiently following scram initiation, ink will be injected. The system takes advantage of the existing redundancy of the Safety Computers since each computer can independently take ABS action.

An external review by qualified consultants in the industry confirmed the effectiveness of ABS-SC in coping with anticipated transients without safety rod scram action.

Fuel Handling

The charge and discharge machines were equipped beginning in 1977 with computerized positioning and fuel and target position identification. This greatly reduced the possibility for inadvertent criticality which might be caused by placing a fuel (^{235}U) in a target (^{238}U) position.

The assembly cooling system on the discharge machine was improved in 1978. Two independent sources of D_2O and two of H_2O were provided with automatic sequencing through their predescribed hierarchy. The supply for the sources was improved and in-line filters provided, and two methods of directing the water to the top of the assembly were provided. Monitoring of system flows and pressures was updated and expanded.

Seismic

Improvements were made beginning in 1976 to meet the seismic requirements for resistance to maximum predictable earthquakes. Projects included:

1. Strengthening the base of the actuator towers and eliminating the spring action of the supporting girders. The spring action increases the response of the tower to dynamic forces.

2. Strengthening the building exhaust stack.

3. Improving the lateral support for the emergency cooling system (ECS) piping and the supplementary safety system (SSS) piping.

1980-1983

ECS

Beginning in 1981, the bottom addition system was converted to a top addition system. Addition valves in each area were updated to a more reliable design.

Automatic Backup Shutdown

ABS-SC protection was expanded in 1982 to provide for backup ink injection for all scram circuits. New digital inputs inform the computers of the current status (bypassed, online, actuated) of all other safety circuit relays. When any safety circuit calls for scram action, the safety computers back up the signal using their own "echo scram" relays. Then an immediate assessment of reactor temperature conditions determines if ink injection is required to achieve sufficient reactor shutdown.

Uninterruptible Power Supplies for Scram Circuits

Uninterruptible power supplies (UPS) were provided to all scram circuits in 1982 and to the Safety Computers (including ABS logic) in 1983. The UPS will maintain the computers and scram circuits online a minimum of 5 minutes following loss of all offsite power.

Fuel Handling

The chuck fingers on the machines were lengthened in 1981 to assure the assembly would be held in the mast if the chuck released the assembly and it rested on the rinse water collection pan beneath it.

A computer system was added in 1982 to control the order of charging of the assemblies, further eliminating manual input and increasing protection against criticality from misload errors.

Seismic

A weld examination program was completed in 1983 to establish the quality of carbon steel piping welds in the cooling water and ECS systems. Samples from over 100 welds in the L-Area system piping were destructively examined and quantitatively characterized as to quality. Also, a large number of additional welds were radiographed and inspected by an outside consultant. The analyses concluded that the quality of the welds in these systems meets the structural requirements of a design basis earthquake.

Studies in Progress

Confinement

The program toward continuous study of improved confinement of radioactive releases includes efforts in several areas:

- Experiments are in progress to determine the effectiveness and feasibility of using solid adsorbents (mordenites, zeolites) for adsorption of noble gases. The program to assess technical feasibility and economic practicality continues with high priority.

- Fuel melt experiments with irradiated SRP fuel samples are planned. The goal is better characterization of the source term to be expected from a fuel melt accident; analyses of the Three Mile Island damaged core material suggest the current assumptions may be vastly overconservative. An improved source term may greatly reduce offsite dose expectations.

- Studies are continuing to evaluate several confinement/containment system design alternatives. Cost estimates and measures of effectiveness are being developed for tall stacks, internal building containment, a containment dome, and a system for temporary holdup of airborne contamination to allow for decay of short-lived products before filtration/release.

ECS

The primary design basis accidents for the ECS are the very large cooling water or process water leaks or the loss of D_2O circulation. Concepts are now being developed for coping with smaller leaks which may propagate to a need for ECS actuation. Ideas include increased D_2O makeup capacity or collection and recirculation of the D_2O leakage. Recirculation of the leakage and/or ECS water back to the reactor inlet lines would reduce the amount of contaminated water which exits the building and provide a non-exhaustive supply of cooling water.

A second booster pump has been proposed and basic data written to provide an improved source of ECS water.

Fuel Handling

Two safety improvements packages for the charge and discharge machines are scheduled for installation in 1983-1985. The projects will provide:

- Expanded computer control of C&D operations
- A directable spray nozzle for cooling a dropped assembly
- Closed circuit television camera monitoring to enhance visual observation of C&D operations
- Improved supply of power for the machines with better handling facilities for the moving cable system
- Assembly temperature monitoring capability
- Better personnel access to the machines and controls to facilitate routine or emergency maintenance

Seismic

Studies to date have provided seismic analyses of and necessary bracing for important systems and building structures. A continuing program is in progress to:

- Complete assessments of the overall integrity of the CW, PW, and ECS piping with respect to weld and pipe failure characteristics
- Complete a stress analysis of the piping systems to define maximum points of stress and magnitudes during a design basis earthquake

- Define the spectrum of piping leaks and consequences thereof
- Establish acceptability bands for seismic safety margins
- Estimate seismic risk
- Retrofit bracing, etc., if necessary to bring risks within acceptable range

Three Mile Island Followup

The March 28, 1979, accident at Three Mile Island (TMI) had significant implications for nuclear power generating facilities. Because the SRP reactors are operated at essentially atmospheric pressures, and because of the absence of the auxiliary systems for electric power generation, a similar incident is not possible at Savannah River. However, there are many lessons to be learned in the areas of operator response, technical personnel availability, instrumentation adequacy, and accident propagation. To take advantage of any possible lessons, a committee was formed on April 4, 1979, to assess implications of the TMI incident for SRP operations. Recommendations were formulated for any changes in operation or improvement programs indicated by lessons apparent from the TMI experience. Three areas in which changes were indicated are discussed below along with the status of each.

Technical

The TMI implications for the technical arena were assessed by SRL and SRP personnel and led to acceleration of major projects already being developed in the areas of alarm diagnosis and system availability monitoring. Other changes of less magnitude are also discussed below.

Diagnosis of Multiple Alarms (DMA)

The Diagnosis of Multiple Alarms (DMA) system is a pioneering application of computer-based diagnosis of malfunctions in operation. This system, developed at SRL-SRP, is designed to aid operators in managing abnormal reactor conditions by automatically analyzing patterns of alarms and sensor inputs. During 1982, the installation of the DMA system was completed in all SRP reactors. Work on this system is continuing with the development of closed-circuit television systems to make possible visual recognition and assessment of leaks in reactor cooling systems. Four cameras are currently installed for evaluation in the P-Reactor process areas.

The heart of the DMA system is the alarm logic tree. These are similar to fault-trees developed in process hazard analysis work. Simple alarm trees define the general problem. More complex alarm trees pinpoint the location within the plant. There are currently 45 logic trees to recognize conditions that could lead to the loss-of-coolant or loss-of-circulation accidents. Any diagnosis is displayed on a large television in the control room. The message defines the root cause of the alarms and identifies the correct emergency procedure to be used.

Essential Equipment Monitor (EEM)

Also beginning in 1982, the Essential Equipment Monitor (EEM) was installed in all areas. The EEM will continuously monitor ECS valves and other essential equipment and give an immediate indication of a majority of electrical failures. A programmable controller in the central control room will scan fault sensing circuits and initiate an alarm for such failures as an open or bad connection, loss of ground, short to ground, or loss of voltage. EEM monitoring will significantly decrease critical equipment unavailability from these common electrical problems and help ensure compliance with operating requirements. The system is in a test mode in all three operating reactors and will be made operational when checkout and procedures for response to alarms are complete.

Postaccident Monitoring

In response to review of postaccident monitoring adequacy, a project was authorized to provide radiation monitoring equipment with increased monitoring range and life expectancy. Such monitoring upgrade is being provided for both airborne and liquid effluent streams under accident conditions. Assessment of confinement system seals indicates sufficient radiation tolerance for system function for extended accident conditions.

Postaccident monitoring and control improvements also include:

- The reactor remote control facilities are scheduled for computerization and upgrade. Control of an evacuated reactor area will be possible from the existing remote facility as well as from another reactor control room.

- Evacuation signals for reactor buildings are being improved by upgrade of area communications systems. Work is complete in 2 of the 3 operating areas.

Other

Other items included in the technical/design area are:

- Concepts have been documented and are currently being evaluated for improving the ECS design to better cope with small leaks which may propagate to accident conditions. A small flow rate light water addition system or increased D_2O makeup are being considered.

- A preliminary report on mechanisms to retain noble gases from a fuel melt release has been issued. The alternatives are being evaluated in terms of cost and benefit.

- A probabilistic risk assessment (PRA) is being conducted for the reactor safety systems to extend earlier analyses of risks versus all postulated accidents. An outside contractor has begun work on a systematic analysis of the electric power system and other systems analyses will follow.

Many TMI-related technical analyses were completed with no resultant action deemed necessary. Studies continue on various other subjects and status is formally reported to DOE-SR periodically.

Training

Assessment of the Reactor Department training program revealed inadequate documentation of the training methods, material covered, and program format. A detailed description was subsequently written entitled "Reactors Personnel Selection, Qualification and Training Manual." Many parts of the training program have been strengthened as a result of the assessment and are described in the Training Manual.

- Position titles, job descriptions, and responsibilities are well defined and understood.
- Level of knowledge requirements were established and detailed for all certified personnel training and retraining programs.
- Specific tasks and training requirements for certified operators and supervisors are being delineated through POSITION TASK ANALYSES performed by an independent contractor. The Training Manual will include qualification and training requirements identified by these analyses.
- Realistic accident scenarios were developed and documented in training material.
- Procedure writer qualifications were established and included in the manual.

In addition to the better definition and documentation of the training program several other changes are being effected:

- A fifth operating shift was effectively established by providing additional operations staffing to allow certified personnel to be relieved for continued training and recertification.
- The training period was lengthened from 9 months to 1 year to facilitate additional classroom training and control room experience.
- An independent certification board was established to review each operator or supervisor candidate. The board of SRL and SRP technical personnel will review the program and oral and written examination performance to accept or reject the candidate.

Simulator

The most significant addition to the Reactor Department training program will be the building of a reactor control room simulator. An experienced outside contractor, Singer-Link, is developing the computer models necessary to provide real-time instrument response to a variety of postulated accidents. A full-size replica of the K-Reactor central control room will be provided. The control room will look, respond, and sound like the K-Reactor control room.

All controls, instruments, and alarms that could be involved in significant training exercises will be interactive with the computer complex. The few controls, instruments, and alarms that are not interactive with the computer complex will appear normal to the operators.

Most of the instruments in the control room will be simulated in a manner to appear real to the operator, but internally will be designed or modified to be compatible with a standard set of drive signals developed for the simulator computer complex. The control and safety computers will be stimulated (receive the same signals as they do in a real control room) so that future control and safety computer software changes can be implemented in the simulator more easily.

The simulator is expected to be operational by 1985.

Programmatic

Procedures Upgrade

Numerous deficiencies identified in procedure format and standardization resulted in increasing the procedure writer staff three-fold to expedite improvements. Items being addressed include:

- All operating, emergency, and master control procedures were reviewed for relationship to or implications from the TMI experience.
- All procedures are being converted to a standard format to facilitate usage, revisions, and training. This effort is more than 80% complete.
- Bases sheets for all emergency procedures are being developed or updated to provide reference documentation of basic concepts and logic.

Quality Assurance

The Savannah River Quality Assurance (QA) program has been formally documented and implementation continues at a rapid pace. Reactor programs have been affected in several major areas:

- Training in basic QA principles and programs were incorporated into all reactor supervisory training and retraining courses.
- A formal program was implemented to incorporate QA requirements into operating procedures. The system consists of a QA assessment and development of detailed Action Plans based on the results. The Action Plans are implemented through normal procedures. Assessment of existing reactor facilities is complete. Resultant procedure upgrade is about 80% complete.
- Surveillance of reactor operating procedures by an independent QA group was increased. Fifteen audits were conducted by the SRP QA Department (wholly independent of operational responsibility) in 1982. Audits included effectivity of compliance to established written procedures as well as determination of whether findings were generic or isolated.

Other

Other programmatic changes effected since the (TMI) review began include:

- Reactor Incident report reviews were incorporated as a mandatory part of the continued training for certified reactor supervisors and operators. Lessons learned from each incident are reviewed with maintenance, operations, and engineering personnel as appropriate.

- Shift checklists were converted to reactor procedures and formalized procedures for shift turnover were provided.

- A document is being prepared to provide plantwide standardization of tool calibration requirements. Tools will be calibrated proportional to need as determined by frequency, precision, and tolerance demands.

- The commitment of Du Pont to the ALARA (As Low As Reasonably Achievable) radiation dose reduction program is being documented. The document describes drills used as part of the radiological controls and the program for conducting internal audits.

- The Maintenance Information and Control (MIAC) system was upgraded to ensure better control of equipment availability and to improve the ability to detect equipment performance trends.